RFPPC2025
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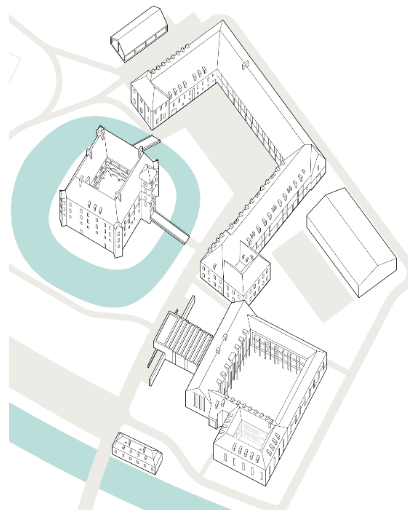

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BOOK OF ABSTRACTS

May, 08, 2025

Review and invited talks

R-1

The ITER ICRF system under the new ITER baseline: latest updates and technological developments

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This paper describes the ITER Ion Cyclotron Range of Frequencies (ICRF) system under the ITER re-baselining and the new ITER Research Plan (IRP). Under the new IRP, the ICRF system will feature one single antenna. During ITER's first operation phase (the Start of Research Operation phase, or SRO), the system will be powered by 4 Radio Frequency (RF) sources for coupling up to 10 MW to the plasma. In this phase, the ICRF system will be used for Ion Cyclotron Wall Conditioning (ICWC) and for testing Ion Cyclotron Heating (ICH) with a full tungsten wall. Based on the outcome of the SRO, the ICRF system could be upgraded to 20 MW for the nuclear Deuterium-Tritium phases of ITER by increasing the number of RF sources and their high voltage power supplies, while keeping a single antenna.

The paper discusses the recent modelling activities for the ITER ICRF system, including the coupling modelling and its thorough benchmarks, the local gas injection modelling and its large beneficial impact on the power coupling, as well as the modelling of the ICRF-specific sputtering. Modelling activities have predicted the compatibility of the ITER ICRF system with high power ICH in the presence of a high Z wall (in particular owing to the system's flexibility in setting the proper antenna excitation law – in amplitude and phase), which will be experimentally validated during the SRO. Additionally, the paper reviews recent modelling of ITER ICH scenarios as well as initial modelling for ICWC. It also illustrates the RF and mechanical design of the antenna and of some of its sub-components, and summarizes recent advancements in full system simulations, including impedance matching analyses.

Finally, the paper reviews the latest advances on antenna protections and arc detection methods (including radar arc detection, sensing of stray microwave radiation, etc.). It also highlights the latest prototyping activities (antenna straps, prototype of the quarter of antenna and its RF testbed, RF windows and their testbed, etc.).

R-2

ICRF boundary-plasma interactions: reflections on progress and challenges

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In the past few decades, the interaction of intense RF waves in the ion-cyclotron range of frequencies (ICRF) with edge plasmas and material surfaces has been a subject of increasing interest. Motivated by the availability of high power ICRF for heating and current drive in fusion oriented devices, this subfield has undergone major advances in understanding brought about through experiments, analytical theory, and computer simulations.

This presentation will reflect on historical and recent progress together with ongoing challenges in RF boundary-plasma theory and modeling. Topics will include RF sheath physics, RF-driven convection, the ponderomotive force, and RF wave propagation in turbulent plasmas.

RF sheaths play an important role in ICRF-surface interactions in fusion devices. Clever experimental techniques have come a long way in reducing the impact of RF sheaths on device performance. Modeling of RF sheaths has progressed on two somewhat separate fronts. On the microscale, or spatial scale of several Debye lengths, capacitive sheath models were first developed, followed by models which included particle as well as displacement currents. The models describe the rectification of RF to DC sheath potentials, and provide an impedance boundary condition for macroscopic full wave codes, enabling a new level of self-consistency in RF wave simulations. This makes improved predictions of sputtered impurity influxes and mitigation strategies possible.

RF sheath-induced convection, arising from the rectified sheath potential and its $\mathbf{E}\times\mathbf{B}$ induced drift, modifies plasma profiles near surfaces and affects transport of both bulk plasma and impurities. High intensity RF waves also affect the plasma through ponderomotive forces that modify plasma profiles and flows. In addition, ponderomotive forces have also been used to stabilize macroscopic instabilities in small plasma devices.

Plasma turbulence modifies the propagation of RF waves, especially propagation through the boundary plasma where turbulence levels can be high relative to background (average) profiles. For ICRF, full wave scattering approaches are often preferred because RF wavelengths exceed or are comparable to the dimensions of blob-filaments that usually dominate edge and scrape-off-layer (SOL) turbulence. Channeling of RF wave energy along plasma filaments has interesting implications for sheath power absorption in the SOL, and for convective cells and SOL transport.

While many of the fundamentals of RF interactions with surfaces and boundary plasmas are understood, challenges remain. These include improving physics model fidelity, better model integration and self-consistency, and an improved understanding of the physics and practical role of RF sheaths at grazing magnetic incidence angles. It is hoped that plasma physicists will continue scientific progress in these areas and further the realization of practical fusion energy production.

R-3

Finite element modeling of RF waves in fusion plasmas: progress in past decades and future role

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This paper reviews the progress of RF wave field simulation, in particular, based on the finite element method (FEM). In order to describe the wave excitation, propagation, absorption, and its interaction with plasma facing components, RF wave simulation needs to handle a broad range of physics processes. FEM provides a rigorous framework to translate physics PDEs to numerical procedure and it has been used for wave simulation from early days. In fact, already in 80s', ICRF wave simulation in the tokamak core on 2D cross-section is seen in the literature. Those works predate the invention of the edge elements and wave propagation/absorption physics was rather simplified. Significant progress has been made since then for utilizing FEM for the wave simulations.

In SOL/antenna, FEM is very successful thanks to its capability of handling complicated geometry. Those regions typically allow for approximating the dielectric response as local response, making the FEM-based discretization straightforward and efficient. Advent of computational hardware and algorithms (Nedelec element, parallel direct solvers, high-order elements, etc.) makes it possible to simulate the wave excitation by geometrically complicated RF launchers installed near the plasma last closed flux surface in 3D space. The SOL/antenna simulations based on FEM became a standard approach to design/analyze RF antenna characteristics. Improvement of these simulations is continuing, evolving them towards more integrated physics simulations including the interaction between background plasmas such as plasma turbulence and plasma wall interactions (RF sheath rectifications).

In the plasma core regions, the spatially dispersive (non-local) nature of dielectric response is much more important and crucial. FEM has been playing a major role in this area. A family of semi-spectral codes (such as TORIC) uses FEM-descritization in the radial direction and incorporates the perpendicular dispersion of dielectric response via the perpendicular differential operators. Such a semi-spectral code can be stitched self-consistently with the FEM-based wave field solution in the antenna/SOL regions. Solely FEM-based approach is an area where active development has been happening from different aspects. The first one is to add iteratively the non-local dielectric current contribution. This approach originates from the LHCD simulation using commercial FEM software, and several efforts have explored extending it to ICRF and other regimes. It is worth mentioning that there are also on-going efforts to include more complete physics of hot plasma dielectric response, including background plasma parameter gradient.

In this presentation, we will review those computational model development efforts from the view point of FEM algorithm application, both exploiting its clear strength and covering weakness. Then, we discuss the experimental validation of simulation models and contributions to realize better RF actuators. Lastly, we will touch remaining issues where further physics model development and algorithm is necessary.

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I-01

Overview of the ICRF heating system in SPARC

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The development of the high-field tokamak approach towards commercial fusion energy relies heavily on the success of net energy gain experiments in the SPARC tokamak [1]. SPARC, currently being built in Devens, MA, USA, by Commonwealth Fusion Systems (CFS), is a high-field, high-density, medium-sized device with $B_0 = 12.2$ T, $R_0 = 1.85$ m, $a = 0.57$ m, $I_p = 8.7$ MA, $n_{e0} \approx 4 \times 10^{20}$ m^{-3}, $T_{e0} \approx 20$ keV and ~10 second pulse flattop duration. Achieving the objective of $Q > 1$ in SPARC's initial operation campaign requires a significant amount of auxiliary heating, supplied exclusively by the Ion Cyclotron Range of Frequencies (ICRF) heating system [2]. Furthermore, a distinct Ion Cyclotron Discharge Cleaning (ICDC) System in SPARC will be used to condition the vessel surfaces and to perform boronisation.

The Radio-Frequency (RF) power generation and transmission system for SPARC is powered by capacitor banks and based on 22 Solid-State RF amplifiers, each delivering up to 2 MW, which are assembled, tested, and commissioned by CFS. The transmitter output signal is tuned to achieve the required 120 +/- 1 MHz bandwidth. The power is distributed between 14 4-strap ICRF antennas arranged in poloidal pairs, of which 5 pairs will be powered for the early SPARC operation phase. Internal device failures, RF arcing, and excessive reflected power will be monitored and mitigated by the ICRF Protection System and a fast controller will maintain the match of preset RF matching stubs via frequency modulation within +/- 1 MHz. The transmission network is largely installed, while most of other components are in production.

Efficient power coupling while minimizing high-Z wall material sputtering will be ensured by tailoring edge plasma conditions, minimisation of image currents by varying the toroidal and poloidal phasing and the field-aligned design of the antenna, which is radially recessed in the limiter shadow. An array of multiple antennas each having a conservative power density requirement of ~ 2 MW/m^2, near or below the level achieved in Alcator C-Mod and ASDEX Upgrade experiments, allows robust operation with relatively low voltage on the antenna and in the transmission lines. Variations in coupled power are analysed across the scenarios considered for the first experimental campaign with full-wave COMSOL modelling. The main heating schemes use a ^{3}He minority and 2nd harmonic T heating at 12 T or a H minority and 2nd harmonic D (3rd harmonic T) heating at 8 T, for both D and D-T plasmas. A window of high single-pass absorption operation is found for the SPARC initial campaign, with the absorption depending significantly on both the minority concentration and the antenna spectrum.

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I-02

Ramp-up and Sustainment Scenarios for Tokamak Energy's Fusion Pilot Plant

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Tokamak Energy is designing a Fusion Pilot Plant (FPP) based on low aspect ratio tokamak for integrated test and validations of technologies, systems and processes required for commercial fusion energy deployment. The FPP, which is targeting start of operations by 2035, will consist of an operationally-relevant fusion environment. By exploiting the inherent plasma physics benefits of the low aspect ratio tokamak, the FPP will demonstrate scalable net power in a fully-integrated system.

The low aspect ratio tokamak geometry enabled by High Temperature Superconducting (HTS) magnets offers some key inherent advantages over alternative approaches for fusion energy. In particular, the potential to access high confinement, high beta, and high bootstrap current fractions in a stable plasma configuration is an attractive proposition for economical power plants. Present parameters used in the modelling are: major radius 4.25 m, aspect ratio 2.0, toroidal field 4.5 T, and plasma current 16 MA.

However, because of the low aspect ratio, the flux swing capability of the central solenoid (CS) is not sufficient for plasma current (Ip) ramp-up to the full operating condition. Therefore, creative Ip ramp-up scenarios must be developed. In this talk, novel Ip ramp-up and sustainment scenarios utilising RF power (EC and IC), bootstrap current, and induction by the CS and other poloidal field coils will be presented.

METIS [1], an integrated tokamak modelling code combining 0D scaling-law normalised heat and particle transport with 1D current diffusion modelling and 2D equilibria, was used for scenario development. The plasma is initiated with EC power injected to the "trapped particle configuration" [2], as commonly used for ST plasma start-up. Following initial ECCD overdrive at low density, plasma is densified and ion heating by IC is applied to initiate fusion burn. The increased plasma stored energy drives the bootstrap current, and additional Ip ramp-up to full current is achieved with the inductive assist from the increasing vertical field. The transition from initial Ip ramp-up by LHCD overdrive at low density to NB heated bootstrap-dominant "advanced tokamak plasma" at high density (without the use of CS) has already been demonstrated on JT-60U [3]. Optimisation of the CS usage during Ip ramp-up is essential to achieve quick Ip ramp-up with minimal use of the CS flux.

Tokamak Energy and its FPP design efforts are supported by the U.S. Department of Energy's Milestone-Based Fusion Development Program.

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Radiofrequency Heating Systems, Experiments, and Modeling on WHAM and Implications for Next Generation Mirror Fusion Devices

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The Wisconsin High-Temperature-Superconducting (HTS) Axisymmetric Mirror (WHAM) is a magnetic mirror built at the University of Wisconsin-Madison utilizing 17 T magnets built by Commonwealth Fusion Systems with support from ARPA-E and Realta Fusion [1]. WHAM utilizes a 500 kW high-field launched 110 GHz fundamental X-mode ECH system for plasma breakdown and heating, like the system found in the Gas Dynamic Trap (GDT) [2], and a 1 MW variable frequency harmonic ($n = 2$) fast-wave (HFW) system for ion heating. The ECH system's design, construction, and operation in WHAM's recent experimental campaigns will be discussed as well as the design and simulation of HFW and ECH systems in WHAM and next generation mirror fusion devices.

The HFW system will be operational this year. Two experimental campaigns utilizing the WHAM ECH system have been completed. These campaigns included experimental studies of ECH breakdown as well as the characterization of a hot (~100 keV) electron tail formed by ECH. The role of hot electrons in mirror plasma breakdown [2] has been experimentally verified through X-ray measurements and fast-camera footage. Hot electron populations have also been observed to remain up to 1 s after breakdown, suggesting a high degree of stability.

These ECH experimental results allow the direct validation of integrated simulations against WHAM data. New magnetic equilibrium, raytracing, and Fokker-Planck (FP) codes are under development by Realta Fusion for the design of a mirror volumetric neutron source and a tandem mirror fusion pilot plant [3] [4] and simulation of WHAM. Recent simulations with these tools indicate Realta's raytracing-FP models for ECH can effectively predict fast-electron generation in WHAM plasmas. The impact of these fast electrons on WHAM's magnetic equilibrium and stability is now being evaluated. Modeling of the WHAM HFW RF system with these tools allows confinement time to be assessed and the effects of frequency tuning, varying central magnetic field, and plasma conditions on core RF coupling. Simulations have also been extended to next-generation magnetic mirrors that operate at higher beta [4]. These mirrors exhibit stronger wave damping, and it is shown that they can operate at the $n = 3$ harmonic.

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RF Heating Experiments and Plans for C-2W and Copernicus

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TAE Technologies is pursuing RF heating scenarios for the next-generation device, Copernicus. Two RF systems are being incorporated into the design of Copernicus: 6-9 MW of Ion Cyclotron Resonance Heating (ICRH) in the confinement vessel and 1.5 MW of Electron Cyclotron Resonance Heating (ECRH) in each of the two fueling cells. The design requirements for each of these systems will be discussed.

In preparation for RF heating on Copernicus, ICRH and ECRH systems have been developed for experiments on C-2W. For ICRH, simulations have been performed which show favorable results for High Harmonic Fast Wave (HHFW) heating as well as ion heating through magnetic beaching of a slow wave. Ray-tracing simulations have been performed to identify operational scenarios for ECRH. For HHFW, a 4-strap antenna and associated matching network designs are complete for launch and ion heating in the confinement vessel. For ECRH, the microwave beam path will intersect magnetic field lines corresponding with the electron cyclotron resonance near the mirror plug of the tandem mirror adjacent to the confinement vessel. The goals for ECRH are to increase confinement, increase fueling rate, and heat electrons in the scrape-off layer to increase fast ion accumulation.

For proof-of-concept testing of ECRH, a 70 kW, 28 GHz gyrotron is installed in the facility of the C-2W device. Commissioning of the gyrotron system is presently underway prior to integration with C-2W. Characterization of the gyrotron system in support of these experiments includes: optimization of gyrotron output power, transmission line optimization, and output mode purity assessment. Initial results of operation in C-2W will also be presented.

I-05

Conceptual design for the ICRF system of CFEDR

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The Chinese Fusion Engineering Demonstration Reactor (CFEDR) is a next-generation device for the Chinese magnetic confinement fusion (MCF) program to demonstrate the technological feasibility of deuterium-tritium fusion. CFEDR aims to achieve a sustainable fusion reaction by confining plasma using strong magnetic fields and applying auxiliary heating techniques [1-2].

ICRF is a key H&CD system and will play a very important role on the conventional H-mode operation scenarios for CFEDR. The ICRF system will provide 20MW RF power with frequency of 40-100MHz by an ITER-type antenna launched from an equatorial port. Conceptual design and analysis of ICRF system for CFEDR has been carried out, including RF source layout, transmission line RF components and antenna conceptual design, as well as the main heating scenarios. As the same time, a national big science facility called Comprehensive Research Facility for Fusion Technology (CRAFT) has been constructing to develop and master fusion DEMO level key technologies, establish the method and standard for manufacture the key components and 14 prototype system for CFETR including ICRF system [3-4]. Recent progresses on ICRF system which including design and R&D will be presented in this talk.

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3. C.M Qin, J.Li, X.J. Zhang et al., “ICRF system on CFETR”, *AIP Conference Proceedings AIP Conf. Proc.* 2984, 030001 (2023). https://doi.org/10.1063/5.0163675
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I-06

Resonant Interactions Between Fast Ions and ITG Turbulence: Implications for Confinement Improvements in Fusion Reactors

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In recent years, significant attention has been devoted to identifying mechanisms capable of reducing outward turbulence-driven fluxes through various physical mechanisms. One promising avenue is the influence of supra-thermal ions generated via external heating schemes. Their presence has been consistently linked to significant improvements in plasma confinement in tokamak experiments [1], with gyrokinetic simulations corroborating these findings by demonstrating a marked reduction in ion-temperature-gradient (ITG) induced anomalous transport [2-5]. However, despite the significance of supra-thermal particle effects in mitigating plasma turbulence, a comprehensive physical explanation has remained elusive for over a decade. In this contribution, we focus on a resonance interaction between fast-ion drift frequencies and linear ITG modes. This resonance mechanism proves particularly effective in reducing ITG turbulence under conditions where the normalized fast-ion temperature gradient surpasses the normalized density gradient, such as those achieved during minority ion-cyclotron-heating (ICRH). Under these conditions, a significant energy transfer occurs from the ITG-driven turbulence to the fast ions, thereby depleting the turbulence drive [6, 7]. The interplay between this wave-particle interaction is explored in detail, with GENE simulation results compared to experimental observations from various JET and ASDEX Upgrade discharges [7, 8] exhibiting reduced transport levels attributed to energetic ion effects. Furthermore, the potential implications of such resonant interactions for improving plasma performance in reactor-relevant conditions are investigated in the context of SPARC [9] and ITER [4] scenarios.

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I-07

40 Years of ICRF Physics on the JET Tokamak: Highlights and Lessons Learned for Future Facilities

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In December 2023, tokamak operations at the Joint European Torus (JET) came to an end. This exceptional fusion research device was designed and built more than 40 years ago to study plasmas in conditions and dimensions approaching those of a reactor. Its four main areas of research were plasma-wall interaction, plasma heating, plasma behaviour as parameters approach the reactor range and the study of fusion-born alpha particles.

The unique combination of JET's large size, ~ 80-100 m^3 volume, its low toroidal field ripple of less than 1% and high plasma current, up to 6MA in the X-point magnetic configuration, allowed effective confinement of high energy ions. This was essential not only for alpha particle studies, but also for radiofrequency heating using waves in the ion cyclotron range of frequencies (ICRF). In addition, its ability to handle tritium allowed the study of reactor relevant D-T heating schemes.

This paper reviews the contributions of JET to ICRF physics and their implications for the progress of fusion research. We will discuss how the JET experiments have provided many unique results in ICRF physics, bridging the gap to future burning plasma devices. First, we will look at how the JET results on ICRF experiments have led to major improvements in the understanding of heating scenarios, wave-particle interactions, ICRF-driven fast particles and their confinement, RF-induced plasma wall interactions, and the rich variety of ways in which ICRF waves can be used to heat plasma. We will then discuss how ICRF heating has become an integral and, in ITER-like wall conditions, an essential actuator to influence plasma behaviour in a controlled manner and to maximise fusion power in high fusion power plasma scenarios.

Progress in RF physics research has been enabled and strongly supported by the continuous innovation in RF technology applications at JET, as discussed in the complementary contribution to this conference by P. Dumortier et al.

I-08

Recent progress in ICRF experiments on EAST

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Ion cyclotron range of frequencies (ICRF) wave heating is not only widely used in current magnetic confinement fusion devices, but also been considered for future fusion reactors, such as SPARC, ITER and CFETR. To improve our understanding on efficient ICRF heating, ICRF-fast ion interaction, ICRF-turbulence interaction and so on, an abundant ICRF experiments were performed on EAST in the last few years. This study summarizes these results, aiming to shed light on better use of ICRF in future devices.

Increasing ICRF heating efficiency requires improving wave coupling at the plasma edge and RF power absorption in the plasma core. On EAST, the ICRF coupling was improved by decreasing the antenna parallel wave number, applying local midplane gas puffing, manipulating the SOL width and divertor strike point position and so on [1]. The core power absorption was increased by methods such as optimizing the cyclotron resonance position and minority ion concentration, applying ICRF-NBI synergetic heating and three-ion heating schemes.

Efficient ICRF heating generates large amounts of fast ions, which can lead to a large variety of Alfvén Eigenmodes (AE). In experiments with Hydrogen minority heating and ICRF power larger than 2 MW, TAEs with different frequencies were simultaneously driven by the spatial gradient of the fast ion distribution. These TAEs can only be excited when the H_{98} factor exceeds 1.15, suggesting that the confinement of fast ions plays an important role. A bunch of high frequency AEs (f=24.8-25.0MHz) were also excited by RF-accelerated minority hydrogen ions [2]. Moreover, by changing the Deuterium fast ion distribution in the plasma core by means of power modulation and phasing manipulation in ICRF-NBI heated plasma [3], the sawtooth period was decreased by a factor of two.

ICRF also interacts with the turbulence in the plasma edge and core. In the edge, ICRF attenuates the large-scale turbulence structures (usually in the form of blobs) by breaking them into smaller ones, leading to a suppression of turbulence in the whole SOL [4]. This turbulence reduction is accompanied by an increase in the high-frequency turbulence fluctuations and an increase in the poloidal velocity in the SOL. This poloidal velocity is generated by the radial electric field shear during ICRF. In the plasma core, it was observed that ICRF can also suppress turbulence, but the reason remains unclear. Further study will explore the mechanism.

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2 Liu L. et al Nuclear Fusion **64** 124002 (2024)
3 Zhang W. et al Nuclear Fusion **63** 056015 (2023)
4 Zhang W. et al Physics of Plasmas **31** 032502 (2024)

I-09

RF power experiments in WEST to prepare for next-step fusion device operation

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The WEST superconducting tokamak, featuring a full tungsten environment and equipped with an actively cooled ITER-grade divertor[1], provides valuable inputs for future ITER operation. One of its distinctive features is that auxiliary plasma heating and current drive is exclusively supplied by radiofrequency (RF) systems. In this contribution, we review recent progress and near-term plans related to plasma scenario performance improvement using RF waves in WEST, as well as other applications relevant to the operation of future fusion devices in full metal environments. As a tokamak focused on the achievement of long duration pulses, LHCD (Lower Hybrid Current Drive) is extensively employed in WEST. Scenario development supported by integrated modelling has allowed pulses exceeding 800s and 1.9GJ of energy to be performed, based on feedback-controlled plasma current sustained by LHCD power in order to achieve a 3mV target value of loop voltage. The ICRF (Ion Cyclotron Resonance Frequency) system has also been successfully employed in several instances. Argon pumpout by ICRF power at 48MHz has been observed in deuterium using a three-ion scheme when the hydrogen concentration was optimal, supported by full-wave simulations. In the context of an ongoing ITPA task devoted to the comparison of RF-sheath simulation tools, measurements performed using a new reciprocating emissive probe provide valuable information on ICRF-induced plasma potentials responsible for enhanced sputtering and heat fluxes to ICRF antenna limiters in metallic devices. Ion Cyclotron Wall Conditioning experiments have been conducted, yielding relevant information for the future application of this method in ITER. WEST has also demonstrated Ion Cyclotron-assisted breakdown in low electric field conditions (E_{loop} ~ 0.5V/m), approaching ITER values (E_{loop} ~ 0.33V/m). Progress in the design of a Travelling Wave Array (TWA) antenna operating in the 42-63MHz range of frequency will be also reported. Finally, starting in 2025, Electron Cyclotron (EC) power is available in WEST. The system will eventually supply up to 3MW of RF power at frequency 105GHz. Integrated modelling confirms that this additional power source will be instrumental in enlarging the operational space of WEST: central electron heating to balance the core power radiation, and current profile control to improve MHD stability. This additional power is also key for exploring improved confinement regimes.

1. J. Bucalossi et al., “WEST full tungsten operation with an ITER grade divertor”, Nucl. Fusion **64**, 112022 (2024).

I-10

ICRH simulations for the Wendelstein 7-X stellarator

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We will present the current status of ion-cyclotron-resonance heating (ICRH) simulations carried out for Wendelstein 7-X (W7-X), a large, superconducting, optimized stellarator being operated in Greifswald, Germany. First experimental experiences with the ICRH antenna at W7-X, in proper dipole configuration, are currently being gained in the experimental campaign conducted from autumn 2024 to spring 2025. Several tools are available for explaining the experimental findings or for the planning of new experiments. They range from simple cold-plasma models, targeting basic ICRH physics aspects, such as the locations of resonances and cutoffs in the plasma, to more advanced full-wave modelling. The reduced models are valuable for quickly comparing different magnetic configurations of W7-X and for checking the sensibleness of more advanced simulations, for which we use the SCENIC suite of codes [1]. SCENIC iteratively couples LEMan [2], an eigenvalue code solving Maxwell's equations, and VENUS-LEVIS [3], a Monte-Carlo code assessing the wave-particle energy transfer and modelling the build-up of a fast-ion distribution function. Especially, the capabilities of the ICRH system to generate fast ions are of interest, because a future stellarator fusion reactor needs to adequately confine such fast ions. We will give an overview of the tools that are currently available and highlight some of the results that have been obtained with SCENIC. We will focus on the minority heating of H ions in a ^{4}He background plasma, which is the primary ICRH scenario considered in W7-X. Different magnetic configurations and plasmas with varying minority concentration are compared regarding their potential to generate fast ions for subsequent confinement studies. We will also explore schemes with changed antenna frequencies, which shifts the resonance location toroidally in the 3D magnetic field of W7-X. Additionally, ICRH can also be operated together with neutral-beam injection (NBI), making use of synergetic effects when both heating systems are used simultaneously, and allowing higher fast-ion energies to be reached [4]. Finally, we will present some results related to the so-called 3-ion scheme [5].

1. M. Jucker *et al.*, "Integrated modeling for ion cyclotron resonant heating in toroidal systems," *Computer Physics Communications*, **182**, 912 (2011).
2. P. Popovich *et al.*, "A full-wave solver of the Maxwell's equations in 3D cold plasmas," *Computer Physics Communications*, **175**, 250 (2006)
3. D. Pfefferlé *et al.*, "VENUS-LEVIS and its spline-Fourier interpolation of 3D toroidal magnetic field representation for guiding-centre and full-orbit simulations of charged energetic particles," *Computer Physics Communications*, **185**, 3127 (2014)
4. M. Machielsen *et al.*, "Fast ion generation by combined RF-NBI heating in W7-X," *Journal of Plasma Physics*, **89**, 955890202 (2023)
5. Ye.O. Kazakov *et al.*, "On resonant ICRF absorption in three-ion component plasmas: a new promising tool for fast-ion generation," *Nuclear Fusion*, **55**, 032001 (2015)

I-11

Overview of ICRF plasma production and heating in gas mixtures in stellarators

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The study of plasma production and heating in the ion cyclotron range of frequencies (ICRF) has a long history in fusion research. The ICRF discharges have been studied in stellarators, tokamaks and mirror devices, mainly. The possibility of efficient additional plasma heating using ICRF is one of the main aims of the studies. Plasma production using ICRF discharges is also studied, but to a lesser extent. Because wall conditioning in both pure gases and their mixtures has been used with lower RF power level.

In support of the ICRF experiments for plasma production at Wendelstein 7-X, studies on the development of an ICRF start-up scenario were initiated on the Uragan-2M (U-2M) stellarator [1]. Experiments with a controlled minority of hydrogen in helium showed a significant increase in the resulting plasma density compared to pure helium and hydrogen [1]. Then, the ICRF plasma production was demonstrated on the LHD with the scenario based on U-2M experiment [2]. Successful experiments at U-2M and LHD showed that the ICRF start-up scenario can be scaled up to large stellarator devices, producing favorable plasma parameters as an initial plasma using ICRF only [3].

This paper summarizes both previous results and presents new studies on ICRF plasma production in gas mixtures. The main focus is on experimental results obtained on U-2M and LHD. In recent ICRF plasma production experiments at LHD, electron and ion temperatures up to 2.5 keV, and electron density of $\approx 10^{19}$ m^{-3} have been achieved. The radial distribution of the electron and ion temperature was measured. The mechanisms of plasma production and heating are discussed.

1. Yu. Kovtun et al., “ICRF production of plasma with hydrogen minority in Uragan-2M stellarator by two-strap antenna,” *Phys. Plasmas* , **31**, 042501 (2024).
2. Yu. Kovtun et al., “ICRF plasma production at hydrogen minority regime in LHD,” *Nucl. Fusion*, **63**, 106002 (2025).
3. V. E. Moiseenko et al., “ICRF plasma production and heating in LHD,” *29th IAEA Fusion Energy Conference (IAEA, 2023), IAEA-CN-316-1901*.

I-12

Exploring ICE Dynamics in Wendelstein 7-X: From Startup to Fast Ion-Driven Instabilities

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Wendelstein 7-X (W7-X) is equipped with an ion cyclotron emission (ICE) diagnostic, consisting of four inductors installed behind the heat shield in the triangular cross-section of the stellarator, and one of two straps of the ICRF antenna [1] in the bean-shaped cross-sections, the latter being available whenever the ICRF system is not used for heating [2]. The ICRF strap is approximately 30 dB more sensitive than the inductors.

It has been predicted that ICE, originating from the magneto-acoustic ion cyclotron instability (MCI) and driven by neutral beam injected (NBI) fast ions, would be very strong in W7-X at high ion cyclotron harmonic numbers (of the order of 10) and at lower harmonic numbers (1-3) [3]. These calculations were based on a local model and the assumption of a ring distribution function of unconfined NBI fast ions. Experimentally, we have been able to partly confirm the low harmonic number predictions through B-dot measurements obtained using the inductors. However, we discovered a very strong dependence of the ICE spectrum on global plasma parameters, such as the magnetic field configuration. Observations using the ICRF antenna revealed a broader range of activity, ranging from no visible ICE activity in the presence of fast ions to a clear manifestation of every single ICE harmonic within the diagnostic frequency range of 0-500 MHz. This type of emission is broadband and corresponds to fast ion cyclotron harmonic locations throughout the plasma. Additionally, in certain settings, we found narrowband activity with frequencies that do not correspond to any specific ion cyclotron harmonic inside the plasma, with frequencies ranging from 20 MHz to 150 MHz.

Thermal ICE was routinely observed at low harmonic numbers, and its appearance depended on the magnetic configuration, electron density, and temperature. This emission is also broadband. We present the results of a systematic scan of plasma parameters and their influence on thermal ICE.

Plasma startup and decay are associated with both narrowband and broadband activity linked to electron kinetics.

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40 Years of ICRF Operation on JET: Achievements and Challenges

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After 40 years of successful operation and multiple Deuterium-Tritium (D-T) campaigns, the Joint European Torus (JET) concluded its operations in December 2023. From the outset, Ion Cyclotron Range of Frequency (ICRF) heating was identified as a key auxiliary heating system, with the first ICRF system becoming operational in 1985. This system underwent significant development, culminating in a high-power (32MW installed), wideband (23-57 MHz range), and highly versatile system. The A2 antennas, installed during the 1993 shutdown, operated successfully until the end of JET's operation, demonstrating an impressive 30-year service without major issues.

This contribution provides an overview of the JET RF system's evolution over its lifetime, highlighting selected technological achievements and operational challenges. It addresses the rationale behind the development and operation of the ICRF system, including the implementation and operation of load-resilient systems (3dB hybrid coupler, External Conjugate-T, ILA), as well as the operational limits and system protection measures, such as the specific mode of operation for Ion Cyclotron Wall Conditioning (ICWC). The impact of transitioning from a carbon wall to the beryllium and tungsten ITER-Like Wall on the system's operation and the challenges posed by D-T operations are also emphasized.

Finally, key lessons learned from the technical challenges, constraints, and achievements of the JET experience are highlighted, with an eye towards the design and operation of ICRF systems in future machines.

Contributions to the progress in RF physics understanding and achievements at JET are further detailed in a complementary paper by M. Mantsinen et al. at this conference.

Radar arc and impairment detection and localization for the ITER ICRF antenna

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A **RA**dar **A**rc and impairments **D**etection and localization (RAAD) system for the protection of high-power RF transmission lines components is presented. It offers several benefits over existing techniques. It can perform fast detection, localization, and classification of arcs and impairments/degradations along the full path of the RF power (< 5 μs have been specified for the protection of ITER ICRF antenna). It also features completely independent operations from the RF power. The detection and monitoring of impairments/degradations enables the prevention of faults while the localization and classification of faults permits pinpoint maintenance operations.

The RAAD system has been simulated, in time domain adopting the Simulink tool, integrated into the complete ITER ICRF system (from the array antenna front-face up to the RF sources outputs, including the full transmission line and matching network). The radar signal is coupled into the transmission lines through a specifically designed coupling unit.

Full wave simulations of the ITER ICRF system components have been performed in the radar bandwidth of operation, with and without arcs/impairments, to obtain S-matrices used in the time domain radar system simulations. All the arcs and impairments inserted in the components have been detected; the location obtained by the radar for each event has been compared to the one estimated realistically with excellent agreement. The radar simulations have demonstrated the ability to detect and localize arcs which cannot be detected by other systems such as low voltage arcs and series arcs.

A solid solution, based on signal synchronization, for coping with high power interferences from ion cyclotron frequency harmonics in the radar bandwidth of operations has been implemented. Other events such as antenna load variations and matching elements modifications can be easily discriminated against arc events, being much slower than radar pulse repetition period.

Next steps will be the implementation, testing and validation of a prototype of the complete RAAD system. This prototype will be tested on the ITER ICRF prototype antenna module and potentially on other operating ICRF installations.

A structure-preserving spline Finite Element solver for the cold-plasma model

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We present a finite element solver relying on B-splines, which preserves the Hamiltonian structure of the cold-plasma model [1] as well as several physical invariants in Cartesian or curvilinear geometries. In particular, it preserves the energy, the total charge or the divergence of the magnetic field. A key feature of the scheme is that in the presence of a time-harmonic source, it is consistent with a high order approximation of the associated time-harmonic solution: this makes the solver intrinsically stable, which guarantees that long simulation runs do not develop unphysical effects. Besides, we will discuss the accuracy, stability properties of the schemes and compare the performance of different geometric time splittings [2].

Our implementation relies on Psydac [3], an isogeometric B-splines Finite Elements library developed at the Numerical Methods for Plasma Physics division of the Max-Planck Institute for Plasma Physics. Psydac can be used to build efficient solvers based on modern numerical methods. In this talk we will present how to set up a wave propagation problem.

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3. https://github.com/pyccel/psydac

Integral dielectric kernel approach to modelling RF heating in toroidal plasmas

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Recent theoretical and numerical treatments such as [1, 2] have sought to express the plasma radiofrequency (RF) response as a nonlocal integral operator formulated in configuration space. This approach promises major advantages with respect to the traditional spectral methods, by enabling *(i)* the use of finite element methods (FEM) - in two or three space dimensions - to model wave propagation and absorption in hot inhomogeneous fusion plasmas; *(ii)* local mesh refinements which were ruled out with spectral methods; *(iii)* better suited RF field representations to address finite Larmor radius effects in toroidal geometry; *(iv)* straightforward connection of the plasma model with RF antenna models, themselves based on the FEM.

The first part of the talk reviews the linearized Maxwell-Vlasov theory of wave-particle interactions in toroidal geometry, emphasizing the simplifying assumptions made across the literature to derive tractable numerical models from the most general expressions such as obtained in [3], and elaborated in e.g. [4]. Our integral approach and the associated kernels are then presented for Maxwellian particle species. We highlight specific features of the dielectric response in toroidal geometry, such as its singular behaviour at rational-*q* surfaces - i.e. on closed magnetic field lines -, as well as desirable future theoretical extensions.

The second part reports on the application of the integral approach to RF wave propagation and absorption in toroidal plasmas. As stressed above, its novelty resides in the treatment of nonlocality in physical space, which is not typical of the FEM. Taking a gradual approach to validate the numerical methods, we are initially implementing the theory to lowest order in Larmor radius, focussing on minority ICRH scenarios. We report on '2.5D slab' investigations of thermal dispersion effects along the equilibrium magnetic field and of the associated collisionless power absorption, using finite elements in both parallel and radial directions. We finally discuss the development path toward implementing the full geometrical complexity of tokamaks and stellarators, as well as additional wave physics, in hot plasma codes based on the FEM.

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I-17

Kinetic Full Wave Analyses in Inhomogeneous Plasmas Using Integral Form of Dielectric Tensor

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Kinetic description of wave propagation and absorption in inhomogeneous plasmas is an important issues in quantitative analyses of wave heating and current drive in high-temperature plasmas. Ray and beam tracing methods based on the geometrical optics can describe the wave behavior in hot plasmas using local kinetic dielectric tensors, but wave optical phenomena, such as tunneling of evanescent layers, formation of standing waves and coupling with finite size antenna cannot be described. The full wave analyses in which Maxwell's equation for given wave frequency (complex in general) is solved as a boundary-value problem have been widely used to describe the wave structure in confined spaces. In order to include kinetic effects in plasma response, various schemes have been proposed and employed, but most of them have limitations due to the use of local kinetic dielectric tensor for fixed wave number. For systematic description of kinetic effects in inhomogeneous plasmas, integral form of dielectric tensor is introduced. The integral form can be derived by transforming the velocity integral in the kinetic dielectric tensor $\overleftrightarrow{\epsilon}$ into the positional integral of the wave electric field $\boldsymbol{E}(\boldsymbol{r}')$ along the particle orbit. The induced current $\boldsymbol{j}(\boldsymbol{r})$ is expressed as $\int \mathrm{d}\boldsymbol{r}' \overleftrightarrow{\epsilon}(\boldsymbol{r}, \boldsymbol{r}') \cdot \boldsymbol{E}(\boldsymbol{r}')$ without wave number.

First this scheme is applied to unmagnetized plasmas. The laser-plasma interaction in a plasma with density gradient is considered. Plasma waves are excited near the plasma resonance and absorbed by the Landau damping. It was found that stochastic heating occurs in a plasma with steep density gradient even in the case of normal injection where no plasma wave is excited. Next magnetic beach heating in magnetized plasmas with linearly increasing magnetic field strength along the field line is considered. Electron cyclotron waves with right-hand-side circular polarization is absorbed in approaching the electron cyclotron resonance. In the unmagnetized case and the parallel motion along the magnetic field can be formulated with plasma dispersion kernel functions (PDKF). The analyses was extended to the case of parabolic dependence in order to describe the cyclotron absorption near the maximum and the minimum of field strength using an extended kernel function (PDKF2). The cyclotron motion perpendicular to the magnetic field causes the finite Larmor radius effects and the Bernstein waves. The perpendicular motion is formulated with plasma gyro kernel functions (PGKF). The O-X-B mode conversion of the electron cyclotron waves excited from the low-field-side of tokamak plasma was well described [1]. Finally two-dimensional analyses of cyclotron waves on poloidal cross section of tokamak plasma is carried out. Since the magnetic field strength has minimum and maximum on the magnetic surface, PDKF2 as well as PGKF is employed. Mode conversion to the Bernstein waves and the cyclotron harmonic absorption are simultaneously described.

[1] Khan SA, Fukuyama A, Plasma Fus. Res. 11, 2403070 (2016)

I-18

Beam Tracing in Phase Space:
Paraxial Description of High-Frequency Wave Beams in Turbulent Plasmas

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Asymptotic methods based on the short wavelength limit constitute the main computational tools for the description of high-frequency wave beams in fusion plasmas. One such method, namely the paraxial Wentzel-Kramers-Brillouin (pWKB) or "beam tracing" method [1], has been proven to be computationally efficient and physically insightful for both electron cyclotron [2] and lower hybrid waves [3]. The pWKB method, however, cannot be directly applied to plasmas with strong density fluctuations, and these can be a concern for the beam quality in large tokamaks such as ITER. Understanding the effect of fluctuations has generated a considerable volume of work in the last years, starting with the paper by Tsironis and coworkers [4]. In this context, we have proposed an approach based on the direct numerical solution [5] of the wave kinetic equation derived by McDonald [6], which describes the wave energy transport averaged over a statistical ensemble of density fluctuations. This numerical approach has been successful, but it is computationally much more expensive than the standard pWKB method.

In this talk, we show that some of the ideas at the basis of the pWKB method can be transferred to the wave kinetic equation in order to obtain a simpler and physically more transparent description of the wave beam. The simplification stems from the fact that the solution of the wave kinetic equation in the geometric-optics phase space is concentrated around a reference trajectory [7,8], opening the possibility to develop a phase-space beam-tracing method. Promising numerical tests of this approach in simplified geometries are presented.

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Experimental Evidence of Helicon Wave Heating and Current Drive in DIII-D

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Helicon waves – fast waves in the lower hybrid range of frequencies – are an attractive candidate for driving off-axis current to sustain advanced scenarios in reactor conditions. Dedicated DIII-D experiments have been conducted with a MW-level system to demonstrate core heating and current drive with helicon waves launched via a traveling wave antenna. Unambiguous electron heating is observed in both L modes with partial (~10 – 50%) predicted first pass absorption and H modes with nearly complete (> 90%) predicted first pass absorption. When modulating the helicon power, the core electron temperature T_e rises during each helicon pulse and falls between them, indicating a coherent response. The measured heating profile is in good agreement with the predicted profile from first pass absorption, once thermal transport effects are accounted for with time-dependent modeling. A scan of β_e confirmed stronger absorption at higher β_e, consistent with Landau damping and in quantitative agreement with ray tracing calculations.

In shots with continuous injected helicon power, a higher core T_e is observed in comparison to otherwise matched shots without the helicon. In the best L mode cases, the core T_e was ~1 keV higher due to 350 – 400 kW of helicon power coupled to the plasma. Moreover, measurements of the time evolution of the local magnetic field pitch angle with motional stark effect (MSE) polarimetry show reproducible differences in a series of L mode plasmas with helicon compared to those without. MSE-constrained equilibrium reconstructions indicate that the on-axis value of the safety factor drops faster over time in shots with helicon waves injected to drive current in the co-I_p direction, which also exhibit a corresponding modification of the sawtooth behavior. Comparison to a third set of shots that replaces the helicon with a comparable amount of electron cyclotron heating aimed near the same radial location as the helicon absorption (with radial launch, to minimize current drive) confirms that the changes in MSE and q evolution in shots with helicon cannot be explained solely by the effect of additional heating on the resistive current diffusion. Furthermore, the plasma density in these shots would have prevented any incidentally excited slow waves from propagating to the core, ruling out another possible explanation.

Two complementary approaches were taken to quantify the helicon current drive profile. The first determines the change in the total current density directly from the MSE array in shots with versus without helicon, while the second isolates the change in the non-inductive contribution to the current by calculating the loop voltage profile from a time series of reconstructed equilibria. Both methods yield a peaked helicon current profile concentrated within $\rho < 0.2$, similar to the measured helicon heating profile, distinct from those found for the equivalent ECH shots, and consistent with ray tracing predictions. Taken together, these experimental results represent strong evidence for the first observation of auxiliary current drive due to helicon waves in any device.

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Design of the Actively Cooled Ion Cyclotron Traveling Wave Array System for WEST

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Ion Cyclotron Resonance Heating (ICRH) is commonly used as a plasma heating technique in magnetic fusion experiments. However, the main drawback of the current ICRH systems is the challenge of coupling a large amount of power without producing high RF voltages, that could enhance the production of metallic impurities or exceed the voltage stand-off of the antenna(s). Future experiments will require higher RF power but the extrapolation of these classic antennas to higher power densities seems to be at the limit of the current expertise. In addition, future system will have to cope with more stringent constraints such as neutron fluxes or not occupying the volume necessary for tritium breeding blankets.

The Traveling Wave Array (TWA) antenna concept provides increased RF coupling, enhanced directivity and intrinsic load tolerance over a broad frequency band, opening up new operational areas. TWA launchers are compatible with a future fusion power plant (materials, reliability and tritium breeding ratio impact).

Unlike other machines, WEST offers sufficient space in the vacuum chamber to integrate a TWA launcher, in a relevant tungsten (W) environment. Following the successful validation of a high-power mock-up of a TWA launcher up to 2 MW under vacuum conditions in 2021, the design of two TWA launchers is on-going for WEST. These two independent TWA launchers, stacked on top of each other, are designed to be load-tolerant in the 57+/- 5 MHz band, to sustain H minority heating at full WEST field or second harmonic hydrogen scenarios at half field. The strap spacing and width were tuned to reach a main parallel wavenumber $k_{//}$ between 8 to 10 rad/m, as a trade-off between easy power coupling and reasonable single pass absorption and ion heating. The dimensions allow for 8 straps in the toroidal space available for each TWA launcher. This yields a rather narrow $k_{//}$ spectrum, hence peaked power deposition profile. The two launchers can be fed from either toroidal extremities, through a single WEST port. In a second phase, an external resonant ring circuit could re-circulate the unabsorbed power to increase the overall efficiency of the system. RF modeling do not show a crucial effect of a Faraday Screen on the parasitic excitation of parallel electric fields on the limiters. Static and electromagnetic loads due to VDEs have been preliminary evaluated and do not present any showstopper for integrating such antennas in WEST, allowing to proceed to a more detailed design.

Exploration in LH coupling and current drive towards long-pulse operation on EAST

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Lower hybrid current drive (LHCD), with 2.45 GHz /4 MW and 4.6 GHz / 6 MW, is one of important tools to sustain long pulse plasma with high performance in EAST. Challenges of LH (lower hybrid) coupling, including hot spots, and anomalous loss of LHCD efficiency at high density in long-pulse and high β_p operation with high LH power have been explored recently.

Aiming at good LHW-plasma coupling required for long pulse plasma, it is the first time that the coupling feedback control is designed and realized in EAST through Proportion Integration Differentiation method by choosing RC (reflection coefficient) of LH power as the reference for the gas puffing feedback, demonstrating the validation of feedback control on LH coupling in long pulse plasma. In order to decrease the interaction of LH antenna with the scrape off layer (SOL) plasmas, a new passive-active-multijunction (PAM) launcher with 2.45GHz has been developed, and good coupling with RC ~ 3% has been demonstrated with the plasma-antenna distance up to 11 cm. By upgrading the guard limiters, the power handling capability of LHCD systems is improved and the 4.6 GHz LHCD system can operate with power > 2.5 MW routinely.

Effect of parametric decay instability (PDI) on current drive efficiency of 4.6 GHz lower hybrid wave in EAST is investigated experimentally, showing the PDI channel bifurcation of LH wave for the first time and demonstrating the role of plasma density in edge region. With lithium wall conditioning and favourable magnetic field, the interactions of LH wave with SOL plasmas are mitigated, and good LHCD efficiency ~ 0.9×10^{19} A/W/m^2 is obtained even when density increased to 5.4×10^{19} /m^3 in H-mode plasmas. High electron temperature T_e >12 keV plasmas measured by Thomson scattering is produced by injecting 1.4 MW electron cyclotron (EC) waves into the center of plasmas sustained by 2.3 MW LH waves.

Finally, significant advance has been achieved with RF waves, including long-pulse H-mode operation up to 709 s with 1.4 MW 4.6 GHz LH power and 1.9 MW EC power, long-pulse I-mode operation up to 1056 s sustained by moderate 4.6 GHz power ~ 1.05MW and EC power ~ 0.55 MW (see Fig.1). In order to enhance the H&CD capability, the 2.45 GHz system will be replaced by a new 4.6GHz/4MW system with a PAM launcher, which is under construction. Studies of coupling and current drive offer a new possible strategy of combining the nonlinear effect in edge region and the wave-plasma coulpling by feedback control in long pulse palsma with high performance.

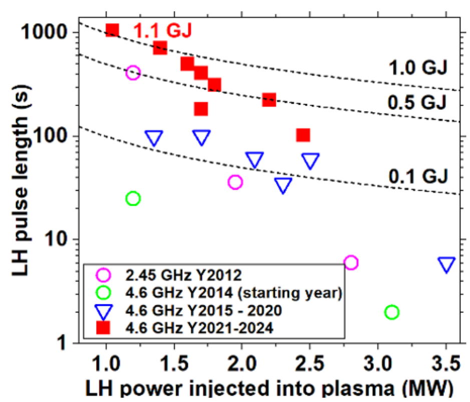

Fig. 1 Advances in the LH power injected into plasma and in their pulse duration on EAST.

First Results from the High Field Side Lower Hybrid Current Drive Experiment in DIII-D

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High field side lower hybrid current drive (HFS LHCD) is a potential tool for efficient off-axis current drive in tokamaks. From the HFS, LH waves are expected to have improved accessibility and penetration [1] and single-pass absorption [2]. The first HFS LHCD system has been installed on the DIII-D tokamak, and first experiments are planned for 2025. From simulations of high q_{min} DIII-D discharges, the system is expected to provide efficient off-axis current drive with peak current density up to 0.4 MA/m^2 and 0.14 MA/MW coupled using $n_{||}$ = 2.7 at 4.6 GHz [3]. Once fully operational, the system is expected to couple up to 1.6 MW with a power density of ~32 MW/m^2. The LHCD coupler utilizes a traveling wave 4-way poloidal splitter and a toroidal multijunction. Internal impedance matching structures within each aperture minimize reflected power, allowing operation in a wider range of edge plasma conditions. Manufacturing of the complex internal coupler structure is enabled by additive manufacturing using a high temperature copper alloy, GRCop-84, originally developed for aerospace applications. Klystron commissioning is ongoing with power levels and pulse lengths relevant to plasma experiment having been achieved. In preparation for experiments, the HFS scrape-off layer density profile has been characterized using profile reflectometry. Using machine learning techniques, the HFS SOL density profile can be accurately predicted using global plasma parameters. This allows for rapid, accurate prediction of local HFS SOL conditions, optimization of LHCD coupling, and more detailed study of RF-SOL interactions. The latest simulations, system status, and experimental results will be presented.

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Spectrum Evolution and Density Limit in Self-consistent Modeling of Lower Hybrid Wave Propagation with Parametric Instabilities

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This work provides self-consistent modeling and simulation of LHWs in the SOL plasma by coupling the propagation of waves to the power transfer among waves by parametric instabilities (PIs) in a 3D model based on the ray-tracing method. The spectral broadening is induced by PIs and, meanwhile, the pump power flux was converted to sideband waves and deposited. When the density increases, stronger PIs transfer most of pump power flux, which results in the density limit of LHCD. For the first time, the density limit of LHCD observed at JET [1], EAST [2] and C-Mod [3] can be successfully reproduced through theory and simulation.

To couple the PI process with the linear propagation of LHWs in the regime of WKB analysis of waves, the model of PI saturation in an inhomogeneous plasma derived by Rosenbluth [4] should be extended to adapt to the scenarios where non-resonant quasi-modes are involved.[5] Subsequent analysis shows that both the radial and toroidal direction needs to be considered to derive the appropriate convective amplification of PI, for the daughter LHWs might propagate across the resonant region through both directions.

A code involving PIs is developed [6] to simulate the self-consistent evolution of the spectrum and power flux of the LH wave in SOL plasma. The power loss due to PIs is found to be closely related to the density and temperature profile of SOL plasma when the magnetic field and the pump are fixed. A cool and dense SOL plasma (namely, the SOL plasma with gas puffing near the LH antenna to improve the coupling of LHW) leads to considerable PI growth rate and convective loss, which further results in the density limit of LHCD caused by PI. A significant broadening of the LHW spectrum is observed when the SOL profile approaches the density limit, which attributes to the amplification of the channels related to discrete harmonics of the ion-cyclotron quasi-modes (ICQM) with large refraction indices ($n_z > 10$). These LHWs induced by PI deposit their energy instantly in the SOL region, thus results in the anomalous efficiency loss of LHCD.

Both the theoretical model and simulation results corroborate the previous experimental findings that PIs can be quenched by improving the magnetic field and LHW frequency as well as other methods to optimize the SOL profile. It is shown that in the parameter regime of ITER, the high LHW frequency, strong magnetic field and higher periphery plasma temperature results in weak PI, indicating that LHCD remains a promising method of driving plasma current for ITER and future fusion reactors.

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Tungsten Erosion and Transport Induced by RF Sheaths at Antenna Structures in the WEST Tokamak

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This study presents STRIPE (Simulated Transport of RF Impurity Production and Emission) [1], an integrated modeling framework designed to analyze material erosion and transport at RF antenna structures in fusion devices. STRIPE combines SolEdge3x for plasma profiles, COMSOL [2] for 3D RF rectified voltage fields, RustBCA for sputtering and reflection yields and GITR for ion energy-angle distributions and global impurity particle tracking. Applied to ICRH discharge #57877 in the WEST tokamak, STRIPE predicts a tenfold increase in W erosion at RF antenna limiters under RF-sheath rectification compared to thermal sheath-only conditions. The analysis strongly suggests the dominant role of higher charged states of oxygen ions (e.g., O^{6+} and higher) in tungsten sputtering under WEST conditions, with modeling results showing reasonable agreement with experimental W-I spectroscopic data, thereby validating the framework. In addition to gross erosion, this study examines net erosion and redeposition patterns, accounting for tungsten self-sputtering. The global migration of sputtered tungsten originating from ICRH antenna structures is also investigated using STRIPE, providing insights into impurity transport dynamics. The framework is currently being expanded to explore plasma-material interactions in other RF-heated devices, including DIII-D, ITER, PISCES-RF, and MPEX.

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Mitigation of ICRF - Edge Plasma Interaction in Alcator C-Mod

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The parasitic interaction between waves in the ion cyclotron range of frequencies (ICRF) and the boundary plasma has long been a challenge to high-power ICRF utilization in magnetically confined fusion plasmas. In this presentation, we report on the successful mitigation of this interaction by power tapering a four-strap field-aligned antenna in the Alcator C-Mod tokamak. With the antenna operated in dipole phasing (0π0π), the ratio of the power coupled by the central two straps (P_{cent}) to the power coupled by the outer two straps was varied at fixed total coupled ICRF power (P_{tot}). With an appropriate P_{cent}/P_{tot}, no enhancement of the plasma potential was measured despite 1 MW of coupled ICRF power. When the antenna is powered, the energy deposited on its four corners increases and a hot spot appears on a neighboring limiter, at the location that magnetically maps to one of the antenna corners. The energy deposited on the antenna and limiter are minimized together for an appropriate P_{cent}/P_{tot}. Additionally, deviation from an optimal P_{cent}/P_{tot} value can cause far scrape-off-layer plasma heating in places not magnetically connected to the antenna. The amount of far-field plasma heating correlates with the amount of power in the low ($< 5\ m^{-1}$) and high ($> 20\ m^{-1}$) $k_{||}$ components of the antenna spectrum. The optimal P_{cent}/P_{tot} is also found to improve ICRF heating efficiency in L-mode and facilitate H-mode access.

The interaction between ICRF-induced **E**×**B** flows and turbulent filaments in the scrape-off layer (SOL) is investigated, and we show that the sheared flows can slow down and destroy filaments, eventually reducing convective transport in the far-SOL. A novel machine learning-based blob tracking algorithm [1] was applied to high-resolution Gass Puff Imaging videos, providing a direct observation of this interaction. Power tapering is shown to eliminate this interaction by reducing the RF-induced **E**×**B** velocity.

Finally, we show that the heat flux pattern on the toroidally aligned (TA) antennas reverses with the direction of the RF-induced **E**×**B** flow, a correlation that is not observed for the field-aligned (FA) antenna. We argue that the RF-induced **E**×**B** flow, which carries high-density plasma from the near-SOL, is aligned with the FA antenna and does not intercept its corners. This could explain the previously reported lower impurity generation by the FA antenna [2].

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Wave-filament Interaction Experiments on the LArge Plasma Device (LAPD)

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It is well understood that filamentary density structures may cause scattering and mode-conversion of waves in the lower hybrid range of frequencies [1]. Both like and unlike mode scattering can occur [2], however additional phenomenon can lead to wave power in the unintended branch. In the regime in which the slow wave is evanescent, the steep density gradient at the filament interface can act to excite a surface mode that would otherwise not be supported by the plasma [3]. In fusion RF schemes this can act as a significant power loss channel if the wave regime, typical filament size, and intensity match the surface mode excitation condition [4].

Experiments were conducted on the LArge Plasma Device (LAPD) at UCLA to explore the excitation of this slow wave surface mode in an easily diagnosable plasma with a simplified geometry, in which RF waves are launched from the edge into a uniform plasma with a field-aligned density enhancement in the core. Two plasma sources create a quiescent background plasma and a Gaussian-like density enhancement in the core representing the filament [5]. Probes are used to measure the wave fields in the vicinity of the filament in 2D planes across the background field. The difference in polarization between the fast and slow wave means that concentrations of the wave electric field parallel to the static background magnetic field ($E_{||}$) is a clear indicator of slow wave-like fields.

The results shown here demonstrate the observation of wave fields indicative of the slow wave surface mode. Concentration of $E_{||}$ at the filament interface is clearly observed. Progression in the phase of the measured fields indicate the presence of a mode structure along the filament density contours. Comparison to a 2D full-wave cold plasma model demonstrates qualitative agreement with the measured mode structure. The 2D model was further used to study the sensitivity of the interaction on filament parameters, and explains some of the variability observed in the experiment. The implication of the surface modes to other experiments, such as heating and current drive efforts on tokamaks, will be discussed.

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Modeling of Turbulence, Transport, and RF-induced Convective Cells in Tokamak Boundary Plasma

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RF antenna operation in tokamaks has experimentally been shown to affect edge density profiles, both of bulk and impurity species, and these effects are generally attributed to the presence of RF-rectified sheaths and RF-induced E x B convective behavior [1]. Impurity sputtering events are triggered as ions accelerate through the RF-rectified sheaths to strike the wall; the transport of impurity species is also affected by the RF sheath fields and the associated RF-induced convection. Likewise, for bulk species, electric fields arising from the RF-induced sheath potentials give rise to convective E x B drifts, which locally alter density.

In this work, we exercise a fluid plasma turbulence model (Hermes, [2]) in a magnetic flux-tube domain to explore the influence of imposed RF-induced sheath potentials and RF-induced convective behavior on the edge plasma turbulence and transport. Sheath potentials are shown to quantitatively affect both the mean species density and the RMS density fluctuation spectra. Through particle tracking within the turbulent fields, we also demonstrate and begin to quantify the effects imparted to the impurity transport by RF-induced convective cells.

In this same geometry, we exercise a nonlinear plasma/sheath model (VSim, [3]) to develop more quantitative predictions for these RF potentials, exploring the role of density fluctuations both in RF sheath formation and field propagation. We also examine how changes to the antenna configuration (e.g. number and relative phasing of antenna straps) may influence the results outlined previously.

Lastly, we discuss ongoing and projected development of a more physically self-consistent coupling between the Hermes and VSim models, based on physical insights we have gained as these models are run as standalone codes and/or loosely coupled.

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Automated ICRF heating surrogate models via machine learning*

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Robust, real-time-capable Ion Cyclotron Range of Frequencies (ICRF) heating surrogate models have been implemented via automated machine learning (ML) workflows. Previous research demonstrated fast and accurate predictions for Lower Hybrid Current Drive (LHCD) [1] and, more recently, for High Harmonic Fast Wave (HHFW) heating at NSTX and Ion Cyclotron (IC) minority heating at WEST [2]. Here, we present a novel Automated Surrogate Generator Suite, providing capabilities to train, test, optimize, save, and deploy ML-based surrogate models. The workflow, based on open-source software, automatically optimizes surrogate model designs for Random Forest Regressors (RFR), Multilayer Perceptron Regressors (MLP), and Gaussian Process Regressors (GPR) using advanced optimization frameworks for hyperparameter tuning and training, such as Bayesian optimization. The development of high-fidelity ML-surrogates allows extending the application physics-models based on traditional computational methods to applications where cost-efficiency requirements often cannot be met, such as scenario optimization, inter-shot predictive modeling, and real-time control. Additionally, some of these ML models can accurately extrapolate beyond the training parametric range, resulting in enhanced robustness in the predictions [3].

The novel workflow is first tested on HHFW and IC minority heating problems using databases based on TORIC simulations [4]. Here, optimized models improve regression accuracies (R^2) and successfully find optimal architectures for each electron and ion heating surrogate models. The suite also achieves GPR models which feature the best regression accuracy and provide uncertainty quantification in the predictions. Additionally, a novel formulation of the radiofrequency waves in hot Maxwellian plasmas via finite elements and a rational approximation [5] is adapted to ML frameworks, enabling the development of ML-based surrogates for ICRF heating in ITER in 1D. The model includes slow wave propagation and the move conversion to the ion Bernstein wave in the vicinity of the ion-ion hybrid resonance. The Automated Surrogate Generator is used to train an ICRF surrogate heating model for ITER, with the aim to be integrated with TOMATOR-1D to investigate ion cyclotron wall conditioning [6].

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Poster presentations

Monday-01

ICRF Antenna Modeling and Coupling Analysis for CFEDR

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The Ion Cyclotron Range of Frequencies (ICRF) could be an important auxiliary plasma heating method in the Chinese Fusion Engineering Demonstration Reactor (CFEDR). Currently, one ICRF antenna located from an equatorial port is beening considered. A coupled ICRF power (P_c) of 10 MW during the ramp-up phase or 20 MW during the flattop phase is required from this antenna to obtain or sustain the conventional H-mode on CFEDR, respecitively. Previously, a conceptual ITER-type ICRF antenna has been designed by Zhang et al for CFETR [1]. Based on this concept, simulations have been conducted using the RAPLICASOL antenna code [2,3] to evaluate ICRF wave coupling for various edge plasmas scenarios. The analysis of the antenna coupling resistance (R_c), fast wave electric field (E_f), and current distribution on the straps (J_s) with varying edge electron density (ne) profiles for frequencies between 50 and 90 MHz is presented. The parallel electric field ($E_{//}$) excited by the antenna is used to evaluate the sheath potential (V_{sh}). Finally, optimized toroidal phasing of strap currents suitable for efficient ICRF heating has been achieved from a phasing scan.

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3. J. Jacquot et al AIP Conf. Proc. 10 December 2015; 1689 (1): 050008.

Monday-02

Suprathermal electrons transport studies in radio-frequency-heated tokamak plasmas

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Radio-frequency waves are routinely launched in tokamak plasmas to drive current by accelerating electrons to suprathermal energies. Different ranges of frequencies are used to achieve different objectives. High frequency Electron-Cyclotron (EC) waves benefit from their ability to drive a highly localized current density to tailor the plasma current profile, thus enabling the control and the mitigation of magneto-hydro-dynamic instabilities to improve the plasma stability. On the other hand, the lower frequency Lower-Hybrid (LH) waves are one of the main choices for efficiently driving and sustaining non-inductive current, enabling the exploration of long-pulse operation in today’s tokamaks, such as in EAST or WEST [1].

However, it is far from straightforward to experimentally characterize the power deposition profile and the associated current density. This heavily relies on simulations, from integrated modelling using simplified models for the wave-plasma interaction to more sophisticated quasilinear drift-kinetic Fokker-Planck models. Even these Fokker-Planck models suffer from the lack of first principles turbulent transport calculation, and the matching of the simulated data with the experimental ones requires the addition of ad-hoc radial transport of suprathermal electrons, ranging between 1 and 4 m^2/s for EC waves [2,3] and 0.1 and 0.3 m^2/s for the LH waves [4].

This contribution reports on power-modulation experiments led to investigate the radial transport of suprathermal electrons, using the hard X-rays as a signature for their spatial distribution, energy and density. From these perturbative studies, it is possible to estimate a diffusion of these electrons, and to compare the results with forward drift-kinetic Fokker-Planck simulations performed with LUKE. This method has been applied in the TCV tokamak for the EC waves, for which experimental and numerical results are consistent, and point in the direction of an increased turbulent transport with EC power, which may explain the increased transport of suprathermal electrons with EC power [3]. For the LH waves, power-modulations are performed in the WEST tokamak, and preliminary results are reported in this contribution.

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Monday-03

ICRF near-field effects sensitivity to magnetic geometry

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Near-field effects are induced at ICRF antenna limiters and locally enhance interactions. We study their dependence with (i) the magnetic field tilt-angle and (ii) the field line incidence on materials. We focus on experiments aiming at constructing 2D maps (radial-poloidal) of a measured parameter (e.g. density, plasma potential, ion energy). For each plasma current, a radial profile is measured by a diagnostic (reciprocating probe, reflectometry) located some distance away from the powered ICRF antenna. The diagnostic is magnetically connected to different poloidal locations of the antenna limiter depending on the value of the plasma current. This technique has been used in several devices like Tore Supra [Kočan 2008], WEST [Diab 2025], ASDEX Upgrade (AUG) and JET [Colas 2015] [Bobkov 2019]. Consistent patterns are generally observed, with potential peaks located at the antenna corners. Yet, the interpretation of these 2D maps as "instantaneous snapshots" can only be made under the following assumptions: (1) the investigated parameter remains insensitive to changes in plasma current (tilt angle); (2) RF sheath effects are insensitive to changes in plasma current; (3) variations are largely dominated by local effects induced by the RF sheath.

We examine to which extent these 3 implicit assumptions may be valid, based on RF simulations using Petra-M and COMSOL codes under the ITPA-DSOL framework. The coupled power and density are held constant over modeled scans of B field tilt-angle. Sheath potentials are calculated at the antenna limiters, and compared with experiments in AUG and WEST, also tracking the evolution of poloidal profiles with visible spectroscopy. Their variations, being sometimes beyond expectations of homogeneity, underscore the need for caution in interpreting current-scan-based 2D mappings. Neglecting tilt-angle effects may be acceptable within 1° to 2°, but modeling exhibit large discrepancies for broader ranges. Curvature effects are also discussed.

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Monday-04

Benchmark of the new release of FELICE solver in TOPICA code with the AUG 3-strap antenna

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The TOPICA [1] (Torino Polytechnic Ion Cyclotron Antenna) code is a computational tool used to simulate and analyze ion-cyclotron (IC) radio frequency antennas. These antennas are essential components in plasma heating systems within magnetically confined nuclear fusion experiments. TOPICA couples a plasma model through the plasma impedance matrix generated by a specialized plasma code named FELICE [2] (Finite Elements Ion Cyclotron Emulator).

Over the past two decades, the same version of the FELICE code originally developed in the 1990s and integrated with TOPICA in 2002 has been in use. While this version has proven reliable, its limitations do not align with modern coding standards. These constraints have prompted the integrating of an updated version of FELICE, released in 2023, into TOPICA. This new version improves accuracy, numerical stability, and computational efficiency.

This paper discusses the integration of the latest release of the FELICE code (2023) with the TOPICA code. A comparison is made between the current and previous versions of FELICE, starting from the plasma impedance matrix itself and, eventually, getting to the standard TOPICA outputs as the antenna input parameters, the coupled power and the radiated fields. The AUG 3-strap antenna is adopted for this comparison, showing very similar results for the two versions of FELICE.

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First simultaneous observation of co- and countercurrent fast ion losses in the ASDEX Upgrade tokamak

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Understanding the mechanisms responsible for fast ion losses is critical for the success of future magnetically confined fusion power plants. Towards this goal, a double pinhole collimator has been developed for a fast ion loss detector (FILD) [1] in the ASDEX Upgrade tokamak. This collimator features a mirrored design of the FILD geometry. FILDSIM [2] simulations were applied to optimize the geometry of the collimator diagnostic [3]. This new FILD open its operational window to simultaneous measurements of co- and counter- current fast ions generated by NBI and ICRH, enabling improving our understanding of their interplay with the objective of ensuring their confinement.

The commissioning of this new probe is carried out in H-mode plasmas with an on-axis magnetic field of B_0 = -2.5 T, and a plasma current of I_p = 0.7 MA. Co-and counter-current fast ion losses are observed with the same energy and trapped pitch angles ($|v_{II}/v| < 0.2$) for ICRH only regimes. Different populations of losses have been identified for both regions of the phase space. ICRH only TAE coherent fast ion losses are also observed. When NBI sources are applied, both ICRH and NBI losses are identified simultaneously on the FILD signals. The highest fast ion fluxes are measured at pitch ($v_{II}/v = 0.5$) and 93 keV, corresponding to the main injection energy of a radial NBI source.

[a] See author list of H. Zohm et al 2024 Nucl. Fusion **64** 112001 https://doi.org/10.1088/1741-4326/ad249d

[b] See the author list of E. Joffrin et al 2024 Nucl. Fusion **64** 112019 https://doi.org/10.1088/1741-4326/ad2be4

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Monday-06

Status of LHFW research in KSTAR and Future Plan

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The LHFW experiment has been conducted after installation of a 2.45 GHz RF system including a Slotted Waveguide Antenna, 10 kW RF power and components such as rectangular waveguide, alumina windows from 2024 KSTAR campaign [1,2]. The S-parameters measured without plasma with a vector network analyzer are in good agreement with the design values, and the coupling to plasma observed with a signal generator shows the expected tendency for plasma density variation regarding launching density of fast and slow wave branches. Analysis is progressing in more detail with PCS data and plasma diagnostic such as interferometer, Langmuir probe. Meanwhile, several kW power injection experiments are not performed due to SWA surface damage by mis-alignment with poloidal limiter probably originated from chamber contraction. An improved SWA is under design and it will be installed in the next campaign. More details will be reported in the conference with a future plan.

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Monday-07

Design of a Symmetric Traveling Wave Antenna for Fast Ion Production on DD Tokamaks

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Future burning plasmas will have appreciable populations of energetic fusion-born alpha particles. Here, a system is proposed to experimentally explore the effects of these fast populations on existing D-D tokomaks via selective RF heating of neutral beam particles through high harmonic fast waves with deuterium cyclotron harmonics of 5 to 9. The proposed system is a high field side symmetric center-fed traveling wave array antenna. Novel antenna features include the center feeding, high field side launch, and additional straps dedicated to image current cancelation for reduced impurity production.

The physics scenario was scoped on a selected DIII-D discharge for high single pass damping and good selectivity for ion damping over electron damping using the ray-tracing/Fokker-Planck codes GENRAY [1] and CQL3D [2]. The antenna design workflow developed includes a custom optimization tool built using a cost function that designs for launched power spectrum, image current cancelation, and low reflection coefficient. The tool uses a combination of finite element method (FEM) analysis software including COMSOL [3] for vacuum optimization, and the Petra-M FEM multi-physics framework [4] for cold plasma optimization, as well as Python RF network analysis packages. We show that the workflow presented here can produce a traveling wave array antenna with a desired power spectrum, reflection coefficient, and reduced impurity production potential via the cancellation of image currents, and that such an antenna placed on the high field side produces an attractive system to produce fast ions in the correct region of parameter space, warranting future more detailed studies. Recommendations are given for how best to take advantage of the traveling wave arrays' ability to passively impedance match using the optimization framework developed here.

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Monday-08

Analysis of O-mode Electron Cyclotron Heating in high density Super H-mode discharges in the DIII-D Shape and Volume Rise Campaign

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Capitalizing on the commissioning of a newly implemented divertor configuration designed to support improved plasma shaping and higher plasma volumes (SVR), some of the largest pedestal pressures ever documented on the DIII-D tokamak have been attained. Guided by recent promising calculations based on the peeling-ballooning framework [1], DIII-D enhanced its attainable plasma triangularity and volume to improve pedestal stability, aiming to access deep Super H-mode as predicted by the EPED model [2] to raise overall plasma performance. Precisely, Super H-mode enables reactor relevant core-edge integration by simultaneously achieving high density and elevated pressure in combination with a radiative divertor [3,4].

Experiments were conducted on DIII-D in the SVR shape to study a wide range of pedestal parameters and modify Super H-mode channel properties, among others through application of O-mode electron cyclotron heating with up to 3.4 MW. On axis electron temperatures reach up to 8 keV in these plasmas, with pedestal temperatures up to 2.3 keV. Due to the high densities of Super H-modes (pedestal density typically 0.7-1.0$\cdot 10^{20} m^{-3}$), the traditional X-mode heating is below the cutoff and hence O-mode heating is required. While the calculated absorption with TORAY is high in these plasmas (>90 %), the typical observed effects are rather local and not significant due to the high density. Select ECE channels near the deposition location (ρ=0.1-0.5) measure temperature increases of up to 400 eV for injected power of 3.4 MW. Our work demonstrates the successful injection of O-mode ECH power into a high-density plasma, thereby raising confidence of its applicability, but also demonstrates the reduced efficiency compared to X-mode injection on DIII-D.

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Monday-09

3D numerical modeling of the ICRF heating in SPARC within a toroidal wedge

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High magnetic field tokamaks, like SPARC [1], rely on ion cyclotron radio frequency heating (ICRF) to reach fusion relevant temperatures. The SPARC tokamak will have 14 ICRF antennas in 7 toroidal locations delivering > 20 MWs of power to the plasma. With each of the SPARC ICRF antennas varying between 40° and 60° of one another toroidally in addition to each toroidal location having two antennas stacked on top of each other poloidally, leads to the question of whether there will be wave interference between the antennas in this configuration. Interference between the antennas can lead to adverse effects such as the loss of heating efficiency. This research uses the 3D full wave finite-element code, Stix [2], to numerically model a toroidal wedge that includes 4 antennas in 2 toroidal locations to see how the waves propagate within both the scrape-off layer and core regions of the tokamak. For further optimization of the ICRF antennas, various phasings of the 4 antenna straps are simulated. Additionally, various plasma scenarios such as minority concentration amount are varied to see the effect of single pass absorption on the interference. Lastly, this research will also report on the newest updates to the Stix code to further improve ICRF modeling.

This work is supported by Commonwealth Fusion Systems.

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Monday-10

Fast Surrogate Modeling of ICRH Minority Heating at ASDEX Upgrade

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Ion Cyclotron Resonance Frequency (ICRF) heating is a key auxiliary heating system in present tokamaks and is planned for future devices such as ITER [1] and SPARC [2]. Full-wave solvers like TORIC, coupled with Fokker-Planck solvers such as SSFPQL [3], provide high-fidelity ICRF power deposition and heating profile simulations. Despite TORIC-SSFPQL being one of the fastest tools available to the community, its runtime remains a major bottleneck for fast transport modeling and hinders real-time applications.

We present several neural network based approaches to develop a surrogate model to speed up ICRF heating simulations of the hydrogen minority in deuterium plasmas for ASDEX Upgrade (AUG). Our results demonstrate that a neural network can achieve real-time-capable power absorption with high accuracy, aligning with findings from NSTX and WEST [4]. It is demonstrated that a pure TORIC surrogate model is limited in its applicability without the parameterization of a realistic minority ion temperature. Therefore, we introduce a surrogate model for the full TORIC-SSFPQL package, overcoming the limitations of TORIC-only models. Specifically, this model is able to capture the interplay between power absorption, the resulting alteration in the minority ion distribution function and the change of coefficients of the wave-equations.

Several possible directions for improving model accuracy are outlined, particularly by integrating additional model parameters to account for the electron density profile shape and equilibrium parameters such as the Shafranov shift, ellipticity, and triangularity, all of which have been shown to play a role in AUG plasmas.

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Monday-11

Development of LHCD system for long-pulse plasma on EAST tokamak

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EAST Tokamak is the first fully superconducting Tokamak dedicated to nuclear fusion research [1], and one of its goals is long-pulse operation with high performance, which is mainly supported by lower hybrid current drive (LHCD). In order to achieve the 1000-second long-pulse operation goal, high-power LHCD systems with a total power of 10MW were developed, laying a foundation for a series of achievements of EAST.

As time progressed, three LHCD systems were developed for EAST: the single-klystron pre-ionization unit, the 2.45 GHz LHCD system, and the 4.6 GHz LHCD system. The pre-ionization system was a temporary unit built in 2006 when the main engine of EAST was just built. This system played an important role in enabling EAST to generate its first plasma. The 2MW with f=2.45GHz LHCD system [2] was built in 2008 and upgraded to 4MW in 2012, helping the device achieve 400-second L-mode long-pulse plasma and 32-second H-mode operation [3]. The 6MW/4.6GHz LHCD system [4] was built in 2013, with which long pulse plasmas with 1000-second I-mode [5] and 1000-second H-mode operation were achieved. During the system construction process, control systems for power and phase were developed [6]. The flexible control of LHCD power and phase effectively supports the plasma operation of EAST.

The LHCD systems were introduced in this paper, where the typical problems during the operation and the corresponding solutions are also described, providing a good reference for the development of high-power long-pulse LHCD system in future nuclear fusion reactors.

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Monday-12

Effects of density turbulence on helicon wave propagation in the core plasmas

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Radio frequency (RF) wave propagation can be significantly affected by density irregularities, such as filaments in the scrape-off layer (SOL) or instabilities in the core plasma. In this study, we examine the impact of edge turbulence on helicon wave propagation using Petra-M. To analyze the effect of edge turbulence, we utilize a realistic background plasma derived from XGC simulations, which includes spatial density fluctuations at the edge. This focus is particularly relevant for the DIII-D configuration, characterized by edge density turbulence of the core plasmas in a wide pedestal QH-mode (WPQH-mode [1]). Previous simulations [2] confirmed that the generation of slow mode waves is minimized when the misalignment angle is in the experimental range near 5 degrees. Furthermore, the slow mode cannot propagate beyond lower hybrid resonance. Therefore, our focus is on helicon wave propagation. The simulation results indicate that edge density fluctuations substantially affect wave coupling due to scattering in the core. The insights derived from these simulations will inform upcoming tokamak experiments regarding helicon antenna coupling in long pulse scenarios.

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Recent advances on ion cyclotron resonance heating scenarios for Divertor Tokamak Test facility

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An analysis of the Ion Cyclotron Resonance Heating (ICRH) propagation and single-pass absorption in DTT (Divertor Tokamak Test facility) [1] heating scenarios has been performed by TOMCAT 1D full-wave kinetic wave equation code [2]. This analysis aimed to explore a wider parameter space compared to previous studies [3,4], with the goal of enhancing the understanding of DTT. Minority heating schemes at 3T and 6T were analyzed, simulating with different plasma species, including varying concentrations of D, ^{3}He, and H, across the frequency range from 50 MHz to 90MHz. A very promising reduced-field configuration will be presented. Furthermore, three-ion heating schemes for full-field configurations were investigated, building on the findings from [5], leading to the identification of an efficient scenario. An analysis with TORIC [6] was subsequently carried out, providing positive feedback on the results obtained with TOMCAT.

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Monday-14

Ion Cyclotron Heating in a Levitated Dipole Fusion Reactor

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OpenStar Technologies is pursuing the levitated dipole [1] as a highly modular, loosely-coupled system that leverages their expertise in high temperature superconductor (HTS) technology. Junior is the first levitated dipole built by OpenStar and follows on the success of the Levitate Dipole Experiment (LDX) at MIT [2]. Electron cyclotron heating (ECH) was the only power source for both LDX and Junior. The next generation experiment at OpenStar, Tahi (Māori for “first”), will demonstrate the generation and confinement of fast ions in a levitated dipole for the first time.

Ion cyclotron range of frequency (ICRF) heating is a leading candidate for energetic ion formation in Tahi. A frequency in the 10’s of MHz range will be used for minority heating on either H or He^3 ions with waves launched from a toroidal current strap located above the floating coil. Unlike a tokamak where the targeted cyclotron resonance is typically a ~vertical path through the center of the plasma, in a dipole the resonance location follows a C-shaped path from the separatrix to the center of the plasma. The value of |B| also varies significantly within the confined plasma resulting in a large number of cyclotron harmonics present in the low field region. Furthermore, levitated dipoles contain a “first closed flux surface” surrounding the floating coil, in addition to the traditional separatrix/last closed flux surface, which should not intersect the wave damping region in order to avoid heating the superconducting coil structure.

Simulation efforts have focused on applying three full-wave codes to levitated dipole configurations: TORIC[3], Petra-M[4], and AORSA[5]. Both TORIC and AORSA contain built-in assumptions regarding tokamak geometry, and accounting for the floating coil region in the center of the plasma requires modification of either the code (*e.g.* internal conducting wall boundary condition) or input parameters (*e.g.* electron density similar to that of a solid metal in the coil region), as well as the addition of a small ($\sim 10^{-4}$ T) toroidal field to avoid division by zero errors. Petra-M has the advantage of arbitrary 2- or 3-D geometry for the vacuum vessel, antenna, and floating coil.

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Monday-15

Progress in the pre-conceptual design of the auxiliary heating and current drive system for the Tokamak Energy Fusion Pilot Plant

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Tokamak Energy (TE) was selected to develop a pre-conceptual design of the Fusion Pilot Plant (FPP) based on a high-field spherical tokamak for the US Department of Energy's (DOE) Milestone-Based Fusion Development Program. The auxiliary heating and current drive (H&CD) system is being designed to aid the conventional ohmic heating to achieve and maintain fusion plasma parameters.

Initial investigations for the flat-top phase of plasma scenarios rely exclusively on the electron cyclotron (EC) waves as an auxiliary H&CD source. This work focuses on the parametric optimisation of the flat-top EC-based H&CD scheme via ray-tracing simulations for three plasma scenarios candidates, each having different magnetic fields and aspect ratios. The aim of the study is to maximise the current drive efficiency ζ. Here we shall present the results obtained and highlight the general trends shared among the presented designs, with dependencies on launcher locations and angles, frequencies, wave polarisations.

Overall, it is shown that EC waves are capable of efficiently driving current throughout the plasma volume, supporting the notion that EC waves can be the single auxiliary power source for the flat-top operation. These findings will inform future investigations of EC performance as the FPP design continues to mature.

Monday-16

Initial Measurements of the Current Profile in High Field Side Lower Hybrid Experiments

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Efficient, off-axis, steady-state current drive is an enabling technology for advanced tokamak power plants [1]. Lower hybrid current drive (LHCD) is one promising method due to its high efficiency [2]. LHCD is made further attractive by launching the waves from the high field side (HFS) as this improves penetration and absorption relative to low field side (LFS) launch [3]. The first-ever HFS LHCD launcher has been installed on DIII-D, with the first set of experiments to occur during the 2025 DIII-D campaign. These experiments will focus on measuring the non-inductive current profile via DIII-D's Motional Stark Effect (MSE) diagnostic. Initial results from these experiments will be presented, including the accuracy of current profile predictions and how this accuracy is affected by changes in plasma parameters. The spectral gap is predicted to be bridged by a double mode-conversion phenomenon, resulting in high single pass absorption compared to previous LFS LHCD experiments that relied on SOL interactions [4]. It is thus expected that simulation accuracy will be improved over previous studies.

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Monday-17

ICRH system for the HL-3 tokamak

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The HL-3 tokamak is a new medium-sized copper conductor tokamak at Southwestern Institute of Physics. A 6 MW ion cyclotron range of frequencies (ICRF) heating system is now under construction for HL-3, to provide central ion heating as well as generate energetic particles that is relevant to the burning plasmas. Radio Frequency (RF) range 25–50MHz with pulses up to 5s is considered. For a Deuterium (D) plasma, D(H) minority heating can be the main ion heating scheme in the start-up phase with RF frequency 33MHz at B_0=2.2T. The 2nd harmonic D heating can then take over when the bulk ion temperature is already high. The H–^{3}He–D, H–^{3}He–^{4}He and H–D_{beam}–D are among the potential three ion schemes to generate energetic particles. Heating NBI ions using 2nd harmonic D heating can also be another useful strategy in terms of generation of energetic ions. Simulation show energetic ions with energy at MeV level can be produced at HL-3 by ICRH solely using three ion schemes or by synergetic heating with NBI ions [1]. The ripple loss and the first orbit loss of these MeV energetic particles are in a few percent with plasma current I_p=2.5MA. For a Deuterium-Tritium plasma, the fundamental ^{3}He and the 2nd harmonic of Tritium with RF frequency 25MHz at B_0=2.5T are considered. The RF generator consists of 4 x 1.5MW transmitters to cross feed 2 antennas. The transmission line incorporates 3dB decouplers in order to divert the reflected power. The matching system is provided by a combination of stub tuner and phase shifter. Two 2-strap antennas with parallel wavenumber $k_{//} \sim 7$ m^{-1} are proposed. Several configurations of 2-strap antenna are compared with each have different feeder connections and ground positions. Antenna geometry is further optimized to get larger coupling resistance, lower the maximum voltage as well as the reflection coefficient at the RF feeder [2]. Simulation shows that each antenna could possibly couple 2.5MW with a tolerable maximum voltage at the transmission line, i.e., $V_{TML,\ max}$~26kV, under presumable HL-3 density profiles. The designs of essential RF diagnostics, i.e., B-dot probe, electrostatic probes, reflectometry, gas injection are also discussed [3-4].

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Monday-18

Validation of ERMES 20.0 Finite Element Code for MAST Upgrade O-X mode conversion

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In this study, we present validation results of the finite element code ERMES 20.0 [1], benchmarked against other numerical codes for the O-X coupling in MAST Upgrade (Electron Bernstein Wave -EBW- regime).

ERMES 20.0 simulations of the Ordinary-eXtraordinary (O-X) mode conversion process will be compared against experimental data from the upcoming EBW experiments on MAST Upgrade and against Finite Difference Time Domain (FDTD) codes [2, 3]. The numerical results will be evaluated in terms of mode conversion efficiency and propagation characteristics in a plasma slab benchmark with a linear electron density profile with varying normalized scale lengths k_0L_n = 1 to 25. Agreement with measurements and other codes will confirm the robustness of the finite element approach in resolving wave interactions with cold plasma.

This work illustrates that ERMES 20.0 is a reliable numerical tool for electromagnetic wave modeling in fusion plasmas. Future developments will focus on extending the code's capabilities to account for warm and hot plasma effects and contribute to the broader effort of improving predictive modeling for EM wave heating and current drive in the next-generation fusion devices.

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Monday-19

Parametric analysis for developing TWA antenna for WEST using minority heating ICRH technique

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Ion Cyclotron Resonance Heating (ICRH) is an effective plasma heating method used in magnetically confined fusion devices, and has been successfully employed in a number of Tokamak experiments around the world, including WEST at CEA-Cadarache. Traditional ICRH antennas typically operate at high power densities, often pushing the limits of permissible electric field levels. In this context, an innovative antenna design, the Travelling Wave Array (TWA) antenna, is proposed for WEST as a significant improvement over conventional ICRH antennas. The TWA antenna operates at reduced voltages within a large bandwidth while achieving enhanced RF power coupling.

This study presents a parametric analysis of the ICRH properties in WEST for minority heating scenarios. Numerical simulations of a TWA antenna using the EVE code [1] were performed to optimize the power spectrum and frequency range for Hydrogen minority heating. Results show that the maximum power absorption by hydrogen occurs at k//=8 to 10 rad/m within the frequency range of 52-62 MHz with eight straps arranged toroidally based on the available space. The poloidal size of the antenna's straps has a minimal impact on power partitioning among plasma species. Moreover, the results show that the best position for localized hydrogen heating and power deposition is between 0° and 22° above the midplane, while power absorption by H peaks at a 45° poloidal phasing angle between the two antenna rows, reaching its highest near the core. In addition, a comparative analysis with the classical WEST two-strap antenna design highlights the improved performance of the TWA antenna, which provides more efficient and localized ion heating.

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Monday-20

3D Simulation of ICRF Wave Heating on EAST Tokamak Based on Finite Elements Method

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Using the fast magnetosonic wave in an ion cyclotron range of frequencies (ICRF) to heat ions and electrons in the plasma core has been demonstrated an efficient way on EAST and other tokamaks. This poster reports the recent advancements in the first developed three-dimensional (3D) full-wave model of ICRF wave heating based on finite element methods (FEM). This model is an extension of previous work by Zhang *et al* [1-3]. To consider the 3D field effects, a realistic and detailed antenna geometry of the newly designed ICRF antennas on EAST is integrated with a 3D core plasma and a detailed 3D scrape-off layer (SOL) plasma geometry in this 3D full-wave model.

Compared with the full-wave codes based on Fourier methods, such as TORIC and AORSA, FEM-based full-wave codes have ability to accurately describe the ICRF antennas and other asymmetric geometries. Besides, FEM-based codes have the potential to mimic ICRF waves in the core and edge plasma regions, simultaneously. Combining these two computational regions is important to improve understanding of wave-plasma interactions and core-edge coupling. In this first developed 3D full-wave model, the core plasma, SOL plasma, and ICRF antennas are treated as a unified computational region. The development of this new 3D full-wave model is believed to perform valuable analysis of ICRF experimental results from EAST and provide important references for future experimental campaigns.

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Monday-21

Full-wave modeling of arcs within the ITER ICRF antenna for usage in the simulations and design of the RADAR Arc Detection system

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The ITER ICRF antenna [1] has been carefully designed to feature electrical fields below tolerable limits (typically, below 2 or 3 kV/mm depending on the location and orientation) when operating at a maximum voltage of 45 kV. In particular, this allows avoiding arcs. However, as for any high-power RF system, arcs can still occur in the ICRF antenna and its power feeding system, during normal operation but in particular during the commissioning.

Whenever an arc is detected, the RF power shall be immediately tripped (μs timescale) to avoid strong local energy distribution at the location of the arc. Undetected arcs are forbidden. To this aim, several complementary and redundant Arc Detection (AD) systems are foreseen to protect the ITER ICRF antenna. Among these AD systems is the RADAR Arc Detection (or RAAD [2]) which is currently under evaluation for implementation on the ITER ICRF system

To provide a first numerical proof of concept of RAAD, full-wave simulations of the ITER ICRF antenna and its power feeding transmission lines have been performed in the radar bandwidth of operation (up to ~ 350 MHz) with the help of CST Studio Suite and ANSYS HFSS commercial codes. In these simulations, the plasma loading has been approximated by a salty water load (ε_r=80 , σ=1S/m), while arcs have been modelled with both perfect electric conductor (PEC) cylinders or lumped element shorts. The obtained S-matrices have been then loaded and processed by the RAAD time-domain circuit simulations and signal processing calculations.

This paper describes the challenges to simulate the full ITER ICRF antenna with arcs, taking into account different loading conditions, different materials and different solutions for the arc insertion. It also provides a comparison of the scattering parameters with and without arcs.

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Monday-22

Energy Transfer and Spectral Evolution Induced by Parametric Decay Instability During the Injection of Lower Hybrid Waves

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Lower hybrid current drive (LHCD) is essential for achieving high-performance and steady-state plasma confinement. We have developed a program called Parametric Instabilities embedded Propagation and Evolution of RF Spectrum (PIPERS) that integrates the propagation and absorption of low hybrid waves (LHW) coupled with parametric decay instabilities (PDI) in the scrape-off layer (SOL) of magnetically confined plasmas. Utilizing a kinetic-fluid hybrid and electrostatic model of PDI [1] and the WKB ray-tracing method, the program can be applied to consistently analyze the energy transfer and spectral evolution of LHW during LHCD.

For the first time, we have successfully simulated the propagation and power absorption of LHW induced PDI in the SOL across different experimental scenarios, particularly in H-mode plasmas with gas puffing around the LHW antenna in order to improve the boundary coupling [2]. In such a cooled and dense SOL, the growth rate and convective amplification of PDI are significant with a strong pump wave, leading to strong scattering and nonlinear cutoff of LHW. Based on specific profiles of the fundamental parameters (background magnetic field, plasma density and temperature) in the SOL, our program can calculate not only the local coupling coefficient between the pump wave and decay waves but also the entire process of convective amplification of the sideband waves along their trajectories. Additionally, the program provides insights into the proportion of energy transfer and the evolution of the frequency and wavenumber spectra of LHW, with appropriate variations in the fundamental parameters of the SOL and the pump wave.

As density increases while other parameters are held constant, the energy of the pump wave diminishes notably around a certain “density limit”, indicating substantial energy transfer to the decay waves within the wave spectrum. The decay waves associated with discrete ion cyclotron quasi-modes (ICQM) and large parallel refractive indices ($n_z > 10$) account for most of the energy transfer and are readily deposited at the plasma edge.

Furthermore, to mitigate PDI-induced energy transfer and restore the LHCD efficiency, strategies such as increasing the background magnetic field and pump wave frequency, employing lithiation, and optimizing configurations with shorter antenna-to-plasma distance coupling can be effective.

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Monday-23

Research Results from the SciDAC-5 Partnership for Advanced Simulation of RF - Plasma - Material Interactions*

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Radio-frequency (RF) actuators are expected to play a crucial role in future fusion power plants. The primary challenge associated with the use of RF actuators in all frequency ranges is the interaction with the scrape-off layer (SOL) plasma and plasma-facing components. The research carried out in the Center for Advanced Simulation of RF - Plasma - Material Interactions leverages high performance computing resources and expertise in mathematics and scientific computing provided by researchers from the Advanced Scientific Computing Research community. A comprehensive suite of high-fidelity simulation tools is being developed that integrates RF-material interactions, realistic antenna structure and first-wall components, and RF wave physics across different regions. This includes examining relevant phenomena such as RF-sheath wall erosion, neutrals, impurity production and transport, and RF wave propagation and power deposition in the SOL, plasma edge, and hot core regions.

Our high-fidelity simulation tools leverage advanced software such as the open source, scalable MFEM finite element library [1], and thus are readily extensible to alternate magnetic field configurations. Their capability to simulate complex, realistic CAD geometries, allow these tools to generate customizable meshes and discretizations for tokamaks with detailed boundary features, to adapt to alternate magnetic field configurations such as those found in stellarators and mirror machines, and therefore, to support a full range of future pilot plant design studies.

Finally, these tools comprise both high-fidelity and reduced RF and material models with varying levels of physics fidelity, including machine learning-based models, that can be integrated into a Whole Facility Model.

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Monday-24

Progress in the analysis of the cavity resonances in the ITER ICRF antenna port plug.

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The ITER ICRF antenna plug can exhibit resonances at specific frequencies, some of them in the relevant range of frequencies for IC heating. These resonances have been identified as eigenmodes of the coaxial cavity, where the array plays the role of inner conductor, that can substantially increase the level of electric fields within the cavity as well as the level of RF losses in the B_4C neutron shielding tiles. As no grounding solution is considered, RF probes will be installed to monitor the RF fields in the port plug cavity and additional simulations of a realistic magnetized plasma are required to properly assess the integration (position, orientation) and their effectiveness.

Several numerical tools are available and have been extensively used to simulate the ITER ICRF antenna, such as TOPICA [2] or CST Microwave Studio (MWS), but none of these codes allow to combine realistic geometries, realistic magnetized plasma profiles, and lossy materials. Therefore in [1] we introduced the modal analysis in the cavity to decouple solving the computationally intensive plasma facing front of the launcher from the cavity. The method was used to reproduce the TOPICA electric fields (with gyrotropic plasma effects) obtained in a given vertical reference plane, in a MWS cavity (including lossy materials) using the multimodal scattering matrix of the cavity obtained with MWS. In this work we pursue the analysis by using the recently extracted magnetic fields [3] from the TOPICA modeling results, which not only provide an alternate way to compute the excitation spectrum of the cavity but also allow to confirm our first results. Accurate levels of RF losses can then be obtained from various plasma profiles and excitation of the antenna straps. Furthermore, these fields and their decomposition in amplitudes of cavity eigen-modes will allow to construct a S-matrix which links the 8 input feeders to the modes considered on the reference plane.

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Circuit modeling and analysis of different matching configurations for the DTT ICRF system

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In Ion Cyclotron Range of Frequencies (ICRF) systems, transmission lines, based on rigid coaxial cables, deliver the power from the radiofrequency generator to the antenna, matching the impedances to protect the generator, and maximize the power transfer to the plasma. A careful analysis of the matching network is of paramount importance to avoid dead bands in the operational frequency range and optimize the number of expensive components to be installed. This work performs such an analysis, employing a circuital approach to evaluate different matching configurations, checking the maximum voltage all over the cables and avoiding heavy electromagnetic simulation for a quick and easy comparison. Simulations were run for the ICRF system of the Divertor Tokamak Test facility (DTT) at the representing frequencies of 60, 75 and 90 MHz, using the circuit simulation tool of Ansys Electronic Desktop. Circuits have been simplified, using just coaxial cables as elements of the design and importing the antenna in the form of a scattering matrix. Three different matching schemes have been considered and analyzed regarding the highest voltage values over all components; the simulations of central and side straps of the DTT 3-strap antenna were conducted separately. After choosing which circuit fits the best in this application, the whole antenna matching circuit was modeled and studied. The analysis has proceeded by drawing the ELM-resilient schemes for two antennas and conducting several simulations. After optimizing the lengths of stub tuners and trombones to get the minimum power reflected to each generator, the results are expressed in terms of the corresponding maximum voltage and current over all the coaxial cables, and the power coupled to the antenna.

Monday-26

Power coupled to the slow wave resonance cone in the cold collisionless limit

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In this work we investigate the power coupled from an ICRF antenna part (e.g. a strap or a limiter) to the resonance cone. This near-electrostatic wave mode exists (and has been experimentally observed [1]) in especially low-density tokamak edge plasmas. In the conditions where this wave mode propagates, its electric field normal to the antenna surface is near-singular at the points where that surface is tangent to the direction of the cone [2] ("resonance cone tangency points"). In the cold collisionless limit, the normal electric field is singular at those points [3]. The question arises whether a finite amount of power is nonetheless coupled to the resonance cone, even in this limit, as is the case for other resonances of the cold plasma theory, e.g. the Lower Hybrid resonance [4] or the excitation of surface waves [5]. We answer this question in the affirmative: the power is indeed finite even in the cold collisionless limit. We show that the coupled power scales with the square root of the Stix parameters S and P, and that it is independent of the source's radius of curvature at the resonance cone tangency points.

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Monday-27

Revisiting the Explanation to Numerical Study of Coaxial and Surface Mode Excitation by an ICRF Antenna in Large Machines

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The excitation of surface waves (SWs) with toroidal wave number $|k_z|$ below the vacuum wavenumber k_0, with frequency ω greater than the ion cyclotron frequency ω_{ci}, within Alfven resonance (AR) regions by an ICRF antenna was numerically demonstrated in [1]. The fast wave field's spatial distribution was obtained by the semi-analytic code ANTITER II in plane geometry by summation of Fourier series over the toroidal and poloidal wave indices.

In this work, we revisit the explanation of numerical results of [1]. Two issues are discussed. First, radially localized SWs were observed in numerical results in [1], but the conventional dispersion analysis in that work cannot explain their localization. In fact, that analysis shows a propagative-evanescent-propagative structure which might be interpreted as a potential barrier, which is entirely incompatible with the presence of a localized wave. The reason is partly the use of the perpendicular component of the wave vector as a proxy for the propagative or evanescent nature of the waves, while radial localization must be explained by the radial component instead. We argue that a non-conventional dispersion analysis which accounts for the plasma density nonuniformity [2] can explain the radial localization, as such an analysis predicts the cut-off to appear in significantly denser plasma than was assumed in [1]. Figure 13 in [1] can be explained as follows: the wave is not (radially) propagative in that part of diagram where the plasma densities are $N \approx 10^{11} cm^{-3}$. The peaks are local ARs, i.e. rapid increases of the wave amplitudes in the vicinity of the points where $u = 0$ in the evanescent background.

Second, we explain the discrepancy between the expected and numerically observed number of ARs by possible resonance overlapping. The inequality $|k_z| < k_0$ is satisfied for nine toroidal mode numbers, $1 \le |n| \le 9$ for the plasma parameters applied in [1] ($k_z = n \times 0.1111\, m^{-1}, k_0 = 1.1\, m^{-1}$). However, there are only four positions visible in figure 13 in [1] where the wave field has peak amplitude. In our opinion, the discrepancy can be explained as follows. The ARs for the wave field harmonics with $1 \le |n| \le 6$ were situated sufficiently close to each other to overlap. Then four wave field peaks observed in figure 13 in [1] corresponded to three separate ARs for the wave field harmonics with $n = 7,8,9$ and one set of overlapped ARs related to toroidal wave indices $1 \le |n| \le 6$. This explanation agrees with higher amplitudes of the wave E_x and B_y fields in figure 13 in [1] in ARs which were farthest from the antenna and which related to the wave field harmonics with the smaller toroidal wave indices. The ARs related to toroidal wave indices $1 \le |n| \le 6$ could overlap in [1] if the effective collision frequency ν_{ef} was chosen sufficiently high: $\nu_{ef} \gtrsim 0.03\omega$.

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Direct Measurement of ICRF-Enhanced Plasma Potentials Using Reciprocating Emissive Probes on the WEST tokamak

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A new reciprocating probe head containing emissive and Langmuir probes was installed and commissioned on the WEST tokamak. The main scientific goals for the emissive probes are (i) to provide a first direct measurement on WEST of ICRF-enhanced plasma potentials responsible for enhanced sputtering and heat fluxes to ICRF antenna limiters, and (ii) to provide data to quantitatively benchmark RF simulation tools equipped with the sheath boundary condition and used to make predictions for future machines (part of a new ITPA-DivSOL task). An extensive experimental survey was conducted under various electrical settings of the 2x2-strap array (coupled power, toroidal phasing, left-right power balance) and plasma parameters (scrape-off layer (SOL) density, edge safety factor, H/(H+D) fraction). Plasma potentials exceeding 100 V were routinely measured when the probe was magnetically connected to the top of an active ICRF antenna.

At fixed SOL density and antenna RF settings, the plasma potential scales linearly with the RF voltages at the strap feeding ports or as the square root of the coupled power. Rectified potentials are extremely sensitive to SOL density, decreasing as the SOL density rises at fixed coupled RF power. This is especially evident in scenarios with intense MHD activity, where the edge density and sheath potentials evolve in opposite ways over fast time scales. Moreover, at a fixed RF frequency, the measured plasma potential increases with the core hydrogen fraction, H/(H+D), or with decreasing single-pass absorption efficiency. This suggests that unabsorbed ICRF waves can significantly impact the plasma potential in the SOL and should be taken into account when predicting ICRF-plasma wall interactions in future machines. The complex spatial structure of the measured plasma potential profiles will be discussed and compared with floating potential measurements from the (non-emissive) Langmuir probes. By scanning the edge safety factor, the magnetic connection from the probes to the antennas was changed, which enabled constructing a 2D R-Z map of the plasma and floating potentials around the top half of an active ICRF antenna. These potentials are peaked close to the top of the antenna box, where maximum heat fluxes and tungsten sputtering are usually observed [1]. Radially, the rectified potential peaks are found to coincide with the leading edge of the ICRF antenna limiters.

Upcoming experiments aiming to study ICRF-enhanced potentials and tungsten sputtering in the ITER-relevant regime where the slow wave propagates in front of the ICRF antenna will also be presented.

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Real-time estimates of the ICRF single-pass absorption for ITER

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Estimating the single-pass absorption (SPA) of a given ICRF heating scenario as function of the plasma and RF wave properties requires a multi-species hot plasma wave solver. Even in 1D and using truncated finite Larmor radius expansion for the plasma dielectric response, the simulations cannot be done much faster than in a couple of seconds. In ITER, it is envisaged to estimate the single-pass absorption of the ICRF waves in real-time to take preventive actions in case of poor absorption for avoiding increased plasma-wall interaction and enhanced heat loads on the plasma facing components. This work proposes a potential solution based on pre-calculated look-up tables of the ICRF single-pass absorption that can be interpolated in real-time by the Advanced Protection System (APS) and the Plasma Control System (PCS) in ITER [1,2,3] during the plasma discharge evolution. The tables can also be used as input for the Pulse Design Simulator (PDS) [4] and other modelling tools that require the ICRF absorption information for control and/or protection. The SPA tables are produced as function of the dominant parameters governing the RF wave propagation and absorption using the 1D TOMCAT code [5] for some ITER-relevant ICRH scenarios.

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Monday-30

ST40 as a testbed for non-inductive technologies for fusion pilot plants

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Although EC-only startup, rampup, and sustainment scenarios are being increasingly suggested for fusion pilot plants due to the immense operational benefits they would offer, there is also broad consensus that such techniques must be experimentally validated as soon as possible to conform with the ambitious timelines of these projects. With a multi-Tesla magnetic field, MW-level gyrotron, capability to achieve fusion-relevant temperatures [1], and plans to introduce lithium coating [2-3], the ST40 experiment [4] is uniquely positioned as a high-performance spherical tokamak on which to test and de-risk reactor-relevant technologies. Here we will describe ongoing preparations for the upcoming experimental campaign dedicated to exploring fully non-inductive and mostly non-inductive (with solenoid assist) operation on ST40. These include the installation and commissioning of a 104/137 GHz dual-frequency gyrotron supplied by Kyoto Fusioneering Ltd capable of delivering 2s pulses of 1 MW EC power, and the addition of a pellet injector that will enable EC-only experiments by removing the need for NBI fueling, among other applications. The flexibility afforded by the dual-frequency gyrotron will enable testing a variety of EC and EBW scenarios. Therefore, we will also present integrated modelling results to help design the planned experiments, with particular emphasis on further expanding the physics basis for EBW startup [5], X-1 rampup [6], and EC sustainment [5] in a high-performance spherical tokamak.

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Monday-31

Application of Solid-state High-power Microwave Source in MPCVD Reactor

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The microwave plasma chemical vapor deposition (MPCVD) technique uses microwave energy to excite gas and then generate plasma under specific conditions to deposit thin films on a substrate. Compared with traditional hot filament chemical vapor deposition (HFCVD) and direct-current chemical vapor deposition (DCCVD), MPCVD relies on electrodeless discharge to generate pure plasma that can avoid pollution caused by electrodes used in these methods. It has been recognized to be the most ideal solution for preparing high-grade diamonds.

In the wake of semiconductor technology development, optical grade diamonds, with the highest thermal conductivity in nature, are the fourth-generation semiconductor materials after carborundum and gallium nitride substrates. Carborundum and gallium nitride wafers (>6 inches) have achieved mass production currently. For the cost reason, there is an increasing demand for large-sized optical grade diamonds. The maximum size of diamonds that a conventional 2.45GHz MPCVD system can produce is only about 3 inches, while the effective deposition area of a 915MHz MPCVD reactor is 2-3 times that of the 2.45GHz system. Thus, it highlights the urgent need for developing the 915MHz MPCVD reactor.

Both domestic and foreign 915MHz MPCVD reactors use magnetrons. Despite the advantages of an early start in technology R&D and mature technical support, magnetrons still have deficiencies such as high operating voltage, limited operating time due to finite filament lifetime, and inflexible control, constituting an obstacle for the development of large-sized optical grade diamond technology increasingly. Solid-state microwave sources, as a new type of microwave source, have low operating voltage, a lifespan of tens of thousands of hours for solid-state devices and flexible control, which have gradually replaced magnetrons in some applications such as accelerators and colliders.

So far, there are no international reports on solid-state microwave sources with a power of tens of kilowatts in the operating frequency of 915MHz. Such a high-power solid-state microwave source requires the combination of dozens of solid-state branches, and the phase-amplitude consistency of the branch power amplifier has a great impact on the combining efficiency, which is also limited by traditional combination ways. Through continuous efforts, our team has successfully overcome the consistency issue of dozens of power amplifier branch modules, while increasing the power capacity of the high-frequency radial combiner, realizing a stable 915M/20kW power output. Eventually, plasma discharge was preliminarily achieved based on a self-designed cylindrical reaction chamber by our team.

Tuesday-01

Radio-frequency sheath simulations with non-adiabatic electrons

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The operation of Ion Cyclotron Range of Frequency (ICRF) antennas in magnetic fusion experiments frequently enhances plasma–surface interactions. This is particularly critical in high-Z machines, such as ASDEX Upgrade (AUG) with tungsten (W) walls, where tungsten release significantly contributes to radiative losses in the core plasma [1]. A major driver of these interactions is the rectified sheath potential near the antennas, which can become large enough to induce tungsten sputtering. Understanding the rectification mechanisms in relation to plasma properties and magnetic field orientation is thus crucial for future fusion reactors.

In this study, 1D3V Particle-In-Cell (PIC) simulations [2] with kinetic ions and electrons were employed to investigate the RF sheath properties of a hydrogen plasma confined between two conductive walls: a grounded right wall and a powered left wall driven by a sinusoidal voltage $V_{RF} sin(\omega t)$. The RF frequency ω spans 0.1 to 5 times the ion cyclotron frequency (4.57 MHz to 228.68 MHz), with plasma densities ranging from 10^{17} to $10^{19}\mathrm{m}^{-3}$. The magnetic field (B=3T) was oriented at angles between 90° and 1.25° relative to the wall.

The simulations reveal that the rectified sheath potential depends strongly on both the RF frequency and the magnetic field incidence. It increases from approximately $\frac{V_{RF}}{\pi}$ at high incidence angles and low frequencies to more than $\frac{V_{RF}}{2}$ in scenarios with grazing magnetic fields and/or high frequencies. This behavior is shown to correlate with the electron time of flight in the sheath region, which approaches the RF period for grazing field lines due to electron confinement along the magnetic field. The ion energy and angular distributions further illustrate the influence of the RF period relative to the ion transit time, providing key insights into sheath dynamics under varying plasma and field conditions.

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Tuesday-02

Design, Installation, and First Results from the Ion Cyclotron Emission Diagnostic on TCV

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TCV is a medium sized tokamak equipped with a large suite of plasma diagnostics, a versatile array of poloidal shaping coils, an electron cyclotron heating system, and a two source neutral beam injection (NBI) system. The NBI system is capable of a simultaneous injection of energetic neutrals in the co- and counter-current directions. The resulting fast ion (FI) populations are used to study a multitude of FI-driven instabilities. While plasma instabilities in the below 1 MHz frequency range can be observed via standard diagnostics such as soft x-ray detectors, magnetics, fast ion loss detectors, and reflectometers, fluctuations in the range of >10 MHz require a specialized diagnostic. For this purpose the TCV tokamak has been equipped with a dedicated ion cyclotron emission (ICE) diagnostic, following a similar approach to AUG [1] and W7-X [2]. The diagnostic consists of a pair of magnetic coils (8 turns, 177 μH, 16 mm long, 7 mm in diameter), oriented orthogonally to each other to detect magnetic field fluctuations in the toroidal and poloidal directions. The coils are housed in a stainless steel electrostatic shield with a slit and are installed on the torus low field side at the midplane location, behind graphite protection tiles. The coil electric outputs are routed to the vacuum feedthroughs via a pair of coaxial cables, at which point one of the outputs is grounded at the feedthrough, from the airside. The second output is then routed to a rectifying radio frequency wave detector and a fast digitizer in the diagnostics rack. The rectifying detector is sensitive to signals in the 10-100 MHz frequency range and can measure the signal amplitude and the phase difference between the two probes, while the frequency information of the signal is lost. The benefit of this detection method is that the output signal can be digitized at a "slow" speed (200 kHz in the case of TCV) with a low cost digitizer and the data volume per plasma discharge is low. The fast digitizer, on the other hand, preserves the frequency information, albeit with a larger data volume. For the case of TCV, a 250 MHz sampling rate digitizer has been temporarily used, while a dedicated 1 GHz digitizer is currently being implemented. First results of high frequency (>10 MHz) instabilities detected in the presence of energetic ions and electrons will be presented.

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Tuesday-03

Direct Cavity Combiner for High Power Solid State RF Transmitter

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Recent advances in fusion technology are accelerating the development and integration of commercial fusion devices. In order to maintain controlled fusion reactions, the plasma must be heated to great temperatures and efficient current drive must be established. The Ion Cyclotron Range of Frequencies (ICRF) have been demonstrated to be effective for these purposes, and for high magnetic field devices, 60-240 MHz radio frequency (RF) systems are envisioned.

Conventional vacuum electron device (VED) RF sources have a high life cycle cost with frequent maintenance, a large footprint, poor efficiency, and a fragile supply chain. Additionally, current solid-state solutions face similar footprint, costly combining and cooling obstacles that are resolved in DTI's solid-state approach.

Diversified Technologies, Inc. (DTI) is building a novel, patented, Direct Cavity Combiner (DCC) VHF Transmitter in a single high power, compact, and efficient amplifier under a Department of Energy Small Business Innovative Research (DOE SBIR)[1] grant for use as a high power solid state RF transmitter. This transmitter, designed to scale to 1.5 MW, is built from multiple RF amplifier modules combined a single RF cavity, with high efficiency, low combining losses, and output power directly proportional to the number of RF modules feeding the cavity. This technology is an alternative to conventional megawatt-class VED RF sources, and overcomes the limited frequency range, reliability, and supply chain issues associated with tetrodes and similar VEDs. Because it eliminates the power combining losses and large footprint typical of conventional solid state amplifiers, it enables high power output and density. The basic transmitter technology can be readily tailored over a wide range of frequencies, which makes it applicable in several technologies, including high power microwave, high energy physics, fusion, radar, and broadcasting. DTI has demonstrated the viability of the DCC concept at L-band and UHF.

Under the Phase I effort, the full-scale 120 MHz cavity was partially populated with 31 modules demonstrating 42.7 kW of output power at 81% efficiency. In the next Phase of funding, DTI intends to combine 384 RF transistors, to achieve an RF output of 500 kW and to produce 100 kW of Continuous-Wave (CW) power. In this presentation, DTI will report on the latest design and test results of the VHF RF cavity and modules.

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Tuesday-04

Construction of generalized quasi-linear diffusion coefficient using neural networks

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The quasi-linear diffusion coefficient (D_{QL}) derived from our machine learning framework shows good agreement with results obtained from GENRAY-CQL3D simulations, and the radial current drive profiles exhibit consistent behavior. These findings suggest that our surrogate modeling approach, which incorporates physical constraints, successfully replicates key wave–plasma interaction characteristics while significantly reducing computational costs.

Traditionally, calculating D_{QL} for wave–particle interactions relies on computationally intensive simulations (full-wave or ray tracing) coupled with Fokker–Planck solvers. To address this challenge, we developed a machine learning-based surrogate model that incorporates physical constraints derived from cold plasma theory and bounce-averaged damping effects.

First, we establish the propagation domain of Lower Hybrid Waves in $(N_\parallel, \psi)$ space by identifying the accessibility limit and determining upper and lower bounds of $N_\parallel$ using the Potential Power Deposition (PPD) method [1]. Subsequently, to construct the training dataset for our machine learning framework, we employ Latin hypercube sampling to systematically explore a wide parameter space.

Leveraging this database alongside the underlying physical principles (e.g. PPD), machine learning methods including Principal Component Analysis, Random Forest regression, and Residual Convolutional Neural Networks [2] are employed to design a physics-constrained machine learning framework capable of reconstructing D_{QL}.

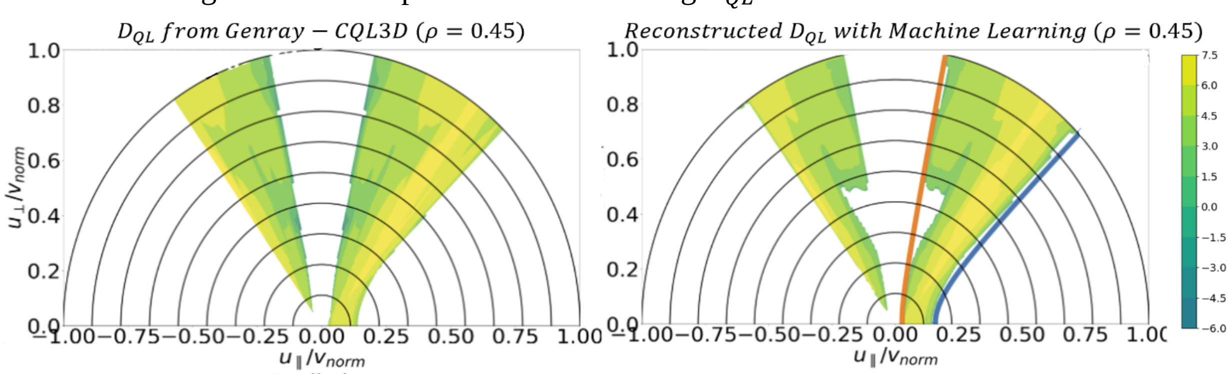

Fig 1. Left Panel: D_{QL} obtained from Genray-CQL3D simulation. Right Panel: Reconstructed D_{QL} using a machine learning model. The simulation were performed with parameters $N_\parallel = 2.3, n_{e0} = 2.5 \times 10^{19} m^{-3}, T_{e0} = 3.4 keV, and\ B_0 = 4.2T$. In the right panel, the orange line indicates the $N_\parallel^{upper}$ limit, and the purple line represents the accessibility limit in a flux surface averaged scheme.

1. Zhai, X. M., et al. "Theoretical analysis of key factors achieving reversed magnetic shear q-profiles sustained with lower hybrid waves on EAST." *Plasma Physics and Controlled Fusion* 61.4 (2019).
2. He, Kaiming, et al. "Deep residual learning for image recognition." *Proceedings of the IEEE conference on computer vision and pattern recognition*. (2016).

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Tuesday-05

Calculation of the magnetic field and its modal analysis in the port plug cavity of the ITER ICRF antenna

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The ITER ICRF antenna plug and the vacuum vessel port form an electromagnetic cavity that can exhibit resonances of the coaxial type in the range of frequencies relevant for IC heating (between 40 and 55 MHz). It can lead to substantially large electric fields in the gap around the plug, and it can increase the level of RF losses in the B_4C neutron shielding located at the back of the port-plug cavity. It was decided that the RF fields in the cavity will be monitored with RF probes. Numerical simulations are necessary to estimate the levels of electromagnetic field and RF losses in the cavity and assess the integration of RF probes.

Various simulation tools are available, but each one presents limitations that do not allow to solve the whole problem involving the antenna front with lossless cavity + lossy cavity (including B_4C tiles) + magnetized plasma in a single model. To overcome these limitations, a modal method described in [1] was proposed. It decouples the magnetized plasma-facing front of the launcher with lossless cavity including a multimode cavity port solving with TOPICA code [2] from the lossy cavity with B_4C tiles. It involves a modal analysis of the TOPICA fields computed on the multimodal port so that this can be used as input for CST Microwave Studio (MWS) to assess the cavity modes generated in the lossy cavity with B_4C tiles.

The previous work covered modal analysis of the electric field. This work is focused on reconstruction of the surface current, calculation of the magnetic field and its modal analysis in the port plug cavity. The method used consists of several steps. First, the antenna detailed model was imported into TOPICA code and run with several plasma profiles. Subsequently, surface current is reconstructed from TOPICA code outputs on the inner conductor of the cavity and the magnetic field in the gap around the antenna plug is calculated from the values of the surface current. After that the cavity model is run with MWS to calculate the cavity port base eigenmodes for the magnetic field and the multimodal scattering matrix of the lossy cavity. The magnetic field is then expanded in a series of cavity eigenmodes obtained from MWS. From this modal expansion the spectrum of forwards mode excitations can be evaluated [1]. As a last step, the MWS cavity will be excited with the spectrum previously obtained, and the magnetic fields will be computed in the lossy cavity. It will be used as input for calculation of the RF losses [3]. Furthermore, they will also help to understand the influence of non-reciprocal antenna load vs. reciprocal on the generation of cavity modes.

1. F. Louche et al., "Modal analysis of the fields in the ITER ICRF antenna port plug cavity," *AIP Conf. Proc.*, **2984**, 060007 (2023).
2. V. Lancelotti et al., "TOPICA: an accurate and efficient numerical tool for analysis and design of ICRH antennas," *Nucl. Fusion*, **46**, S476–S499 (2006).
3. F. Louche et al., "Progress in the analysis of the cavity resonances in the ITER ICRF antenna port plug", this conference.

RF Qualification of a Monolithic Additive Manufactured High Field Side Lower Hybrid Current Drive Launcher

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Additive Manufacture (AM) of Radio Frequency (RF) launchers is a key enabling technology for rapid production of complex RF geometries unachievable by conventional machining. High Field Side (HFS) Lower Hybrid Current Drive (LHCD) launchers [1] currently installed on DIII-D were AMed in segments using Laser Powder Bed Fusion (L-PBF); increased build volume now enables launchers to be printed as a complete monolithic structure [2], integrating phase shifters, aperture impedance matching elements, internal cooling channels, and traveling wave power dividers into a single solid structure that eliminates secondary machining and welding operations during launcher assembly. Monolithic AM reduces build time, cost, and eliminates single-points-of-failure in weld junctions. Chemical-Polishing (CP) is tested as a method of refining internal surface roughness in areas without mechanical access. Hot isostatic pressing prior to CP polishing consolidates sub-surface voids that result in negative skew porosity. Post-CP surface roughness is quantified on internal surfaces to predict RF loss. Signal-level RF measurements of monolithic AMed launchers are compared to COMSOL simulations in pre-CP and post-CP test articles to validate dimensional accuracy of the L-PBF and CP polishing processes in full-scale monolithic launcher modules. These techniques will enable future monolithic launcher production for replacement of existing modules, or addition of new modules in the DIII-D tokamak.

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[1] Seltzman, A. H., et al. *Nuclear Fusion* 59.9 (2019): 096003. https://doi.org/10.1088/1741-4326/ab22c8
[2] Seltzman, A. H., and S. J. Wukitch. *AIP Conf. Proc.* 2984, 100002 (2023) https://doi.org/10.1063/5.0162657

Tuesday-07

Parameter Study of Parametric Instabilities in Helicon Wave Current Drive Experiments

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The parameter dependence of the parametric instability (PI) growth rate during Helicon wave injection is investigated in this work. Using the hybrid fluid-kinetic model, the local PI dispersion relation and the analytical expression for growth rate was derived [1]. Cannel analysis was then performed using the PIPERS code [2], with calculations based on typical scrape-off layer (SOL) parameters.

The numerical results show that the main PI decay channels are low-frequency ion cyclotron quasi-modes and sideband lower hybrid waves, which agrees well with the experimental results from the DIII-D tokamak [3].

A higher electron temperature T_e can reduce the local PI growth rate, similar to the results for lower hybrid waves [4]. In the low-density regime, an increase in electron density n_e leads to a decrease in the growth rate of the PI; while in the high-density regime, where the density approaches the pedestal density, the PI growth rate increases with n_e. The dependence on the background magnetic field B_s is more intricate, since B_s modifies the wavevector channels of the PIs.

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[2] Z.H. Su, “Simulation research on parametric instabilities during the injection of lower hybrid waves into tokamak plasmas”, Ph.D. dissertation, Department of Engineering Physics, Tsinghua University, 2024

[3] R.I. Pinsker et al, “First high-power helicon results from DIII-D”, *Nucl. Fusion*, **64**, 12(2024)

[4] A.H. Zhao and Z. Gao, “Parameter study of parametric instabilities during lower hybrid wave injection into tokamaks”, *Nucl. Fusion*, **53**, 8(2013)

[5] M. Porkolab et al, “Parametric decay instabilities driven by high power helicon waves in DIII-D”, *AIP Conf. Proc.*, **2984**, 1(2023)

Tuesday-08

Design and optimization of a curved three-strap antenna for DTT ICRH system

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The Divertor Tokamak Test facility (DTT) aims at demonstrating possible solutions to the power exhaust issue to pave the path to DEMO. Here we present the numerical design and optimization of a three-strap ICRH antenna suitable to deliver Ion Cyclotron RF Power on DTT plasmas [1,2]. The launcher operates in the frequency range 60 - 90 MHz and here has been studied and optimized by using the commercial RF simulation software CST Studio Suite. The plasma is considered as an equivalent, high permittivity, lossy dielectric. Considering the mechanical and operational severe constraints of DTT, we firstly designed an antenna flat model with the objectives to optimize the structure for coupling a power ≥ 1.5 MW to the dielectric load with a progressive phase shift of 180° between toroidally adjacent straps.

The second part of the work regarded the design and optimization of a parametric curved antenna model in CST, which employs poloidal and toroidal curvatures suitable to better couple RF to DTT plasmas. The antenna curved model has been re-optimized in terms of coupled power and electric field values to match DTT requirements.

[1]. Ceccuzzi, S., et al., The ICRF antenna of DTT: Design status and perspectives. AIP Conference Proceedings. Vol. 2984. No. 1. AIP Publishing, 2023.

[2]. F. Romanelli, et al., Divertor tokamak test facility project: Status of design and implementation. Nucl. Fusion 64 (11) (2024) 112015, 29th IAEA Fusion Energy Conference (FEC 2023) Overview Papers

Tuesday-09

Recent experimental results of ECH in KSTAR

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KSTAR ECH power has been increased to ensure steady-state advanced operation of KSTAR. The planned ECHs are a total of six 1 MW, 300 s ECH systems, of which four 105/140 GHz dual frequency and one 170 GHz Russian gyrotrons have been installed and commissioned. The final unit, a multi-frequency gyrotron similar to Japan's ITER gyrotron is scheduled to be installed and operated early next year. This multi-frequency gyrotron system, even if only short pulse operation at a specific frequency is possible, will be very useful as it can be applied to KSTAR over a wide operating range from Bt 1.7 T to 3.7 T without vacuum breaking of EC TL. For advanced operation KSTAR, stable and reliable operation of ECH is very important. ECH assisted start-up and EC wall conditioning were routinely performed using one ECH with power of 0.5~0.7 MW. ECH is an essential tool for long pulse, high Ip operation, and NTM control to KSTAR operation. The maximum ECH power injected was approximately 1.5 MW by three ECH systems. EC power can be controlled by PCS which allows pre-programming of injection time, beam angle, and deposition position for optimizing operating conditions at start-up and at flat top of the discharge. This paper describes the development status of KSTAR ECH system and reports the operating results of ECH system in KSTAR.

Tuesday-10

First Experiments on 3D Plasma Calculations Using Positive Maxwellian Operator

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A new form of time-harmonic Maxwell's equations is developed on the base of the standard ones [1] and used here for numerical modeling. It is written for the magnetic field strength **H**, electric displacement **D**, vector potential **A** and the scalar potential Φ. There are several attractive features of this form. The first one is that the differential operator acting on these quantities is positive. The second is absence of curl operators among the leading order differential operators. The Laplacian stands for leading order operator in the equations for **H**, **A** and Φ, while the gradient of divergence stands for **D**. The third feature is absence of space varied coefficients in the leading order differential operators that provides diagonal domination of the resulting matrix of the discretized equations.

The model domain is rectangular parallelepiped. A lowest order 3D finite difference method is used for discretization. The antenna model chosen is the divergence free electric current directed in *y* direction. Plasma is 3D non-uniform and without magnetic field. The attempt to obtain similar solutions (both regular and singular) as in paper [2] had been made.

1. V. E. Moiseenko & O. Ågren, "Curl-free positive definite form of time-harmonic Maxwell's equations well-suitable for iterative numerical solving," *Plasma Physics and Controlled Fusion*, **63**, 124007 (2021).
2. J. Zhang et al., "2D simulation of the electromagnetic wave across the non-uniform reentry plasma sheath with COMSOL," *AIP Advances*, **9**, 055316 (2019).

Tuesday-11

Toward the Development of a Two-Dimensional Microscale Radio-Frequency Sheath Model for Grazing Incidence Magnetic Fields

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With the primary aim of accurately evaluating the radio-frequency (RF) sheath properties near curved wall surfaces inside fusion devices, we developed a two-dimensional (2D) microscale fluid model [1] based on the one-dimensional (1D) model proposed in Ref. [2]. After the improvement of the boundary conditions for the ion velocity and the ion density in the finite element scheme, we demonstrated several important physical behaviors such as the ion cyclotron admittance resonance similar to the results in Ref. [3] and the reversal of the ion flow for cases where magnetic field lines are almost tangent to the wall surfaces [4]. In addition, significant differences in the local admittance parameter between the 1D and 2D models were shown when the local radius of curvature of the wall surface is comparable to the non-neutral sheath width, primarily due to the difference in the electrostatic potential profile [4]. So far, all the numerical results in 2D geometry have been obtained under the assumption of Maxwell–Boltzmann (MB) electrons.

The validity of the MB relation is not retained at a high RF frequency or when the background magnetic field is tangent or nearly tangent to the surface. In such cases, the MB approximation for electrons may be replaced with an electron fluid model and a kinetic expression for the electron particle flux [5]. Here, we present our strategy for constructing a generalized 2D microscale model from the already-established 1D model [5] in which the two-fluid equations and a kinetic electron extension are incorporated.

1. H. Kohno and J.R. Myra, "A finite element procedure for time-dependent radio-frequency sheaths based on a two-dimensional microscale fluid model," *Comput. Phys. Commun.*, **291**, 108841 (2023).
2. J.R. Myra and D.A. D'Ippolito, "Radio frequency sheaths in an oblique magnetic field," *Phys. Plasmas*, **22**, 062507 (2015).
3. M. Rezazadeh, J.R. Myra and D. Curreli, "Resonance in radio frequency sheath admittance and enhanced impurity emission near the ion cyclotron frequency," *Nucl. Fusion*, **63**, 126024 (2023).
4. H. Kohno and J.R. Myra, "Investigation of two-dimensional radio-frequency sheath properties using a microscale fluid model," *Nucl. Fusion*, **65**, 026012 (2025).
5. J.R. Myra and H. Kohno, "RF sheath two-fluid model with a kinetic electron extension," 66th Annual Meeting of the APS Division of Plasma Physics, Atlanta, Georgia, USA (2024).

Full-wave simulation of ion cyclotron range of frequency heating in a mirror device*

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We are adapting the all-orders spectral algorithm (AORSA) [1] from tokamak geometry to axisymmetric magnetic mirror geometry. We will use the modified AORSA to study ion cyclotron range of frequency (ICRF) heating in the Wisconsin HTS Axisymmetric Mirror (WHAM) magnetic mirror device [2]. ICRF power will be used in WHAM to accelerate high-energy neutral beam injected deuterium ions with ion cyclotron absorption at the second to fourth harmonics of deuterium. At these harmonics, for the spatial scales present in WHAM, full wave simulations most accurately capture wave propagation, including transmission, reflection, and absorption at the cyclotron layer. The antenna is modeled as a single-strap with m = 0 excitation, where m is the azimuthal mode number. We present some preliminary results for wave coupling and heating.

1. E. F. Jaeger *et a*l, Phys. Plasmas **8**, 1573 (2001).
2. D. Endrizzi *et al*, Journal of Plasma Physics **89**, 975890501 (2023).

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Tuesday-13

Theoretical Scaling of the Density Limit of Lower Hybrid Current Drive

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The anomalous power loss at the scrape-off layer (SOL) plasma during lower hybrid current drive (LHCD) at high plasma density is considered as a major challenge for achieving effective non-inductive current drive at future tokamaks, leading to the widely concerned density limit problem of LHCD. In this work, the density limit of LHCD is scaled by both theoretical modeling and simulation of the energy deposition of the injected LH waves in the SOL region through parametric instability (PI), and we have obtained a scaling law of the density limit of LHCD consistent with previous experimental results.

The evaluation of PI power loss is presented under the framework of WKB analysis of waves and the electrostatic approximation of PI coupling. The model of PI saturation in an inhomogeneous plasma is extended to adapt to the scenarios where non-resonant quasi-modes are involved [1]. Subsequent calculation of the amplification factor of PI shows that the decay channels involving ion-cyclotron quasi-modes (ICQMs) dominate the anomalous power loss when density limit is reached, leading to an effective explanation of the broadening of the spectrum of the LH waves observed at EAST [2] and Alcator C-Mod [3].

Theoretical analysis reveals that the power deposition through PI is determined directly by the SOL parameter n_e^{SOL}/T_e^{SOL}, and the anomalous power loss through PI occurs only if a region with relatively high plasma density ($n_e > 10^{19}m^{-3}$) and low temperature ($T_i \approx T_e < 25\text{eV}$) exists in the SOL plasma. The gas puffing technique used to improve the coupling of LH wave significantly attributes to the forming of such a region. Both wall conditioning [4] and improving plasma current [3] might minimize the factor n_e^{SOL}/T_e^{SOL} to improve the efficiency of LHCD.

The density limit of LHCD scales as $n_{lim} \propto \left(P_{LH}/L_y\right)^{-2/3} \omega_0^2 B_0^{2/3} T_e$, indicating that the density limit can be improved by higher toroidal magnetic field, larger frequency of the LH wave and longer poloidal length of the antenna. For the scenarios where PI is suppressed to a relatively low level (for example, some of the EAST experiments with 4.6GHz LHCD [2]), the linear accessibility limit of LH waves might be reached at high plasma density before the PI power loss becomes critical.

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[2] M. H. Li et al., Experimental investigation on spectral broadening of lower hybrid waves with different frequencies in the EAST long-pulse plasmas, *Plasma Phys. Control. Fusion* **61**, 065005 (2019).
[3] S. G. Baek et al, High density LHRF experiments in Alcator C-Mod and implications for reactor scale devices, *Nucl. Fusion* **55**, 043009 (2015).
[4] B. J. Ding et al., Investigations of LHW-plasma coupling and current drive at high density related to H-mode experiments in EAST, *Nucl. Fusion* **55**, 093030 (2015)

Tuesday-14

Almost-off-the-shelf tools for ICRH modelling

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Studying the dynamics of the wave-particle interaction underlying Ion Cyclotron Resonance Heating in tokamaks or stellarators requires the development and solving of the relevant wave equation. Over the last decades, a large number of authors (see e.g. [1]) have contributed to pushing the realism of the models forward so a lot of detailed expressions – with various degrees of realism while concentrating on specific subtopics, each deserving attention - are available for exploitation. Similarly, gradually more sophisticated tools become available for solving partial differential equations in complex geometries. Combining the strength of both offers perspectives for the development and exploitation of "almost-off-the-shelf" tools with high degree of efficiency as well as richness of physics.

Crudely speaking, there are 2 main challenges when modelling wave heating in magnetic confinement devices while attempting to capture key aspects of the dynamics of the wave-particle interaction of RF created or fusion-born fast particle populations:

(i) the perpendicular dynamics is typically described in terms of guiding center variables and requires accounting for finite temperature (finite Larmor radius) effects, and

(ii) as the magnetic field cannot act on the particles along its field lines, the impact of the parallel motion on the wave-particle interaction is generally less localized than the perpendicular motion; when the magnetic field varies along the orbit, the associated modification of the dielectric response needs to be accounted for.

The present contribution summarizes some recent work aimed at adopting this mixed-tool approach, highlighting opportunities as well as bottlenecks. In particular, it is the somewhat more pen-and-paper counterpart of the contribution by V. Maquet [2] at this conference, the latter putting the focus on the numerical aspects.

1. P.U. Lamalle et al, "Integral dielectric kernel approach to modelling RF heating in toroidal plasmas", this conference.
2. V. Maquet et al, "Implementation of a fast 2D wave solver using the Budé method with NGsolve", this conference.

Tuesday-15

ECRH and ECCD Studies for DEMO and for a Volumetric Neutron Source (VNS)

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Electron Cyclotron (EC) waves are foreseen as a prime heating and current drive method in future fusion reactors. Their attractiveness is due to the limited space requirements of the EC launcher on the vacuum vessel (and thus a small impact e.g. on tritium breeding), the absence of wave-plasma coupling issues (EC waves propagate in vacuum), and the flexibility in targeting the absorption location (which allows a variety of physics applications). In the frame of the EUROfusion Consortium, physics and design activities for both a demonstration fusion power plant [1] and a volumetric neutron source (VNS) for component testing and qualification [2,3] are under way. Applications of ECRH and ECCD include in both machines bulk heating and current drive, and control of Neoclassical Tearing Modes (NTMs). The determination of the optimum launcher parameters for the envisaged applications is under way. In this contribution, selected modelling results obtained with the beam tracing code TORBEAM [4] are presented.

The EU DEMO design focuses on a machine slightly larger than ITER, with a major radius in the range of 8 m, a central density of the order of 10^{20} m^{-3} and a central temperature around 30 keV (the design point has not yet been finalized). For these parameters, a comparatively high current drive efficiency (around 50 kA/MW) can be reached in the plasma core [5]. The efficiency, however, drops when the deposition is moved towards the edge, making an ECCD-sustained steady-state scenario economically unattractive. VNS (baseline 2024) is a much smaller machine with major radius of the order of 2.5 m, higher density and lower temperatures. First investigations show that the smaller major radius compensates in large part the decrease of T_e/n_e, so that a central ECCD efficiency above 40 kA/MW is obtained. The design of the DEMO EC system is currently being adapted to the new baseline, while the integration of the allocated EC power under engineering constraints in VNS is still being assessed. The consequences of different design options for NTM stabilization are discussed.

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Tuesday-16

Evaluation of Neoclassical Impurity Transport affected by ICRH with a 4-D Fokker-Planck Code

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Several experimental results of Ion Cyclotron Resonance Heating (ICRH) mitigating the high Z impurity accumulation have been reported. As one reason of the mitigation, the strong poloidal asymmetry of tungsten density in a toroidal geometry, driven by plasma flows, can be reduced when ICRH is applied. Conventional approaches separately model the ICRH heating and neoclassical transport, where the heating is computed using the RF wave code/Fokker-Planck code such as TORIC-SSFPQL [1,2], while the neoclassical transport is evaluated with a neoclassical solver such as NEO [3] under the assumption of the time scale separation. However, using bi-Maxwellian distribution and the separate evaluation of the parallel electric fields may reduce the accuracy of impurity transport calculations. A more integrated approach is required to accurately represent ICRH-driven impurity transport dynamics.

The newly developed FP4D code addresses these limitations by simultaneously computing RF heating and neoclassical transport. It utilizes a two-dimensional real-space grid (radial, poloidal) and a two-dimensional velocity-space grid (perpendicular, parallel), and includes the drift term, the RF source term, and a fully nonlinear Fokker-Planck-Landau implicit collision operator for multiple species.

In this study, we performed the self-consistent computation of neoclassical impurity transport in several scenario of RF waves. For example, as shown in JET ICRH minority scheme for Hydrogen [4], the perpendicular temperature increase of the minority species changes the poloidal electrostatic potential, which was initially given by the strong flow, and redistributes the poloidal variation of tungsten.

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Tuesday-17

Near Field Analysis of the ICRF Travelling Wave Array Launcher for WEST

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Ion cyclotron resonant heating (ICRH) is a technique used to deposit energy to the ions in magnetized plasmas. It can provide several functionalities like core ion heating, sawteeth control, plasma start-up and landing, and wall cleaning. An ICRH system is characterized by three main components; a block that generates the radiofrequency (RF) power, a block that transports the RF power to the device vessel, and a block that launches the power in the plasma. The launcher system interfaces with the plasma. The waves in the ion cyclotron range of frequencies are evanescent in the thin scrape off layer plasma and become propagative once they reach a certain density. The launcher needs to provide a large field to bridge this gap. This large field in the region of the launcher could give rise to detrimental effects. A large electric field parallel to the background magnetic field could produce sputtering of impurities from the metal surfaces magnetically connected with the launcher, which diffuse into the main plasma and remove energy.

Large aperture fields are needed to inject the large amount of power required in plasma experiments. Several mitigation techniques have been envisaged to reduce the detrimental effect of these aperture fields. The spatial spectrum of the aperture electromagnetic field could be optimized to reduce the interactions with the wall. Traditional launchers, like the one installed on WEST, use Faraday screens to reduce the parallel component of the field. In contrast, a Travelling Wave Array (TWA) launcher shows reduced near fields and its special spectrum could be optimized to reduce the content of unwanted modes.

The TWA system proposed for WEST consists of two launchers. There are two rows of radiating elements displaced poloidally above and below the equatorial plane. A single vessel port is used to support and feed the antenna with cooling and RF power. Two limiters define the radial position of the antenna.

In this contribution we analyze the near field generated by the TWA launchers for different plasma configurations. We focus on the parallel component of the electric field on the limiters, as it is recognized as a proxy for the sheath rectification, thus related to impurity production. We show how this component of the field differs between the TWA and the WEST ICRH launchers, when calculated for the same plasmas. The phase difference between the two row impacts the field distribution. We discuss how this dependence could be used in future experiments to validate the models.

Tuesday-18

Experimental comparison of millimeter wave power monitor designs

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The ASDEX Upgrade electron cyclotron resonance heating system [1] uses multi-hole waveguide couplers integrated into the matching mirrors ‘M2’ for power monitoring. While the full Gyrotron output power (105 or 140 GHz, typ 1 MW) is essentially optically reflected at the M2 mirror, a small fraction of the order of several milliwatt is diverted via the coupler and transmitted via fundamental mode waveguides to a beam lead Schottky diode, which transforms the received power into a voltage signal. The system is calibrated by a calorimetric measurement, where short pulses of the full Gyrotron power are used to heat water.

On different experiment days, the calibration of some of the older power monitor designs showed variations of more than 20%, clearly above the uncertainty of the calorimetric calibration procedure. These designs employed couplers operating in 0^{th} order, which required coupler waveguides operating close to cut-off, due to the small (12.2 degree) angle of incidence at M2, and distances of adjacent holes < 1 mm.

With newer power monitor designs, these variations seem to be significantly reduced, even on the timescale of seven years, where hardware was unchanged. These later designs employed couplers in -1^{st} order, waveguides with standard dimensions, and large distances between adjacent coupling holes to scatter the probe power into the backward direction. Together with the changes in the coupler design, the low-power waveguide flange connections were significantly improved with respect to mechanical precision and stability, which allowed for a higher clamping torque. Individual statistical analysis per Gyrotron is ongoing.

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Tuesday-19

Development of a 400 kW ICRF system for the J-TEXT tokamak

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Ion Cyclotron Resonance Heating (ICRH) is a critical auxiliary heating mechanism in magnetic confinement fusion devices, playing a key role in increasing plasma temperature and improving confinement performance. In 2024, the J-TEXT tokamak was equipped with a new Ion Cyclotron Range of Frequencies (ICRF) system designed to operate within the frequency range of 20–35 MHz, delivering up to 400 kW of power for durations of 1 second. This system consists of two subsystems, each capable of generating 200 kW, and includes components such as generators, transmission lines, antennas, and diagnostic probes.

During the commissioning phase, the system demonstrated its ability to deliver the power at typical operating frequencies. Diagnostic tools, including high-frequency magnetic probe, electrostatic probe and voltage/current probes, were used to evaluate the coupling between ion cyclotron waves and the plasma. In the future, our focus will be on enhancing the performance of the system. This system will primarily be utilized for coupling experiments and to improve the parameters of the plasma on J-TEXT.

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Tuesday-20

Coupled Fokker-Planck/full-wave simulations of fast ion ICRF heating in a mirror plasma using CQL3D-m and AORSA*

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The CQL3D-m continuum bounce-average Fokker-Planck code is adapted for magnetic mirror plasmas [1] and is now routinely used in no-free-parameter classical integrated modeling of mirror devices [2, 3]. In the present effort, we report on coupling of CQL3D-m to a special version of full-wave code AORSA [4], also adapted for mirror plasmas [5]. The RF scenario involves ICRF heating of fast ions (FI) at 2^{nd}-4^{th} harmonic, where FI originate from neutral beam injected at 45 degree to the magnetic field. The scenario has been previously explored with ray tracing in GENRAY-C/CQL3D-m runs and shows an efficient ion heating near the FI bouncing point [6]. Different from the tokamak applications, CQL3D-m provides an evolving self-consistent ambipolar parallel electric field, which determines the shape of the loss cone and hence an accurate confinement time of both ions and electrons. Also, it includes a description of ion and electron sources and sinks (related to charge exchange and impact ionization) which are updated at every time step. CQL3D-m utilizes a fully nonlinear Coulomb collision operator that is important for significantly non-Maxwellian ion distributions typically established in mirror plasmas.

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Tuesday-21

Integral dielectric kernel implementation to model RF heating in toroidal plasmas

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As discussed in [1], recent theoretical and numerical treatments [2, 3] have sought to express the plasma radiofrequency (RF) response as a nonlocal integral operator formulated in configuration space. Analytical expressions of the integral kernels are available for Maxwellian particle species. The present contribution focuses on the concrete application of this approach.

We are exploring physical and numerical aspects through the development of a new “inhouse” finite element (FEM) code. It solves Maxwell’s equations in the frequency domain in the presence of the nonlocal kernel in 2.5D slab to lowest order in finite Larmor radius as a first step, i.e. with focus on wave dispersion effects along the equilibrium magnetic field and the associated fundamental cyclotron and Landau dampings. This integro-differential formulation is untypical of FEM [7]. We present preliminary results and describe the future developments.

In addition, the extension of advanced FEM libraries, such as the Psydac [4] code, to incorporate the nonlocal kernel is under study as one would benefit from all the features already available, such as meshing, efficient linear solver and visualisation tools among other. Psydac rests on tensor B-splines which provide basis functions with a high-level of regularity to treat complex geometries with FEM. The full 3D cold plasma run is reported and the sparsity pattern of the matrix with the nonlocal kernel in a 3D W7-X equilibrium is evaluated.

The realistic simulations of ICRH of warm plasmas involve very large linear systems that are less sparse than in cold models due to the nonlocal terms. Unfortunately, in the time-harmonic case, they are also highly indefinite, which prevents the use of standard iterative (Krylov) methods [8]. We will therefore combine iterative and direct solvers through dedicated domain decomposition approaches [9], exploiting the framework of the GetDDM and GmshFEM [6] codes. To this end, the computational domain will be divided into subdomains solved independently. On the interfaces between the subdomains, proper transmission conditions need to be developed. Each subdomain will then be solved directly in parallel and combined iteratively with its neighbours.

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Tuesday-22

Modelling of ion distributions under ICRH

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Fast ions accelerated by ion-cyclotron resonance frequency (ICRF) waves tend to form highly anisotropic distribution functions that play an important role in the collisional redistribution of the absorbed power to the thermal background ions and electrons, stabilising or destabilising both MHD eigenmodes and plasma turbulence. The modelling of these distribution functions normally involves solving computationally intense 2D or 3D equations.

One such solver is the LUKE code [1], which is widely used for modelling electron cyclotron resonance heating (ECRH) and lower hybrid current drive (LHCD) as well as runaway electrons. The code solves the linearized bounce-averaged relativistic Fokker-Planck equation for electrons and is now being extended to also model ion distribution functions, reusing the quasilinear framework and the collision operator from the electron model.

An alternative route was used in the recently developed Foppler code [2] that calculates a 2D distribution function by solving a 1D equation. Foppler combines a pitch-angle averaged Fokker-Planck equation with a model for the pitch angle distribution, presently based on the Dendy model [3]. The next step in this development is to benchmark the Foppler model against the ion-version of LUKE, both codes using wave-fields from the full-wave solver FEMIC [4]. In particular, it is planned to investigate different models for the pitch angle distribution, including the Dendy model [3] and the PION model [5].

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Tuesday-23

Ionization of Gas by Electron Cyclotron Waves for Efficient Startup

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For an effective functioning of fusion reactors, the startup phase of operations is of practical interest. The conversion of neutral gas to a plasma has to be efficient and controlled. The usual breakdown of gas in a tokamak, an ohmic startup, is through the use of a central solenoid which induces an electric field in the confined gas. In a reactor, the penetration of the electric field is through thick material walls which will be confining the fusing plasma. Instead of expending the stored energy in the solenoid on breakdown of a cold gas – an inherently inefficient process for thick walls – it is preferable to use the energy to control and confine a hot plasma. Among the different techniques that have been proposed, the use of radio frequency electromagnetic waves for breakdown is preferable as it is essentially independent of the magnetic field geometry. Furthermore, the same waves can be easily manipulated to impart energy and momentum to the ensuing plasma. Our research is to systematically quantify and optimize the experimental requirements for ideal breakdown in different magnetic field configurations. It is necessary to minimize the wave power and the time required for ionization of a hydrogenic gas. If the power is high, it can damage interior components in the device. If the power is low, the efficiency of ionization is reduced.

In the startup phase, the neutral hydrogenic gas is converted to a plasma predominantly by electron impact ionization. In a usual gas discharge, seed electrons are most likely created by cosmic ray muons. The energy of these electrons has to be above the threshold of ionization so as to further increase the population of electrons in the gas. An appealing mechanism for increasing the energy of electrons is through their interaction with electron cyclotron waves. The time needed to transition from a gas with some seed electrons to an ionized plasma depends on an understanding of two key physics points. The first point is related to the mean free path an electron has to traverse in a given magnetic field geometry, prior to an ionization event. It sets the minimum time required to complete the startup phase. The dependence of the mean free path on energy of an electron is well documented. The second point has to do with the time required for an electron to gain enough energy to enhance the probability of impact ionization. If this is time is greater than the time needed to negotiate a mean free path, then the breakdown time is determined by the energization process. We will discuss the physics of startup within the context of tokamaks and mirrors.

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First Implementation of a Fast 2D Wave-Solver Using the Budé Method with NGSolve

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This paper presents the first 2D implementation of the so-called "Budé" approach [1] to solve the integro-differential wave equation for ion cyclotron resonance heating (ICRH) as a high-order differential equation.

Up to now, two traditional approaches to incorporate finite temperature corrections to the hot plasma description have been followed. The first one relies on a truncated Taylor series expansion of the dielectric tensor assuming that the Larmor radius is much smaller than the perpendicular wavelength $\rho_L\, k_\perp \ll 1$, where ρ_L is the Larmor radius and $k_\perp$ is the magnitude of the perpendicular wave vector component. However, both fusion-born or RF accelerated high energy particles and short wavelength modes often violate this assumption. The second approach solves the all-FLR integro-differential wave equation in Fourier space [2]. Although powerful, that approach is computationally very demanding.

The Budé approach consists in fitting the dielectric tensor components in k-space with high-order polynomials. After Fourier inversion, this results in a high-order differential system in real space that can naturally be solved with any Finite Element Method (FEM) tools. The fit allows expanding the description to all relevant short wavelength modes, going beyond the truncated Taylor expansion approach. On top of this, the approach significantly reduces computational effort compared to brute-force Fourier-mode discretization methods.

The method was originally proposed and tested in 1D for radio-frequency waves in absence of poloidal fields. Using the open-source finite element software NGSolve [3], the Budé method is reproduced in 1D for benchmark purposes. The solutions of the high-order differential system approach those of the all-FLR equation when adopting sufficiently high-order fits, validated up to fourth order in this study. The work is then extended to 2D models, demonstrating the ability to simulate wave propagation in realistic geometries as well as linear wave transformation processes, e.g. the fast wave confluence to the ion Bernstein mode. Moreover, the 2D demonstration of the method opens up the possibility to future 3D applications.

The paper therefore advocates the use of a new open-source numerical tool to concentrate on physics development, and doing so presents a first step towards efficient full-wave computations for wave heating fusion applications.

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Tuesday-25

Argon pumpout by ICRF three-ion heating in Alcator C-Mod and WEST with projections for tungsten in SPARC

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Pumpout of argon (Ar) ions by ICRF waves has been observed in C-Mod deuterium L- and I-mode plasmas with substantial hydrogen dilution, with a reduction in Ar reaching up to 80% [1]. X-ray and VUV spectroscopy were used to infer impurity charge state density profiles to understand where in the plasma impurities were being pumped out and how many ions were removed. This is done using ImpRad, a Bayesian optimization framework that samples transport coefficient inputs into the Aurora impurity transport forward model until synthetic measurements match the experiment [2, 3]. The inferred density profiles agree well with the turbulent transport code TGLF when the ICRF is off, but significantly differ when the ICRF is on. It is observed that a local hollowing of the Ar density profile occurs at the radius where ICRF power damps as well as a decrease in Ar density across the plasma volume.

These Ar pumpout experiments were then repeated in WEST confirming that pumpout is triggered at an optimal hydrogen-to-deuterium ratio as well as illustrating the dependence on ICRF frequency. Full orbit (SPIRAL[4]) and Fokker-Planck (CQL3D[5]) simulations with the ICRF electric fields (TORIC[6]) support transport analyses for both C-Mod and WEST shots. In this poster we will discuss the physics of the observed Ar pumpout and the potential to utilize a three-ion heating scenario to actively pumpout tungsten in SPARC [7].

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Tuesday-26

Novel ICRH Coupler Configuration for High-Field Tokamak

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The SOL in high-field tokamaks, such as SPARC, has high-density plasma, in very close proximity to the wall and ICRH couplers. In this scenario, there is little threat from the highly cut-off slow wave, and little-to-no low-density tunneling distance for the fast wave's coupling. This is very different, for example, from previous tokamaks and from ITER. 3D coupling studies of the SPARC ICRH antenna show remarkable radiation patterns which appear to emanate, in a beam-like fashion, from the coaxial waveguide apertures, instead of launching, in plane-wave fashion from the straps. These unusual characteristics may be taken advantage of, to allow for significantly different coupler geometry, with far fewer plasma facing components, for the high-field tokamak configuration. In this poster, we explore one such novel coupler geometry, a vertical coax aperture (VCA) antenna.

In the VCA geometry, a vertical coaxial feed is shorted adjacent to an aperture, with no Faraday shield. This short provides a very small impedance, which matches the very low impedance of the nearby high-density plasma. We study the coupling as various parameters are varied, such as width, height, and shorting geometry, with aperture power coupling near 25% in one case.

The simulations also include non-linear RF sheath voltage estimates in the vicinity of the aperture, and also along side walls / limiters, magnetically coupled to the aperture. Such sheath voltage estimates can be used to estimate sputtering and impurity production. An important conclusion from this is that side wall sheaths can be reduced if the inner coax is shadowed by the outer coax, along field lines. For conventional antennas, this same effect is produced by having a box and Faraday shields surround the strap, e.g., considerably more, and more complex, plasma facing components than the simple aperture.

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First results of high-δ pulses with ICRH-only in ASDEX Upgrade in preparation for SPARC operation

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In preparation for SPARC [1] operations, high performance and divertor-compatible regimes need to be tested in current fusion experiments under the expected SPARC-like conditions. The quasi-continuous-exhaust (QCE) regime seems to be a promising H-mode regime, which may fulfill the challenging requirements for power plant operation by combining high density and ELM-free operation [2]. So far, it has been achieved on two metallic devices (JET and ASDEX Upgrade) [2,3]. Despite its attractive performance characteristics, it has not yet been experimentally confirmed that such a H-mode regime can be achieved with ion cyclotron resonance heating (ICRH) alone, which will be the exclusive auxiliary heating system on SPARC. This topic warrants further investigations as further next-generation devices such as ARC, ITER and DEMO will be operated temporarily or exclusively with external wave heating systems during their operation.

The ASDEX Upgrade (AUG) Tokamak is an ideal testbed for such SPARC studies due to its comparable dimensions, identical wall material, reactor-like design, powerful auxiliary heating systems and long experience in power exhaust studies. To investigate fusion plasmas under expected SPARC-like conditions, dedicated plasmas that employ ICRH as the only auxiliary heating system were tested at AUG. Scans in heating and fuelling were carried out to cover a sufficient range in the AUG operational space. The frequency spectrum analysis with magnetic coils shows the presence of a *quasi-coherent mode* (QCM) between separatrix and pedestal top in these discharges in a parameter range which is compatible with the separatrix operational space (SepOS) [4]. The energy confinement is lower in comparison to NBI-heated QCM discharges, possibly due to the high radiative fraction connected to the ICRH operation.

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Tuesday-28

Evaluating the Impact of Turbulent Tokamak Edge Plasma on Ion Cyclotron Wave Propagation and Absorption: A Stochastic 1D Model in a DTT Plasma Scenario

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Ion Cyclotron Resonance Heating (ICRH) is recognized as a powerful approach for plasma heating and current drive in tokamak experiments. In our earlier studies [1,2], we developed a 1D semi-analytical and numerical model to describe the propagation and absorption of an Ion Cyclotron wave in a DTT (Divertor Tokamak Test) plasma scenario, benchmarking it against more comprehensive full-wave codes such as TORIC [3]. Building upon this foundation, the present work introduces stochastic density fluctuations in the tokamak edge region. Such fluctuations, commonly observed in experiments [4], can significantly influence the antenna-plasma coupling by modifying wave reflection and altering power absorption profiles.

In this study, the 1D wave equation employed in earlier analyses is modified to include random perturbations of the local plasma density, characterized by specific correlation lengths and amplitudes. A Monte Carlo procedure generates multiple realizations of these fluctuations, and for each realization the wave equation is solved numerically to obtain reflection coefficients and power deposition profiles. These findings can pave the way for more accurate predictions of ICRH performance in forthcoming devices, where steep density gradients and turbulent fluctuations are expected. Future works will refine the fluctuation model based on experimentally measured correlation functions and compare the present 1D stochastic results with selected full-wave simulations in realistic tokamak geometries.

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Tuesday-29

High Field Side Lower Hybrid Current Drive Experiment in DIII-D Overview

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Initial high field side lower hybrid current drive (HFS LHCD) operation is planned for the DIII-D FY25 run campaign. The coupler design is optimized for high qmin, DIII-D discharges where efficient off-axis current at r/a~0.6-0.8 is desired. At full operating parameters, the HFS LHCD system is predicted to generate ~0.14 MA/MW coupled with a peak current density up to 0.4 MA/m^2 using $n_{||}$~2.7 at 4.6 GHz[1]. With HFS launch, LH waves are expected to have improved accessibility and penetration[2] resulting in single pass absorption.[3] In preparation for experiments, the HFS scrape off layer density profile has been characterized and found to have steeper profiles and less fluctuations than the low field side. This is expected to mitigate coupling and plasma material interaction challenges.[2] Furthermore, the HFS SOL density profile can be accurately predicted using global plasma quantities using machine learning. A compact launcher, located behind the center post protection tiles, is designed to distribute power poloidally utilizing a traveling wave, 4-way splitter and toroidally with a multi-junction that determines the launched wave spectrum.[4] To minimize reflections and electric fields in the coupler, each aperture has an integrated matching structure. The launcher is additively manufactured using a high strength, high conductivity copper alloy, GRCop-84.[5] In the future, the Laves-phase precipitate, Nb2Cr4, can replaced by another suitable Laves-phase precipitate compatible with neutron environment, for example Ta2V4. AM allows RF match elements to be embedded in each aperture and minimize reflections and electric fields in the coupler. With AM, the primary limitations imposed are component size and surface finish which were overcome through optimized joining techniques and incorporating chemical polishing. To minimize power conditioning, a vacuum window is located at the input of the poloidal splitter. The largest electric fields are in the phase shifters within the pressurized waveguide section. The vacuum window is a half wavelength alumina brick brazed into a CuCrZr window sleeve. To improve the brazing quality, the CuCrZr window sleeve is copper plated prior to the brazing to avoid issues associated with CrZr despite its low concentration. One challenge with HFS location is the 30R neutral beam strike during low density plasma operation that results in high heat flux, >10 MW/m2, for ~1 s. To avoid this heat flux, a split launcher is proposed to avoid the high heat flux region while maintaining power spectrum and directivity. The latest simulations, design and system status will be presented.

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Tuesday-30

Evaluation of Fast Ion Distribution in ICRF Experiments on EAST using a TORIC/CQL3D Full-wave/Fokker-Planck Code

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In the recent ion cyclotron range of frequencies (ICRF) heating experiment on EAST [1], new two-strap antennas are installed with an optimal parallel wavenumber ($k_{||}$), demonstrating improved coupling and a corresponding improvement in core heating efficiency. By doubling the strap distance to ~0.43 m, the peak $k_{||}$ in the antenna spectrum is reduced (14 $m^{-1} \rightarrow$ 7.5 m^{-1}), leading to an increase in coupling resistance from ~5 Ω up to ~15 Ω in the H-mode plasmas. As a result, core heating efficiency nearly tripled (~35 kJ/MW) as well. The demonstration of a long-pulse (> 300 sec), high-power (~ 2 MW/antenna) operation suggests that the new ICRF system has the potential to contribute to developing a high-performance plasma scenario on EAST.

Motivated by this evidence of significant core heating, this study proposes analyzing the self-consistent fast ion distribution in EAST plasmas using a newly developed modeling framework based on the coupled TORIC/CQL3D full-wave/Fokker-Planck code package [2]. It has been validated against the higher fidelity model, AORSA/CQL3D, using the Alcator C-Mod minority heating experiments. It is also computationally efficient with the finite Larmor radius approximation ($k_\perp \rho_i << 1$) employed in TORIC.

As an initial step, we aim to apply this framework to the EAST plasma and evaluate power absorption and the resultant fast ion distribution as a function of minority ion fraction, antenna phasing, density, and plasma current. It is important to evaluate the dependence on phasing to characterize both the old and new ICRF antennas on EAST, including the coupling effect. The fast-ion fraction is also a critical parameter in determining core confinement and stability. In CQL3D, a banana loss model is employed to estimate fast ion losses. The results will be compared with analytic theory and other available modeling tools. The study here will contribute to a better understanding of ICRF operation on EAST.

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Tuesday-31

Design and Development of ICRF Heating System on the Wisconsin HTS Axisymmetric Mirror (WHAM)

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The Wisconsin HTS Axisymmetric Mirror (WHAM) experiment is a record-setting magnetic mirror with $B_{max} = 17\,T$ and a variable mirror ratio $R_m = 15 - 50$ under operation at the University of Wisconsin-Madison [1]. Electron Cyclotron Resonance Heating (ECRH) at 110 GHz is used for plasma breakdown, while Neutral Beam Injection (NBI) at 25 keV seeds the plasma with a fast, sloshing ion population. This poster will review the Ion Cyclotron Range-of-Frequencies (ICRF) heating system, which is intended to launch a fast magnetosonic wave to accelerate the fast ion population at the second and third cyclotron harmonic resonance. Each heating system has a nominal 1 MW heating power, resulting in a very high $75\,kW/L$ of volumetric heating power density, enabling WHAM to explore the limits of the collisionless, classical confinement limits of the axisymmetric magnetic mirror.

2D axisymmetric finite-element method (FEM) full-wave simulations was performed with COMSOL Multiphysics. The propagation and cut-off characteristics of the fast wave within a bounded plasma column are explored at the frequencies between 3-30 MHz, utilizing the full anisotropic cold plasma dielectric tensor. The launch antenna design for the ICRF system is studied in both 2D and full 3D simulations.

Three different methods of including the kinetic effects in the full wave simulations will be described. First, the Bi-Maxwellian hot plasma tensor is used with wave vectors obtained from cold plasma bounded dispersion relations. Despite not being self-consistent, the model demonstrates strong damping of the fast wave in the presence of sloshing fast ion populations. Second, the fully non-local, integral-differential wave equation including the contribution of the hot plasma dielectric kernel [2] is solved, but computational challenges limit the fidelity of such a model. Finally, a time-domain, fully kinetic Particle-in-Cell simulation performed with WarpX was performed to study the wave propagation in a high temperature mirror plasma.

Finally, the engineering design of the ICRF antenna and transmitter will be reviewed together with the RF diagnostics, including two directional couplers, Rogowski coils, an axial and azimuthal B-dot probe array. Experimental results from the first campaign of WHAM will be briefly described, and a plan for low to high power ICRF operation will be presented.

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Wednesday-01

The Design And Implementation Of The Microwave Exciter For CRAFT LHCD System

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The design and implementation of the 4.6GHz microwave exciter specifically developed for the CRAFT (Comprehensive Research Facility for Fusion Technology) LHCD (Lower Hybrid Current Drive) system is presented here, with the primary objective of effectively driving klystron amplifiers. This exciter system is meticulously design and consists of 9 modules. There is one 4.6GHz solid-state source module, which contains a microwave phase-locked power source with 10-6 Frequency stability and an 8-channel power divider, it enables the efficient distribution of the microwave power. The power divider is engineered to divide the incoming microwave power into 8 individual channels, each of which provides an equal power output with a 0° phase difference. Each of the power amplifier module have a 3-stage GaN-Based power amplifier, a 6-Bit phase shifter and a microwave switcher. The utilization of GaN technology allows for better efficiency, higher power handling capabilities, and enhanced reliability compared to traditional amplifier materials. The output power range of each power amplifier module is designed to be from 0 to 10W, which can be adjusted according to the specific needs of the LHCD system. Similarly, the phase range is set to be from 0 to 360°, providing a full circle of phase adjustment possibilities. The power, phase and switches of each channel can be controlled independently and remotely. This control method also enables the LHCD system to adapt to different tokamak operating scenarios and optimize its performance based on real-time feedback, ultimately enhancing the overall efficiency and effectiveness to driving the klystron amplifiers.

Wednesday-02

An implicit-explicit time splitting strategy for the far SOL plasma fluid model with DG-FEM discretization

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We consider a far scrape-off layer (SOL) plasma fluid model of ions that is governed by a Braginskii-type model: a one-dimensional, nonlinear system of advection-diffusion equations coupled with a diffusion equation for neutral particles. Our motivation for studying this system arises from the coupling between the edge plasma and radiofrequency (RF) heating, where solving a far SOL plasma fluid model provides critical insights into edge plasma dynamics.

Numerical simulations of plasma fluid models require advanced computational techniques to achieve both efficiency and accuracy, especially when resolving the boundary layer in magnetically confined plasmas. In this work, we propose an implicit-explicit time operator splitting strategy that allows for an efficient solution algorithm, where the diffusive terms are treated semi-implicitly requiring only linear solve, while the advection part is handled explicitly using a strong-stability-preserving Runge-Kutta (SSP-RK3) scheme. This leads to a fully decoupled system in which the diffusion and advection sub-problems can be solved separately, simplifying the overall solution procedure and allowing for efficient parallelization, which is particularly relevant for exploring the impact of RF heating on the SOL plasma.

The main challenge of the discretization is due to the strong coupling between diffusion and advection, particularly through the boundary conditions. This makes implementation of such a scheme in an accurate and stable manner nontrivial. We discuss in detail how to split the equations and manage boundary conditions to maintain stability and well-posedness for each subsystem. We also describe a spatial discretization approach, which is based on the discontinuous Galerkin finite element method (DG-FEM) and present numerical results for a one-dimensional system.

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Wednesday-03

RF heating/current drive scoping study for the SMall Aspect Ratio Tokamak (SMART)*

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The SMall Aspect Ratio Tokamak (SMART), constructed and operated by the Plasma Science and Fusion Technology (PSFT) Laboratory of the University of Seville, had its first plasma in the end of 2024. SMART is a spherical tokamak (ST) with unique flexible shaping capabilities, which will allow it to operate at both negative and positive triangularity [1]. Installation and commissioning of a neutral beam injection (NBI) system for auxiliary heating is also ongoing. Therefore, this device is an ideal platform for testing RF heating/current drive (H&CD) schemes in a wide range of ST operational regimes. Moreover, RF actuators would also significantly expand the SMART operation regimes.

For this reason, in this work we survey a variety of H&CD schemes for anticipated SMART plasma scenarios. We consider the three planned phases of operations [2] with different magnitudes of magnetic field ($B\simeq 0.1$, 0.4, and 1 T) and the flexibility to operate in both negative and positive triangularity. Different H&CD regimes are considered such as helicon, lower hybrid, and electron cyclotron, taking into account different plasma shaping. Initial SMART operational phase with lower B field is expected to make the EC/EBW H&CD scheme more attractive whereas for higher field operation different and/or a combination of H&CD schemes can be considered. This RF H&CD scoping study is carried out by using a ray tracing model, which allows to quickly perform a series of scans on the main SMART plasma scenarios. Advantages and limitations in terms of antenna location and port availability will also be discussed.

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Real-Time Impedance Matching for Ion Cyclotron Resonance Heating System in EAST Using Capacitor-Based Feedback Control

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Ion Cyclotron Resonance Heating (ICRH) is a key technique for efficient plasma heating in fusion devices, requiring stable operation under high RF power. In the Experimental Advanced Superconducting Tokamak (EAST), the success of ICRH depends on overcoming impedance variations during long pulse discharges. This study presents a novel real-time impedance matching system designed to respond to antenna load fluctuations. The system utilizes capacitors for impedance matching, incorporating input and reflect voltage from the transmission line as a feedback parameter. By interfacing with a programmable logic controller (PLC), the system quickly adjusts motor controllers to minimize reflection voltage, achieving real-time impedance matching within 1 second. This rapid adjustment capability ensures the stability of high-power, long-pulse ICRH operations in EAST.

Additionally, a load-tolerant matching network with a 3-stub tuner and conjugate-T structure was developed for the ICRH system in EAST. This network operates effectively with a 30Ω to 50Ω transmission line, maintaining a low reflection ratio over a wide range of resistance values. This design supports high-power operations without the need for frequent impedance adjustments. The conjugate-T structure, with its λ/2 length difference in the two arms, mitigates current imbalance and optimizes antenna poloidal phasing, thereby improving the input impedance and performance of the antenna. The successful implementation of both the real-time impedance matching system and the new matching network demonstrates their effectiveness in meeting the operational requirements of the EAST experiment, paving the way for more reliable and efficient ICRH heating in future fusion devices.

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Operation of the Electron Cyclotron system in the ITER new baseline

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The ITER Research Plan (IRP) has undergone significant modifications based on the change of first wall material from beryllium to tungsten (W). Such a change, together with the constraint of achieving Q=10 for 300s with a neutron fluence of ~1% of the total ITER lifetime fluence, calls for a modification of the heating mix, now dominated by Electron Cyclotron Resonance Heating (ECRH), to ensure robust plasma operation and provide the most flexible operational space, w.r.t. to Ion Cyclotron Resonance Heating (ICRH) and Neutral Beam Injection (NBI).

This contribution gives first a brief overview of the revised heating mix in the new ITER baseline (more details will be provided by [1]), followed by a description of the planned new configuration of the EC system, i.e. the set of Upper and Equatorial launchers needed to inject up to 60-67 MW to ITER plasmas. Delivery of this increased power requires the installation of a second equatorial launcher for the DT phase, and the number of installed gyrotrons will be such that each specific beam entry will receive the power from a dedicated gyrotron. There will therefore be no need for switches to redirect the power from one launcher to another. The various contexts for which the EC system will be applied cover plasma initiation, wall conditioning, heating, current drive, W and MHD control, to achieve high-performance plasmas and ensure the completion of ITER milestones for the various phases of the IRP. The operational strategy for the various phases of the IRP will be described, in terms of optimal choice of launchers, applied power and polarization, taking into account the plasma evolution during a discharge. The operational conditions are adjusted to achieve the best plasma performance while ensuring that stray radiation does not damage in-vessel components.

The modelling work presented in this contribution relies on the TORBEAM beam-tracing code [2] applied within the Integrated Modelling and Analysis Suite (IMAS) [3] as post-processing on existing scenarios from the IMAS scenario database.

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Wednesday-06

Evaluation of neoclassical current and transport directly interacting with RF waves by developing a 4-D Fokker-Planck code

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The kinetic effects of RF wave heating and current drive in a toroidal geometry has been evaluated by the bounce-averaged Fokker-Planck equation solvers [1], in which the adiabatic invariants of energy and magnetic moment are assumed to have little change in a bounce time. The wave quasilinear velocity diffusion requires the sufficient decorrelation between kicks, so the bounce-averaged quasilinear diffusion is well applicable for the toroidal geometry. However, when the bounce time is comparable to the collision time (e.g. the trapped particle in tokamak core), the change of the adiabatic invariants by the RF wave within a bounce motion can provide the hidden kinetic effects. Furthermore, when the confinement time is low (e.g. magnetic mirror) or the RF wave resonance only locally occurs in a short radial and poloidal distance, the source effect can be sufficiently large to affect the drift-kinetic equation and make new neoclassical effects. We have developed a four-dimensional (radial, poloidal, perpendicular and parallel velocity) Fokker-Planck code (FP4D), which includes the many types of sources (RF quasilinear diffusion operators, neutral beam source, and alpha particles), the fully nonlinear Landau form collision model, and the parallel dynamics (parallel streaming, poloidally varying parallel electric field, and magnetic drift terms).

As some applications of the new code, we evaluate the neoclassical bootstrap (BS) current and transport directly interacting with RF waves. The poloidally localized up-down asymmetric cyclotron source without the wave momentum transfer [2] can results in the increase of the RF current drive efficiency (O(10) percent RF current for plateau regime), although it is irrelevant to the BS current. The up-down symmetric ion cyclotron source can modify the neoclassical distribution due to the quasilinear velocity diffusion correction to the pitch angle scattering collision and likely reduce the BS current (O(1) percent of BS current, O(1) KA/m^2 for $1MeV/m^3$ power).

The impurity accumulation mitigation by RF waves can be evaluated more self-consistently in this new code with the RF wave source in the drift-kinetic equations. The high degree of Tungsten poloidal asymmetric density due to the flows is reduced in the presence of the ion cyclotron wave quasilinear velocity diffusion on the main ions or minority ions, giving the mitigation of impurity accumulation. Instead of the bi-Maxwellian with the anisotropic temperature, the full velocity space distribution is used and the parallel electric fields are self-consistently calculated with all particle behaviors.

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Wednesday-07

Further upgrades of the ASDEX Upgrade ICRF system

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The original plans were to operate the tokamak ASDEX Upgrade for ten years. The flexibility of the experiment lead to the fact, that ASDEX Upgrade is now operating for over three decades. As the ICRF was ASDEX Upgrade's first additional heating system, which went into operation in 1992, regular upgrades of different ICRF components are mandatory for different reasons.

For example, due to lack of supply it became necessary to replace the pre driver stage tetrode by a widely available triode in the ICRF generators. This change reduced the frequency range of the rf generators from 30 – 120 MHz to 30 – 60 MHz, still covering the experiment requirements. A long time limitation was the behavior of the high voltage supply of the driver stage of the rf generator. A new power converter has been extensively tested on a test bed with promising results. Further upgrades include a redesign of the antenna vacuum pumping system to replace the maintenance intense cold heads. A redesign of the antenna vacuum feed through has been made to increase the voltage standoff. For the implementation of a new rf generator in the near future, which will extend the frequency range from 30 MHz down to 24 MHz, changes in the antenna matching systems were made on two of the four ICRF antennas, e.g. adding a third stub tuner. For the implementation of this rf generator additional coaxial switches, which were recycled from a shortwave broadcasting station, are integrated in the ICRF system.

Wednesday-08

Studies of Various Lower Hybrid Wave Launch Scenarios for Non-inductive Start-up on TST-2

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Efficient non-inductive start-up of a spherical tokamak may be achieved by using lower-hybrid waves that have high current drive efficiency. On the TST-2 spherical tokamak, three lower-hybrid launchers are installed at the outer-midplane, top and outer-off-midplane to explore efficient start-up scenarios [1]. Plasma current ramp-up up to 26 kA has been achieved so far [2], which is about a quarter of the Ohmically driven plasma current. The previously installed outer-midplane and top launchers were both found to drive current far off-axis that resulted in substantial orbit loss of fast electrons that limited the driven current [3].

The off-midplane launcher was newly developed to achieve robust current drive as well as to minimize fast electron losses. The off-midplane launcher driven plasma had higher electron temperature with modest x-ray radiation compared to the previous two launchers, qualitatively in agreement with the theoretical predictions [1,3]. The coupled power was limited due to frequent arcing at the antenna, however, that limited the driven plasma current. The antenna limiters were modified to reduce antenna-plasma interactions and increase the power handling capability of the off-midplane launcher.

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Wednesday-09

Extended benchmarking of the MFEM Anisotropic Plasma Solver (MAPS) 2D fluid transport code for coupled radiofrequency (RF) and fluid transport simulations.

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Understanding the way radiofrequency (RF) waves and the far-Scrape Off Layer (far-SOL) equilibrium plasma profiles interact is critical to ensure efficient delivery of power from the antenna to the core plasma. Observation of density modification, hotspots, and increased impurity generation during RF operation are likely driven by enhanced sheath voltages and can lead to increased power loss in the far-SOL. A high-fidelity fluid transport solver self consistently coupled with an electromagnetics wave solver is required to robustly describe these effects.

A difficulty arises when trying to balance conflicting mesh requirements. Alignment of the mesh to the magnetic field reduces numerical error caused by highly anisotropic transport, while antenna mesh should conform to its geometry to apply boundary conditions properly. Using a high polynomial degree finite element method allows for meshing flexibility, while also mitigating the numerical error in the plasma transport calculation caused by a non-aligned grid.

We present progress on the arbitrary polynomial degree, 2D finite element MFEM [1] Anisotropic Plasma Solver (MAPS), which includes benchmark results using a realistic single and double null magnetic field configuration. We discuss implementation of an additional solver, and the implications of these results for future code development.

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Damping RF waves with low reflection in simulations of slab or curved magnetized plasma: parametrization, verification and implementation of Bermudez Perfectly Matched Layers

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Most RF antenna simulations in magnetic fusion plasmas are computationally demanding: they model only the immediate vicinity of the radiating elements. Emulating radiation at infinity is then necessary at some boundaries of the simulation domain. To this end, the Perfectly Matched Layer (PML) technique amounts to stretching artificially the (real) spatial coordinates into the complex plane [1]. While polynomial stretching functions are abundantly used, their finite coordinate stretching induces residual wave reflection from the PML, even in the continuous limit. Following Bermudez et al. [2], this contribution parametrizes, tests numerically and implements unbounded stretching functions, whose only numerical limitations arise from the PML discretization: refining the mesh can reduce the reflections arbitrarily low, at the expense of a larger numerical cost.

We first parametrize the Bermudez stretching functions to attenuate plane waves oscillating spatially as $\exp(i\mathbf{k.r})$ in slab geometry. The relevant wavevector for the damping is the component k_n normal to the Plasma/PML interface [1]. We tune the PML to ensure low wave reflection coefficients in the spectral range [$k_{n,min}$, $k_{n,max}$] and we extensively quantify them using 1D Finite Element (FE) simulations. The PML behaves best if its depth is smaller than $k_{n,min}^{-1}$ [2]. In this case, the minimal number of FE layers necessary in the PML depth (hence the minimal numerical cost) is $N_{PML} \sim k_{n,max}/k_{n,min}$. We extend the formulation to PMLs in cylindrical geometry (R, Z, φ), where wave eigenmodes are Bessel functions [1]. The slab approximation applies for $k_n R_0 >> 1$, with R_0 the radius at the PML entrance. We propose corrections to N_{PML} as a function of $k_n R_0$.

We implemented slab radial and toroidal Bermudez PMLs in realistic multi-2D RF simulations, to attenuate the propagative Slow Waves (SW) parasitically emitted by the ITER ICRF antenna into a tenuous scrape-off layer [3]. We tuned the toroidal PML to damp the typical emitted SW $k_{//}$ spectrum, at a minimal numerical cost. The local SW dispersion relation at the plasma/radial PML boundary yielded a relevant $k_\perp$ spectrum at this place. While representing only a modest fraction of the simulation mesh, the PMLs damped efficiently the incoming resonance cone patterns. Bermudez PMLs are envisaged in future RF simulations of the WEST Travelling Wave Array [4].

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Wednesday-11

Development and Preliminary Experiments of ECRH/ECCD system on HL-3

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The major radius of HL-3 is 1.78m and minor radius 0.65m. The maximum toroidal magnetic is expected to be 2.2T and plasma current is about 2.5MA.[1] 1MA plasma current has been realized at the end of 2021 after one year device upgraded to divertor configuration since the first plasma at the end of 2020.[2] For the HL-3 tokamak, Electron Cyclotron Resonance Heating and Current Drive (ECRH/ECCD) system will be acted as one of the key plasma heating methods for central electron heating, current profile control and NTM suppression. A preliminary design of 8MW ECRH/ECCD system for HL-3 tokamak has been conducted and finished in 2016.[3] Key components of this system have been designed, manufactured and tested since then. At the end of 2022, a 7MW ECRH system has been developed on HL-3 tokamak for half a year installation, which consists five 105GHz/1MW/3s subsystems and two 140/GHz/1MW/3s subsystems. The 7MW high-power microwaves are produced by seven GYCOM 1MW gyrotrons, transported by seven Φ63.5mm evacuated over-mode corrugated transmission lines (TLs) and injected into plasma by three fast steerable launchers. All gyrotrons are settled in the RF heating hall which is in the south of HL-3 tokamak hall. The consideration of TLs routing and TLs installation was mainly focused on getting the best solutions of attenuation issues. The mode purity could be reached about 94% which was analyzed by phase retrieval method through testing the power distribution by thermal imager and the transmission efficiency is about 90% for ~40m TLs [4]. For the three launchers, the 4MW mid-plane, 2MW 1# upper and 1MW 2# upper launchers have been integrated on HL-3.[5] In the poloidal direction, more complicated push rod framework for three launchers was employed for NTM suppression in real time. The total response time of control activities is less than 100ms and the dynamic response time of mechanism for the full scan range is less than 200ms in the poloidal direction for full scan range. In 2024, 1.8MW output power of four 105GHz subsystems was injected into plasma. The power was deposited at the high field side and plasma heating effect was significant at the range of 1.5-1.7T toroidal field. With 800kW ECRH, the storage energy was increased about 50% when the plasma current is 500kA and toroidal field is 1.69T. With 1.2MW NBI, 0.3MW LHCD and 1MW ECRH, high confinement mode discharge was achieved when the plasma current is 1MA and toroidal field is 1.59T.[2] ECRH has been used in pedestal experiments on HL-3 tokamak and ELMy was suppressed successfully by 0.38MW/68GHz ECW. the electron internal transport barrier (ITB) was formed and sustained by applying ECRH heating during current ramp up phase and the core Te can be increased by ~20% with eITB. By optimizing the ECRH and NBI heating strategies, ion ITB was observed for the first time on HL-3 tokamak. The injection power of ECW will be increased step by step and more advanced experiments will be explored on HL-3 by ECW heating in the future.

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Wednesday-12

Design and Analysis of the Main Passive RF Components for the DTT ICRH System

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The Divertor Tokamak Test (DTT) is a new facility designed by the Italian DTT Limited Liability Consortium (S.C.a.r.l.) [1] aimed at validating an integrated solution for the power exhaust in support of DEMO [2]. DTT, hosted in the ENEA Frascati Research Centre, is presently in the initial phase of realization. The Ion Cyclotron Resonant Heating (ICRH) system, included in the DTT complex of additional heating systems, in its final configuration shall couple to the plasma up to 6 MW in the 60–90 MHz frequency range. The system is conceived in modular units. Each module, powered by four, solid state, high power, high frequency generators (SSPG), relies on a couple of launchers fed in parallel through 3-dB hybrid couplers [3]. One module (3 MW) is scheduled for the DTT initial operational phase. The upgrade to 6 MW (two modules) may be decided at a later stage.

All the main RF components of the coaxial cable transmission lines: stubs, 3 dB directional couplers, vacuum feedthroughs and phase shifters have been designed, simulated and optimized with the Ansys High Frequency Structures Simulator (HFSS) computer code.

The 3 dB directional couplers are mainly used to split the RF power generated by each individual SSPG to adequately feed the pair of three double folded current straps launchers of a ICRH module with a 0–π–0 phase configuration between straps. Branch line couplers with both one and two cells have been examined with satisfactory results.

Stubs and phase shifters are instead used for matching the 50 Ω Main Transmission Line (MTL) to the 30 Ω Vacuum Transmission Line (VTL). Innovative dielectric oil stubs and phase shifters have been examined with the aim of eliminating metallic sliding parts and thus increasing the reliability of these components.

Finally vacuum feedthroughs are used to isolate the air pressurized transmission line sections from high vacuum transmission line sections. Two different ceramic windows: corrugated disk windows and conic windows, have been examined.

Results of the optimization of these components are presented and discussed in the paper.

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Wednesday-13

Validation of ERMES 20.0 Finite Element Code for JET A2 antennas coupling studies

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In this study, we present validation results of the finite element code ERMES 20.0 [1], benchmarked against experimental data and other numerical codes for Ion Cyclotron Resonance Heating (ICRH) coupling studies for the JET A2 antennas.

ERMES 20.0 will be benchmarked against the well-established TOPICA code [4]. Simulations focus on the coupling properties of the JET A2 ICRH antennas. The comparative analysis will highlight the possibilities of ERMES 20.0 in contributing to sheath rectification and wave-edge plasma interaction modelling. The agreement between ERMES 20.0 and TOPICA will reinforce the validity of the finite element approach for ICRH antenna-plasma coupling studies.

This work illustrates that ERMES 20.0 is a reliable numerical tool for electromagnetic wave modeling in fusion plasmas. Future developments will focus on extending the code's capabilities to account for warm and hot plasma effects and contribute to the broader effort of improving predictive modeling for EM wave heating in the next-generation fusion devices.

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Experimental observation of power dissipation of ICRF waves at the edge on EAST

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Ion cyclotron radio frequency (ICRF) wave heating of plasma plays an important role in both existing and future magnetic confinement devices. However, the fields generated by ICRF antennas at the edge of the plasma can cause impurity sputtering. In a full-metal wall environment, these fields can reduce plasma confinement and even lead to discharge disruptions. For ITER, the initial 10 MW ICRF wave will operate under the metal wall torus. Further research is needed on how to effectively reduce the level of wave induced impurities and maintain effective heating. Regarding the impurity phenomena caused by ICRF antennas, theoretical explanations include slow wave dissipation [1], standing waves formed by edge fast waves, scattering caused by drastically changes in edge density [2], and the existence of surface waves [3]. There may be multiple physical mechanisms coexisting in the issue of edge wave dissipation, and this process increases the difficulty of experimental research to identify the physical processes.

A series of experiments on the edge power dissipation of ICRF heating for the all metal wall had been carried out on EAST. We actively separated the lower hybrid resonance (LHR) layer and the steep density region, adjusted the antenna toroidal wave spectrums, used multiple sets of probes to measure the near field and far field, and observed the changes in impurity concentration. The experimental results show that there is a strong electric field in the steep plasma region, and there is a certain amplitude fluctuation of the electric field at the LHR layer at low toroidal modes. The experimental results indicate that the power loss at large edge density gradient and the LHR layer has global characteristics, and confirm the existence of these two theories and also give a comparison of the intensities of their influences on the field. In addition, finite element simulations with the hot plasma load including the scrape-off layer region and the antenna model have been done. The simulation results clearly show that when the toroidal mode exists, a strong electric field exists in the region at a large edge density gradient region and near the lower hybrid resonance layer.

The dissipation of waves at high density gradient and lower hybrid wave (LHW) in the edge plasma have been experimentally observed, which will provide a direction for the optimization design of antennas in improving the application of ICRF waves in all metal wall devices.

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Wednesday-15

GENRAY/CQL3D modeling for NSTX-U with combined high harmonic fast wave and neutral beam heating and current drive

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The results presented here on high harmonic fast wave (HHFW) heating and current drive in NSTX-U are part of a broader effort with the goal of predict-first modeling and experimental demonstration of fully noninductive scenarios in NSTX-U. The study aims to make predictions and find optimal parameters for applications of the 6 MW 30 MHz HHFW system. Parameter scans of launched wavenumber, effective charge, plasma density and temperature are performed based on reference NSTX-U shots. For speed of calculation, the ray-tracing code GENRAY was used for the parameter scans. In many cases, a non-negligible power fraction is predicted to be damped on the fast ions. The Fokker-Planck code CQL3D coupled to GENRAY was used for a more accurate description of the fast ion population, as well as allowing the modeling of quasilinear effects such as HHFW-induced modifications of the distribution function. Quasilinear effects start manifesting themselves at coupled powers on the order of 1 MW. HHFW absorption on fast ion distributions, approximated with equivalent Maxwellians, is compared to the more accurate description of HHFW absorption on slowing down distributions [1], through both analytical theory and simulation. Fully self-consistent simulations where the heating and current drive due to HHFW is iteratively fed back into the transport modeling are underway.

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Wednesday-16

Impact of ECRH on runaway electrons in the TCV tokamak

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Runaway electrons (REs) present a serious risk to tokamak fusion reactors. In the event of a disruption, the presence and size of a pre-existing RE seed may be critical in the formation of a RE beam, which may cause significant damage to the first wall. Central electron cyclotron resonance heating (ECRH) applied in the Tokamak à Configuration Variable (TCV) was found to reduce the RE seed density by up to three orders of magnitude within just a few hundred milliseconds [1]. These investigations have been extended by varying the applied ECRH power between 100 kW and 1500 kW and measuring its impact on the RE population. A threshold in ECRH power $P_{ec} \gtrsim 250$ kW is determined above which RE avalanche generation cannot compensate RE losses, resulting in a significant reduction of the RE density. Above this threshold, the RE expulsion rate and consequent RE current reduction are found to increase with applied ECRH power. At moderate ECRH power levels $250 \lesssim P_{ec}\ (\mathrm{kW}) \lesssim 600$, the decrease in RE density results primarily from a reduced loop voltage due to thermal electron heating. At high power levels $P_{ec} \gtrsim 600$ kW, RE transport is found to increase linearly with ECRH power. Proposed for heating, current drive and instability control in fusion reactors such as ITER and STEP, ECRH is a promising tool to expel RE seeds and prevent or limit RE beam formation should a disruption occur later.

J. Decker, et al, “Expulsion of runaway electrons using ECRH in the TCV tokamak”, *Nucl. Fusion,* **64** 106027 (2024)

Wednesday-17

Development of an actively cooled radio-frequency test bench for the ITER ICRF windows

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Radio-Frequency (RF) windows (or feedthroughs) are under development for the ITER Ion Cyclotron Range of Frequencies (ICRF) system (40-55 MHz). In the frame of a contract with the ITER Organization, CEA will carry out the performance testing of the two types (called rear and front) of feedthroughs as delivered by the IO. To test these RF windows units, an actively cooled radio-frequency coaxial resonator was designed and is currently under construction.

This testbed is specified to reach 50 kV peak maximum voltage or 2.5 kA peak maximum current at the middle of the test articles in continuous-wave operation (3600s RF pulses with a ¼ duty cycle). The operating frequency of the resonator is set to 55 MHz. The conditions of the tests are 0.6 MPa (dry air) at one side from the RF windows, vacuum (10^{-4} Pa) environment at the other side of the front window and 0.3 MPa (dry air) at the rear window. As it is not possible to maximize both voltage and current at a given position of a resonator, two resonator configurations are foreseen, for each of the front and rear windows (hence 4 different configurations):

(1) DUT (each of the front and rear windows) under maximum voltage to validate the voltage standoff of the RF windows and to validate their cooling under maximum dielectric dissipation in the ceramics,
(2) DUT (each of the front and rear windows) under maximum current, to validate the RF windows cooling under maximum metallic losses.

The test bench was designed to be modular, allowing the positioning either the rear of the front windows for voltage or current RF test, hence leading to cost reduction.

This paper summarises the characteristics of the testbed and details its engineering design (RF modelling, thermomechanical and fluid dynamic analysis), as well as the expected test procedures.

Experimental characterization of 2nd and 3rd harmonic ICRH with FIDA and neutron measurements

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ICRF heating is planned for future fusion devices such as ITER and SPARC. To garantuee their success, it is crucial to study the physical mechanisms and verify modeling codes already in present-day devices to ensure reliable predictions.

ASDEX Upgrade is very well equipped to contribute to this important task due to its well-developed suite of fast-ion diagnostics. In particular the Fast Ion D-Alpha (FIDA) spectroscopy [1] can provide measurements of the fast-ion phase space with simultaneous velocity-space and real-space resolution. However, measuring the high-energy tails caused by RF is challenging, because few particles are accelerated to high energies, resulting in low velocity-space densities, and consequently low FIDA signals [2]. For this reason, measuring the 1st harmonic minority ions is not possible, but discharges with optimized parameters allow to achieve good FIDA measurements for 2nd and 3rd harmonic heating (of D). This also allows to use neutrons as an additional diagnostic, lacking spatial resolution but being very sensitive to high-energy tails.

Within a single discharge, 2nd and 3rd harmonic heating was compared (applying 36.5 MHz and 55.1 MHz ICRH respectively) with phases without ICRH while NBI was running through. A good quantitative measure of the effectiveness of the ICRH heating is the signal enhancement during the ICRH+NBI phase, compared to the NBI-only phase. Here, we obtained the intriguing result that the FIDA signal enhancement is stronger in the 2nd harmonic heating case, whereas neutron enhancement is stronger for 3rd harmonic heating.

Neutron rates are sensitive to highly energetic fast ions, and FIDA is more sensitive to fast ions in the vicinity of the NBI injection energy (because of the involved fusion & CX cross-sections, respectively). Hence, a possible explanation is that in the 3rd harmonic case the distribution function (d.f.) of the fast ions is higher at high energies and lower at low energies (wrt. 2nd harmonic heating). We have investigated two hypothesis how this could be explained:

The first was, that at 3rd harmonic heating, there is simply more power available (no competition with minority H absorption), driving the fast-ions towards higher energies and therefore decreasing the FIDA signal. This hypothesis could be ruled out by conducting power scans both for 2nd and 3rd harmonic heating. They showed that the FIDA signal (i.e. the “slow” fast ions) increases just as the neutrons with increasing RF power – there is no roll-over where the FIDA signal would be higher at lower ICRH power.

The second hypothesis is that this is an intrinsic feature of the 3rd harmonic heating, e.g. related to the Bessel functions which describe the strength of the RF-induced velocity diffusion wrt. the fast ion Larmor radius. Here, 3rd harmonic is associated with Bessel functions favoring larger Larmor radii. To further investigate this hypothesis, modeling activities with TRANSP-NUBEAM-TORIC and TORIC-SSPFQL will be presented.

1. M. Weiland et al., *Plasma Phys. Control. Fusion* **58** 025012 (2016)
2. M. Weiland et al., *Nucl. Fusion* **57** 116058 (2017)

Wednesday-19

ICRH modelling of the Baseline D-T scenario in JET

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In the 2021 and 2023 D-T campaigns in JET various scenarios with potential for application in fusion reactors have been studied. The mandate of the "Baseline" experiments was to explore the possibility to operate at high density, magnetic field and current. Although extremely promising results were obtained in D plasmas in the running-up to the actual D-T campaign and up to 8MW of fusion power was produced when adopting this scenario in D-T [1,2], it was – in contrast to the record T-rich scenario - not possible to sustain D-T shots for the envisaged 5 seconds. The underlying reasons are still being explored, offering perspectives for possible cures. The present paper concentrates on the detailed modelling of ICRH aspects and on the synergy between ICRH and NBI, allowing a better understanding of the key role of the auxiliary heating in these high-performance shots. One aspect setting the scenarios apart and which has several repercussions is that the density – purposely - is higher, which changes the beam penetration, modifies the collisionality, and necessitates the auxiliary power level to be higher to sustain fusion-relevant temperatures.

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Wednesday-20

The effect of high-power transient events on tungsten coatings used for radio frequency launcher applications*

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High-temperature plasma-facing material coatings used for radio frequency (RF) launchers need to be robust enough to survive RF breakdown arcing or other transient events from the plasma (e.g., an edge localized mode) without causing a catastrophic failure of the coating. High-power transient effects are being explored by using an RF-induced vacuum arc to determine the robustness of tungsten coatings made by a variety of manufacturing methods. A 1/4-wavelength resonant section of vacuum transmission line terminated with an open circuit electrode structure with a well-defined electric field (30-40 kV/m) produces repeatable arcing conditions. The transient time of the arc is on the order of microseconds, and the light emitted by the arc has been characterized by a filterscope. The initial focus is on tungsten as a plasma-facing material, including sintered tungsten, additively manufactured tungsten, tungsten coatings on steel produced via physical vapor deposition (PVD), and functionally graded tungsten/steel coatings deposited by low-pressure plasma-spraying (LPPS). Thin PVD coatings (1-2 microns) without an interface layer with the steel fail catastrophically from an arc and result in severe delamination of the coating. The arc-induced damage of thicker coatings, such as those made via LPPS, tend to be restricted to the top few microns of the surface. Arcing often initiates on sharp surface microstructures and causes localized melting of tungsten at the surface of all the materials (sintered, additively manufactured, PVD, etc.) and results in resolidified melt pools with surface cracks and pores. This modification of the surface is expected to influence the hydrogen retention and is currently being explored. Experimental details and hydrogen retention as a function of arcing conditions will be presented.

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Wednesday-21

Modeling RF Sheath Formation in Turbulent Tokamak Boundary Plasma

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During ICRF antenna operation, complex interactions between turbulent density profiles, nonlinear RF sheaths, and RF-induced convective transport are observed to alter the density of bulk and impurity species in the tokamak edge [1]. In this work, we explore the physics of such interactions via numerical modeling, using a nonlinear EM/plasma/sheath code (VSim, [2]) and profiles obtained from a fluid plasma turbulence code (Hermes, [3]) in a 3D slab domain containing biased side-wall limiters. RF sheath formation on antenna and limiter surfaces is observed as electromagnetic waves launched by the antenna are refracted through the turbulent density profile. On transport timescales, such sheath potentials have been shown to influence both the mean species density and its RMS fluctuation spectrum [4]. On the faster RF timescales, the converse is also true – regions of high plasma density near material surfaces give rise to the highest sheath potential amplitudes. We explore this correlation between density and sheath potential, and its consequences, for various antenna operation modes (strap phasing, input power) and plasma profiles (uniform/radial/turbulent). Effects of nonlinearity in the RF sheath are also considered.

When density is turbulent and spatially nonuniform, localized regions of high sheath potential (hotspots) may develop where high-density filaments intersect material surfaces. Such hotspots are of particular concern as sources of impurity sputtering, and we explore their behavior in response to changes both to the local plasma density and to the physical antenna parameters and structure. Related results exploring the role of Faraday shields and/or enclosing structures in suppressing high sheath potentials for other devices (e.g. SPARC) will also be shown.

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Progress of high power helicon experiment in KSTAR

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The fast wave in the lower hybrid wave frequency region, known as the helicon wave, has been revisited and proposed for efficient current drive systems at mid-radius in high electron beta plasma. A megawatt-level RF system with klystrons for the helicon current drive experiment has been installed at KSTAR with a large helical traveling wave antenna. However, progress has been slow due to damages to the coaxial vacuum feedthroughs and inconsistent antenna characteristics. During the 2023 KSTAR campaign, RF windows operated stably after replacing the coaxial VFTs, which were prone to arc damage due to multipactor, with twin-disk type windows. However, high-power applications to the antenna were unreliable due to inconsistent antenna characteristics, such as sudden and random increases in reflections that interrupted RF power for the system protection. The antenna assembly was removed from KSTAR and inspected before the 2024 campaign. It was found that the inconsistent antenna characteristics were due to imperfect electrical contact of the helical structure. After repairs, the antenna showed characteristics similar to the design and stable megawatt application and load-resilient characteristics in ELM conditions. This presentation summarizes the antenna conditioning for the reliable application of megawatt power and the preliminary helicon experiment results from the 2024 campaign.

Wednesday-23

|Commissioning and first experiments with the ICRH plant at Wendelstein 7-X

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For the operational phase OP2.1 of Wendelstein 7-X (W7-X) an ion cyclotron resonance heating (ICRH) plant was commissioned. This plant was designed, installed and is being operated within a collaboration between the Laboratory for Plasma Physics, Brussels, the Forschungszentrum Jülich and the Max-Planck Institute für Plasmaphysik.

The ICRH system consists of the two former TEXTOR ICRF generators that provide power to a two-strap antenna located on the low-field side near the bean-shaped plasma plane. The generators can operate between 25 and 38 MHz at powers of up 1.5 MW for 10 s every 3 min. The antenna has two independently powered straps, no Faraday-screen, no septum, internal prematching with remotely operable capacitors, is radially movable, fully water-cooled and equipped with an internal reflectometer and local gas inlets. A coaxial feed-forward transmission line matching sets the antenna strap phasing and relays the antenna to the generators. Thus the system can allow for some k||-shaping of the RF spectrum, fully cope with the heat load during up to 30-minute-plasma operation and optimize RF coupling for all magnetic field configurations of W7-X.

The operation of the ICRH system in OP2.1 to OP2.3 served to demonstrate plasma heating, fast ion production, plasma generation and ion cyclotron wall. The antenna was operated with RF powers up to 800 kW. Resonant heating scenarios was successfully demonstrated for hydrogen minority in helium, ^{3}helium in hydrogen, ^{3}helium in a 3-ion-scheme together with hydrogen and helium. Further experiments included ^{3}helium in helium and second harmonic hydrogen heating. An overview of these first results will be presented.

Wednesday-24

Force Free Vacuum Interface for DIII-D High Field Side Lower Hybrid Current Drive System

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High field side lower hybrid current drive (HFS LHCD) is promising method for efficient, off axis current drive.[1] The system is designed to drive off-axis current at r/a~0.6-0.8 with peak current density up to 0.4 MA/m2 with n||~2.7 at 4.6 GHz for target DIII-D discharges [2]. One of the primary mechanical challenges is to accommodate differential thermal expansion without a bellows component in-vessel. This requirement was driven by the lack of reliable RF bellows compatible with vacuum conditions. To have the RF bellows outside the vacuum, a force free interface was developed and simultaneously satisfied the radio frequency (RF), bake, and vacuum requirements. The force free interface must accommodate differential thermal expansion of the long (~2m) waveguides with respect to the DIII-D 0 degree R-1 port extension without damage to the waveguide or excessive force on the vacuum seals. Design of a custom bellows assembly with RF feedthrough that allows >15mm of waveguide thermal expansion is presented. Fabrication of this bellows assembly required simultaneously vacuum brazing multiple 304 Stainless Steel flanges to an oxygen free high conductivity copper waveguide section which required significant manufacturing development. Mechanical testing of and process details on the stainless steel to copper vacuum brazes using the BAg-8 (72 wt% Ag 28 wt% Cu) filler metal are shown. Further, the assembly and vacuum leak checking procedure for the outer wall waveguides and expandable vacuum interface at DIII-D are presented. The latest analysis, tests, and installation status will be presented.

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The construction and commissioning of the Electron Bernstein Wave Heating and Current-Drive System for MAST-U

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The UK's Spherical Tokamak for Energy Production (STEP) program was established to design and build a prototype powerplant with the aim of achieving net energy production [1]. Heating and Current drive (HCD) is a key driver for a fusion power plant design and it has been concluded that the optimum HCD system for STEP is microwave-based, using a combination of the Electron Cyclotron and Electron Bernstein Wave (EBW) approaches [2]. A development program has been initiated to integrate a 1.8MW EBW system on the MAST-U tokamak [3] and provide an experimental test of the EBW Current Drive technique in a Spherical Tokamak [4]: to examine open issues regarding the LFS coupling scheme (O-X-B mode conversion), examine effects of density fluctuations, collisional damping and non-linear effects, verify current drive capabilities and extend the original MAST experiments on EBW-based solenoid free start-up [5].

The MAST-U EBW system [3] includes dual frequency Gyrotrons (34.8 GHz 1.5-1.6 MW delivered to plasma for HCD, 28 GHz 650-750 kW delivered to plasma for non-inductive start-up or HCD) [6]. The microwave power travels to the MAST-U vessel through evacuated hybrid (HE11, 88.9 mm dia.) corrugated waveguides and mitre bends. The microwave power can also be directed into a long pulse dummy load located in the sources area for local Gyrotron conditioning. Inside the MAST-U vacuum vessel, a steerable in-vessel launching system provides flexible options via separate launchers: off-axis, on-axis and high field side start-up [10]. Stray microwave power detectors (sniffer probes) and protection components (interceptor plates) in the path of the first reflection will also be installed in the MAST-U vessel.

The 2025 objective is to commission the gyrotrons in a short pulse dummy load (few tens of ms), then in the long pulse test load (few seconds pulses). This requires commissioning of the High Voltage Power Supplies, Gyrotrons and all their ancillary systems. In-vessel components (launcher, steering mechanism and in-vessel protection components) will be installed in MAST-U during the 2026-27 engineering break, devoted to the installation of the EBW system and additional neutral beam heating capabilities. In parallel, we will finalise the C&I design for launch control (steering mechanism, polarisers), the machine protection system based on sniffer probes and IR cameras, and the integration of the EBW system in the MAST-U machine control (MC) environment. EBW HCD will be tested after the 2026-27 MAST-U engineering break.

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Effect of Density Irregularities on Radio Frequency Wave Propagation in Ionospheric Plasmas

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Density irregularities, such as filaments in the scrape-off layer (SOL) or instabilities in the core plasma, have significant effects on radio frequency (RF) wave propagation for fusion plasmas. Similar plasma irregularities are found in space plasma and can greatly influence plasma wave behavior. This review focuses on the impact of density irregularities on RF wave propagation in the ionosphere. The ionosphere is a thin layer that spans altitudes from 100 km to 2,000 km and contains dense electron concentrations ranging up to 10^{12} m^{-3}. Ionospheric density irregularities occur over a wide range of latitudes from the equator to the polar region and local times. Since ground-to-ground communication using high-frequency (HF, 3–30 MHz) waves relies on wave reflection in the ionosphere, and satellite-to-ground communications involve waves passing through the ionosphere, the effects of these density irregularities are critical for both types of communication. Many efforts have been made to study wave propagation in the ionosphere; however, several challenges arise in modeling the propagation of HF waves. For instance, the wavelength of radio waves in the ionosphere is much shorter than the ray path and the thickness of the ionosphere. Therefore, simulating these waves requires high-performance computing resources. This challenge is one reason why previous research has predominantly focused on ray tracing and analytical calculations. As a result, the properties of HF waves—such as reflection, refraction, scattering, and scintillation—are not fully understood in the context of density irregularities. In this presentation, we utilize the Petra-M code to conduct numerical examinations of how ionospheric density irregularities affect HF wave propagation. For simulations, we utilize density irregularity structures, such as equator plasma bubbles, derived from the fluid simulations as background ionospheric density structures, and we analyze characteristics of HF waves transmitted from the ground station. It is shown that HF waves can scatter in different directions when they encounter density irregularities in the ionosphere. As a result, receivers may capture weaker signals or fail to receive signals altogether.

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Dimensional Analysis of Pyrolytic Graphite Grids for 4CM2500KG Tetrode by Automated Laser Inspection

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Ion Cyclotron Range of Frequency (ICRF) plasma heating systems are planned for many next generation fusion tokamaks. To enable ICRF systems to operate on high field tokamaks, such as ITER and SPARC, reliable megawatt-level high-frequency (>100 MHz) sources must be developed. Tetrodes are a proven technology for reliable high-power amplification for ICRF systems; long pulse operation has been shown for the CPI 4CM2500KG tetrode at a frequency of 131 MHz at 1.74 MW for up to 5.4s [1]. Additionally, short pulse operation of the 4CM2500KG at a frequency of 120 MHz at 1.7 MW has been shown [2]. Due to lack of demand for tube-based transmitter systems from industry, the 4CM2500KG is no longer in production by CPI and thus, re-establishment of the manufacturing supply chain for critical components of these tubes is paramount. The most critical of these components are the pyrolytic graphite control and screen grids. Over the past 5 years, the chemical vapor deposition, machining, and cutting processes required to produce these grids have been redeveloped with suppliers in the US and the EU. Due of the large chemical vapor deposition process parameter space, the required precision of the geometric features (<+/-.01 mm), and lack of industry expertise, the manufacturing of these components is extremely difficult. As a result, two separate supply chains were developed to maximize success probability. Both supply chains were shown to be viable for producing screen and control grids, but due to their different approaches, there are significant differences in the final components. Through the use of an automated laser-based inspection device developed by Bold Laser Automation, the geometric differences between these components, specifically in the critical grid pattern areas, are reported and recommendations for the continued production and refinement of these components are detailed.

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Wednesday-28

The present status of the TORIC-SSFPQL codes

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The TORIC and SSFPQL [1] codes have been developed into a user-friendly package that allows detailed simulations of auxiliary heating scenarios in the ion cyclotron frequency range, with moderate CPU memory and time requirements, compatible with installing the codes on a personal laptop computer. The iteration to achieve consistency between the solver of the wave equations in toroidal geometry (TORIC), which evaluates the power deposition profiles needed to estimate the amplitude of the quasilinear diffusion operator on each magnetic surface, and the solver of the kinetic equation (SSFPQL), which provides the quasilinear distribution functions used to calculate the coefficients of the wave equations, can be started automatically by specifying some parameters in appropriately organized namelists. The effects of toroidal trapping on the QL distribution functions [2], and the possible presence of Neutral Beam Injection [3] and/or thermonuclear alpha particles in the deceleration [4] can be taken into account. A set of auxiliary routines is available for the preliminary study of the scenario (location of resonances and cutoffs in the poloidal cross section, local dispersion relation, etc.), including an adapted version of the FELICE code [5] for the detailed study of the IC antenna.

Two examples of such simulations are presented. In the core of a typical minority heating experiment in a medium-sized tokamak such as ASDEX Upgrade, the hf power density can reach several Watt/cm^3. Under these conditions, the heating rates predicted by TORIC and SSFPQL for a Maxwellian plasma differ substantially, and 5 or more iterations may be required to reach agreement between the two codes. The physics behind a particular heating scenario is well illustrated by the difference between the results for the Maxwellian plasma and those for the self-consistent QL distribution functions. Interesting conclusions can be drawn about the effects of toroidal trapping and the poloidally varying ambipolar potential that must arise in the toroidal geometry to maintain charge neutrality in the anisotropy of the ion distribution functions.

In reactor-grade plasmas such as ITER, the available RF power per ion will be at least an order of magnitude less than in current experiments, and probably even less. As a consequence, one or two iterations are usually sufficient. A series of simulations with increasing background temperatures are presented to illustrate the effects on IC ion heating of the increasing competition from absorption by the electrons and, in certain scenarios, from the increasing population of thermonuclear alphas in the slowing down.

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Comparative Analysis of I-mode and L-mode LHCD Capabilities on EAST

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Lower Hybrid Current Drive (LHCD) serves as a crucial non-inductive current drive method for steady-state tokamak operations [1]. While L-mode has been extensively studied for LHCD applications, the recently discovered I-mode regime offers promising confinement characteristics with reduced edge instabilities [2]. Up to now, comprehensive comparisons of LHCD efficiency between these regimes remain limited. LHCD characteristics across I-mode and L-mode operations were investigated in this paper, focusing on the effects of boundary parameters and pedestal temperature profiles on current drive capability.

Statistical analysis reveals enhanced LHCD capabilities during I-mode operation compared to L-mode. At low densities, while Parametric Decay Instability (PDI) [3] remains comparable between the two modes, I-mode exhibits an elevated temperature pedestal, resulting in more pronounced LHCD broadening and improved drive efficiency. In high-density scenarios, I-mode operations demonstrate superior LHCD performance due to lower edge density and higher temperature profiles, which effectively suppress PDI effects. These findings highlight the significant influence of boundary parameters and pedestal temperature characteristics on LHCD efficiency, with I-mode consistently showing advantageous current drive capabilities across operational regimes, and providing a promising path toward reliable non-inductive current drive systems essential for maintaining steady-state operation in future fusion power plants.

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Wednesday-30

Hybrid Plasma Source (HPS) application in the design of a physical plasma antenna

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Combining an Electron Cyclotron Resonance (ECR) and a Helicon plasma sources, an innovative Hybrid Plasma Source was constructed and patented. The motivation of this invention was to create an ultra-high electron density at low electron temperature physical plasma source for more efficient semiconductors material processing [1].

Physical plasma sources with such characteristics find applications in the flexible design of the next generation plasma antenna technology. In this research, an engineered design of a modern plasma antenna will be presented with a computational analysis of its operational parameters. In addition, a discussion will be presented on an issue that limited the operation of the original HPS, i.e., neutral depletion and repletion via ionization, and how it can be mitigated. Further, other physical plasma sources such as plasma displays were shown as standalone switchable antennae with characteristics that distinguish them from other types of metal antennae [2].

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Parametric study of helicon wave current drive in CFETR

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We evaluate the feasibility of helicon current drive (HCD) in a hybrid scenario for the China Fusion Engineering Test Reactor (CFETR). Utilizing the GENRAY/CQL3D package, a large number of simulations (over 5 000) were conducted, with parametric scans in the antenna's poloidal position, launched parallel refractive index ($n_{||}$), and wave frequency. The analysis reveals that helicon has excellent accessibility under reactor-level conditions, and smaller $n_{||}$ and higher wave frequency result in enhanced wave absorption. The simulations demonstrate an optimal launched $n_{||}$ of approximately 1.6 for the CFETR hybrid scenario, with helicon achieving a maximum drive efficiency of 2.8×10^{19} $A\cdot W^{-1}\cdot m^{-2}$. The best launch position is found to be within a poloidal angle range of 25 degrees to 65 degrees. Additionally, it is preferable to have a narrow $n_{||}$ spectrum for wave absorption when operating below the threshold value of $\Delta n_{||}$ (~0.6), beyond which the effect of $\Delta n_{||}$ on wave absorption is negligible. This study provides valuable insights into the potential application of HCD in CFETR.

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