Signal Tests of High-Power Broadband Combining Stages for the DTT ICRH Transmitter

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The Divertor Tokamak Test (DTT) facility requires a broadband (60–90 MHz) Ion Cyclotron Resonance Heating (ICRH) system capable of delivering 3 MW to the plasma. This system utilizes for its transmitters a novel 5-stage corporate combining network to aggregate 128 Solid-State Power Amplifier (SSPA) modules into a single 1.2 MW output.
This paper presents the circuit modeling and signal test validation of the high-power stages (Stages 3, 4, and 5). These stages utilize 2-stage coaxial impedance transformers designed to handle megawatt-level power with high efficiency. Stage 3 and 4 serve as intermediate 4-input combiners, while Stage 5 provides the final 1.2 MW output.
Initially, combining stages were studied separately using analytical MATLAB models and Ansys HFSS simulations; then they were cascaded to simulate their global behavior. This work compares these numerical predictions with experimental measurements provided by the industrial supplier. The results demonstrate excellent agreement across the full bandwidth, validating the electromagnetic performance of the hardware. This confirmation is a vital milestone for the full-chain system integration and upcoming multi-physics thermal analysis.

#12720

Design and Modeling of the AMAROK High-Power Neutral Beam Positive Ion Source development platform

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

The Advanced Multi-coil Antenna for Radio-Frequency (RF) Operations at Kilowatts (AMAROK) is a 200 kW, 2–4 MHz inductively coupled plasma (ICP) ion source development platform designed to advance neutral beam injection (NBI) technology. Compatible with DIII-D NBI accelerator, made to demonstrate a path towards replacing the common long pulse source, AMAROK provides a flexible, well-diagnosed environment for systematic studies of antenna
topology, Faraday shield (FS) design, and RF coupling under NBI-relevant operating conditions. The system employs four solid-state 50kW RF generators that can be operated independently for a single-turn antenna configuration or combined for the Multi-turn antenna operation.

Plasma load projections for AMAROK will be informed by measurements from SupRISE, a pre-prototypical ICP source of similar volume and geometry that operates with a multi-turn antenna in the 4–8 MHz range at up to 50 kW. Analysis of RF voltage, current, and power data will be used to derive scaling laws that will be used to project the RF loading envelope for AMAROK at higher power, providing a quantitative basis for anticipating matching requirements for plasma operation in AMAROK beyond the validated SupRISE regime.

At high power, the FS must simultaneously maximize RF coupling by minimizing eddy current losses, protect the dielectric window from direct plasma exposure, and avoid excessive thermal loading. To address this challenge, a suite of modular, replaceable FS designs will be developed and assessed using integrated mechanical, electromagnetic and thermal modeling. SolidWorks and COMSOL Multiphysics will be used to evaluate shielding effectiveness, coupling trends, thermal loading, and mechanical stress, with actively cooled concepts expected to reduce post-pulse cooldown times by approximately 30%.

Together, high-power multi-generator operation, experimentally informed load projections and physics-informed FS design establish AMAROK as a scalable and versatile testbed for high-power ICP antenna and FS development in support of next-generation NBI systems.

This work is supported by the U.S DOE under the awards DE-SC0024523 and DE-FC02-04ER54698.

#12760

Driving Frequency and Chamber Length Studies at High Power in the SupRISE ICP Test Device for Neutral Beam Injection at DIII-D

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

As part of the ongoing effort to develop inductively coupled plasma (ICP) ion sources for the DIII-D Neutral Beam Injection (NBI) system, experimental studies have been performed on the SupRISE (Superior Radiofrequency Ion Source Experiment) test device to investigate how RF drive frequency, chamber geometry, and other operating conditions impact plasma production and uniformity at high power. Using a modern solid-state RF generator, hydrogen plasmas have been routinely sustained in SupRISE while coupling up to 50 kW through the quartz vessel over a frequency range of 4–8 MHz. Parametric sweeps were conducted while characterizing plasma properties, including plasma density and electron temperature, using an adjustable Langmuir probe and optical emission spectroscopy.

Frequency studies at high power are presented showing how the maximum central plasma density scales with drive frequency under otherwise identical operating conditions. These results explore a relatively unstudied regime of large-volume ICP operation at elevated frequencies and powers relevant to neutral beam ion source requirements. In parallel, chamber length studies have been performed to determine how ICP length influences both achievable plasma density and RF power coupling efficiency while maintaining a centrally located 5 turn racetrack antenna.

To better understand thermal loading on plasma-facing components, experimentally measured temperature data of the quartz vessel and copper Faraday shield were obtained using a transmission-calibrated infrared camera over various duty cycles with and without the Faraday shield installed. These measurements are used to validate finite element analysis (FEA) thermal models previously developed for the device.

Plasma uniformity measurements have also been performed at the extraction plane using a recently developed probe array diagnostic. These measurements show how the ion density distribution varies as a function of frequency, pressure, and chamber length, directly addressing the uniformity requirements necessary to minimize beam divergence in neutral beam sources.

Results from these studies will be used to guide design choices and future parametric investigations on the full-power ICP ion source development platform, AMAROK, at North Carolina State University. This research is conducted in close collaboration with North Carolina State University to jointly advance ICP ion source development for neutral beam injection at the DIII-D National Fusion

Facility.

This work is supported by the U.S DOE under awards DE-SC0024523 and DE-FC02-04ER54698

Toward a Next-Generation RF Ion Source for the DIII-D Neutral Beam Injection System

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

Neutral Beam Injection (NBI) is a critical tool for non-inductive heating, current drive, fueling, and diagnostics in magnetic confinement fusion devices. The DIII-D tokamak operates eight NBI ion sources based on the U.S. Common Long Pulse Source (CLPS) design, providing up to 20 MW of injected neutral beam power. While incremental upgrades over several decades have resulted in a highly flexible system—including on- and off-axis injection, co- and counter-current capability, and programmable energy and power waveforms—the long-term reliability of the NBI system has increasingly been challenged by aging filament-driven ion sources and associated power supplies.

To address these limitations, the DIII-D Neutral Beam group is pursuing the development of a next-generation RF-driven inductively coupled plasma (ICP) ion source compatible with existing DIII-D beamline infrastructure. Building on earlier conceptual and modeling work, significant progress has been achieved since 2024 with the construction and commissioning of the Superior RF Ion Source Experiment (SupRISE), a dedicated test platform designed to validate RF ion source performance at relevant plasma parameters and operational duty cycles.

SupRISE incorporates a flexible RF antenna and plasma chamber geometry and is supported by a comprehensive suite of diagnostics developed specifically for ion source validation. These include optical emission spectroscopy, single and multi-tip Langmuir probes, ion flux probes, and infrared thermography to characterize plasma uniformity, density, electron temperature, ion flux, and thermal loading of plasma-facing components. Together, these diagnostics enable detailed benchmarking of experimental results against predictive modeling.

The development program continues in close collaboration with North Carolina State University and has expanded to include capstone design projects and student research contributions from local universities, strengthening workforce development in RF ion source engineering. Modeling efforts combine hybrid kinetic-fluid plasma simulations with finite-element analysis using COMSOL Multiphysics and SolidWorks to assess electrical, thermal, and mechanical performance, inform antenna and material selection, and guide operational limits.

Current efforts are focused on optimizing RF coupling efficiency, plasma stability, and component robustness, as well as planning the integration of an RF-driven ion source into existing DIII-D NBI infrastructure. This includes interface definition, controls integration, and compatibility with present power, cooling, and vacuum systems.

This paper presents an overview of the SupRISE RF ion source development program, including experimental results, diagnostic validation, modeling comparisons, and a roadmap toward
beamline implementation, representing a key step in the long-term modernization of the DIII-D NBI system.

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BREWSTER-ANGLE WINDOW: DESIGN VERIFICATION WITH LARGEST-TO-DATE DIAMOND DISK AND EXPERIMENTAL LOSS TANGENT DISTRIBUTION

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The chemical vapour deposition (CVD) diamond Brewster-angle window represents a compact broadband window solution for high power, continuous wave (CW), frequency steering gyrotron operations in electron cyclotron heating (ECH) systems, required by nuclear fusion devices. With a waveguide aperture of 63.5 mm, compatible with a millimetre (mm)-wave power of 2 MW-CW, this window needs a very large diamond disk with minimum diameter of 180 mm. A thickness target of about 2 mm would be required for an appropriate mechanical stability. In 2020, a first-of-its-kind disk was manufactured with 180 mm diameter and thickness of about 1230 μm at centre and in the range 1100-1150 μm at the edge. However, the large disk development was then put on hold.
This paper shows thermal and structural analyses aiming to check the Brewster-angle window performance with the large 180 mm disk at the minimum achieved thickness of 1.1 mm. As a novelty, in contrast to the constant input value used in the past, the analyses were carried out by the experimental measurement of the loss tangent distribution over the disk as input. First, comparative analyses assessed the impact of such input change. Analyses were run for both torus and gyrotron side of the ECH system considering HE11 and Gaussian modes, respectively. The window design was then verified with the achieved disk thickness for normal operation (NO), considering also the off-normal event of 1 bar overpressure on the torus side. An overpressure of 2 bar acting on the disk would lead to stresses on the tensile side of the disk greater than 200 MPa, beyond the allowable limit. The use of the measured loss tangent distribution over the disk as input to the window thermal analysis is recommended to obtain a better picture of its performance and detect any difference with reference to the transverse mode of the mm-wave beam.

A brand new power supply system for the TCV diagnostic neutral beam injector

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

A Diagnostic Neutral Beam Injector (DNBI) is operated on the TCV Tokamak since 2000. Whereas the plasma source and the ion optical system have benefited from several upgrades, the power supply (PS) system has remained unchanged since its last revision in 2006. To allow for a long term reliable operation, a compete replacement of the PS system has started two years ago. This includes a high-speed current source rated 500A/100V/2s for the arc-discharge plasma source (called ARC), a High Voltage Modulator (HVM) rated 55kV/5A/2s for extraction and acceleration of the ion beam, along with a new control system.

ARC is basically a DC/DC converter, built upon standard components from Imperix. The output current is split over 8 power blocs in parallel, each independently controlled. HVM consists of a stack of 48 power modules (PM) jointly developed by SPC and Imperix. The PM output is finely controlled providing high dynamics and low voltage ripple. HVM is thus able to supply the first two grids of the ion optical system at different current and voltage values. An additional PM is used to supply the third grid (suppression grid). The other PSs, namely for arc ignition, magnetic insulation, bending magnet, and gas injection, are built with commercial components.

This paper addresses the development of the new PS system, starting from the requirements and the design of each subsystem. The ARC and HVM designs are discussed in details, with emphasis on the modeling of the HVM multi-secondary transformer. The mechanical installation of the PS system is described, as well as the integration into the TCV safety system. Then, the commissioning tests are presented and discussed, while the final validation of the whole PS system on TCV is scheduled for the second half of this year. The detailed description of the control system is presented in a separated paper.

This work was supported in part by the Swiss National Science Foundation.

Design and Simulation of a Regulated High-Voltage Power Supply for ICRF Tetrode Transmitters Using DTT as Study Case

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

High-power tetrodes represent a robust and reliable technology for transmitters in the Ion Cyclotron Range of Frequencies (ICRF), which is adopted in both old and new plasma heating systems and is the subject of research and development programmes. Tetrode cavity amplifiers have been shown to be capable of delivering continuous-wave, MW-level, ICRF power to weakly unmatched loads. Furthermore, it has been demonstrated that the operational capabilities of these amplifiers in mismatching conditions can be enhanced through the use of an agile regulation of their anode voltage. High-Voltage Power Supplies (HVPSs) are in charge of providing such voltage, whose typical values are in the range of a few tens of kV, with currents that can exceed 100 A. The implementation of a closed-loop control system for the anode voltage as well as the prediction of its performance through an accurate and comprehensive model of the power supply is of crucial importance to ensure the required stability and to assess the dynamic response of the system.
The present work sets forth the design and analysis of a regulated HVPS composed of multiple Switching Power Supply (SPS) modules fed by multi-secondary transformers that, since a few decades, have superseded the series tetrode regulator. Each SPS module is a fully-featured AC-DC converter. The modular architecture allows the overall output voltage to be synthesized with stringent requirements on regulation range, ripple, and slew rate. Special attention is devoted to the design and modeling of multi-secondary transformers. This includes loss scaling, equivalent circuit parameter estimation, and harmonic mitigation through phase-shifted star–delta secondary configurations.
A detailed MATLAB/Simulink model was developed and used to simulate steady-state and transient operating conditions, validating analytical sizing and demonstrating compliance with all system requirements. Finally, assuming that the ICRF transmitter features two tetrode-based amplification stages, a dual-output HVPS configuration is proposed to enable the simultaneous and independent supply of both amplification stages. The results confirm the feasibility and robustness of the proposed architecture that, relying to the largest extent possible on off-the-shelf devices, stands out for modularity and versatility.

Efficient coupling of radiofrequency to negative ion source plasmas

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

The negative ion sources (NIS) are a fundamental part of neutral beam injector (NBI) for current drive, and require the formation of a very low temperature plasma, with relatively high density ne. For durability reasons, these plasmas are preferably energized with radiofrequency (RF), with an external coil antenna separated by a dielectric window from plasma. Source is enginereed so to have regions of plasma at different electron temperature Te typically the driver region (near to coil, Te1>4 eV), an expansion region, and a region near extraction (Te2

#12861

Experimental Validation of Hybrid Fluid-Kinetic Plasma Model for RF Inductively Coupled Plasma Positive Ion Sources

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

In this work, we report the first experimental validation of a hybrid fluid-kinetic plasma model for radio-frequency (RF) inductively coupled plasma (ICP) neutral beam injector (NBI) ion sources. RF ICP ion source development for NBIs has traditionally focused on maximizing power delivered to the plasma via
reduction of RF power losses and optimization of RF power coupling efficiency. Less focused upon is source improvement through optimization of plasma parameters for ion production and spatial uniformity. Hybrid fluid-kinetic plasma modeling provides an advantageous approach to plasma parameter optimization never before applied to high-power ion source design. By treating electrons kinetically and ions and neutrals as fluids, hybrid models can accurately capture electron kinetics, such as non-Maxwellian electron energy
distribution functions, characteristic of low-pressure, low-temperature plasmas used in positive and negative NBI ion sources. Accurate electron kinetics, absent in traditional fluid models, strongly impact predicted ionization rates, dissociation fractions, plasma chemistry, and ultimately ion flux, species composition, and uniformity. To study positive ion source plasma parameters and validate our hybrid fluid-kinetic plasma modeling approach, experimental investigations on LUPIN — a reduced-scale prototype RF ICP positive ion source delivering up to 11 kW RF power at 2 MHz to pure helium, hydrogen, and deuterium plasmas — have been conducted in conjunction with computational modeling of LUPIN using the Hybrid Plasma Equipment Model (HPEM). We present experimental measurements of plasma properties in pure hydrogen plasmas at pressures and delivered powers ranging from 10-100 mTorr and 500-2500 W, respectively. Studied parameters include electron density, measured via microwave resonator hairpin probe, and gas temperature, via optical emission spectroscopy of the Fulcher-α bands of molecular hydrogen. Electron densities were observed to exceed 10^17 m^-3 at the plasma edge and demonstrated clear, linear dependency with delivered power over the entire 500-2500 W measurement range. Gas temperatures ranging from 500-1000 K were recorded, along with significant shifts in the relative intensities of rovibrational states indicative of a deviation from expected two-temperature distributions of rovibrational state populations. Comparisons of HPEM simulations of LUPIN with experimental measurements demonstrated quantitative and qualitative agreement between experiment and model over a wide range of delivered power and pressure. Experimental validation is a critical step toward applying hybrid modeling to the design of RF ICP positive ion sources for the DIII-D NBI system, with potential applications to modeling negative ion sources where electron kinetic effects are even more pronounced and fluid assumptions break down further.

Recent Progress in the Conceptual and Electromagnetic Design of a Vacuum Feedthrough Developed for the DTT ICRH System

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The Ion Cyclotron Resonance Heating (ICRH) system of the Divertor Tokamak Test (DTT) facility requires vacuum feedthroughs (VFTs) capable of handling high radio-frequency (RF) power while ensuring reliability, vacuum compatibility, and adequate margins against electrical breakdown. VFTs for high-power ICRH applications are critical electromagnetic and structural elements with stringent electromagnetic, mechanical, thermal, and vacuum requirements to avoid RF breakdown and surface flashover within the structure.

This work presents recent progress in the conceptual and electromagnetic design of a VFT developed for the DTT ICRH system.

Several configurations exploring different profiles and transitions between the ceramic and the metallic parts of the VFTs have been investigated. Special attention has been devoted to the reduction of electric field enhancement at the edges of the ceramic and at the brazed joints, which are known to be particularly susceptible to breakdown and long-term degradation. Smooth geometrical transitions and optimized profiles have been shown to significantly reduce peak electric fields, thereby increasing the safety margin against surface flashover without compromising mechanical robustness.

The shape and dimensions of the rigid coaxial waveguide and of the inner conductor have been addressed and several shapes have been analysed with particular attention to reducing the component of the electric field parallel to the ceramic surfaces. Tapered sections and locally enlarged inner and outer conductor radii have been introduced to mitigate sharp field gradients and reduce peak values of the electric field for a given transmitted power. The impact of these modifications on the characteristic impedance and on RF matching has also been assessed to ensure compatibility with the overall ICRH transmission line.

The design-oriented approach adopted in this work aims at identifying an optimal compromise that satisfies electromagnetic breakdown constraints while remaining compatible with mechanical and layout limitations of the DTT ICRH system and preserving reasonable component size and manufacturability. The combined optimization of ceramic window geometry, inner conductor shape, and waveguide dimensions provides increased margins against RF breakdown and represents a robust and promising solution for high-power VFTs for ICRH applications in DTT.

Thermo-Hydraulic and thermal-structural analysis of Actively cooled Ion Cyclotron Travelling Wave Array for WEST

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

This paper presents the different thermo-hydraulic analysis that were performed during the design of the two antennas of a new Ion Cyclotron Range of Frequency (ICRF) Travelling Wave Array (TWA) system for the WEST facility. The TWA system aims at addressing the limitations of the current ICRF antenna, focusing on improved coupling, reduced RF sheaths voltages and enhanced reliability.

The TWA Antenna is made of 316L stainless-steel in order to be compatible with future fusion plant environment. This brought challenging thermal design requirements as this antenna is a plasma facing components with plasma radiation that can reach 0.2 MW/m^2 and RF losses up to 1 MW/m^2 on the straps edges. The design was performed in accordance with RCC-MRx, the French code for the design and construction of mechanical components for nuclear installations, particularly research reactors and fast neutron systems, in order to closely reflect future design of fusion power plant.

The antenna design is first briefly presented: Due to verry limited space inside WEST Vacuum Vessel, RF lines section had to be changed from classical cylindrical coaxial shape to a more flat rectangular design, which led to challenging dimensioning and manufacturing issues.

The main hydraulic design parameters and thermal loads are then introduced: the two launchers, the two antenna protections and outer conductors will be actively cooled with WEST B30 cooling loop (70°C, 33 bars), while the inner conductors will be cooled by the STEFFI cooling loop (25°C, 10 bars). HF losses were modelled on ANSYS HFSS and plasma radiation were modelled thanks to ray tracing simulation.

The main analysis and the results are then discussed: first a strong cantilever effect was observed on the straps due to bad thermal conduction of 316L, which could impact HF performances. Also, keeping a good HF contact between the straps and the launchers' structure was challenging. The thermo-hydraulic and thermal-structural analysis of the flat-line and RF lines is then presented, with specific attention to vacuum feedthrough bellows that required optimization regarding the number of waves. Finally, global hydraulic

analysis of the whole antenna demonstrated a good balancing of flows within the two launchers. The output design of these various analysis does not impact HF performance while respecting RCC-MRx design criteria.

#12946

Upgrades and Experimental Results of the DIII-D Helicon RF System

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

Helicon plasma waves provide a promising technique for auxiliary heating and non-inductive current drive in magnetic confinement fusion experiments. At DIII-D, a 476 MHz radio-frequency system has been installed to launch helicon waves with up to 1.2 MW of injected power, enabling detailed evaluation of coupling, heating efficiency, and driven current. This report describes recent commissioning campaigns, focusing on hardware performance, corrective maintenance, and the resulting advances in plasma operation.

Across several fiscal years, the helicon system underwent a sequence of improvements to enhance reliability and extend operational flexibility. In FY22, upgraded vacuum feedthroughs were installed to address prior insulation weaknesses and improve power handling margins. During FY23 experimental operations, inspections revealed damage to in-vessel components near the antenna, including thermocouples and RF probes, likely caused by localized heating and arcing. These components were removed or replaced to prevent further degradation. More serious limitations emerged on the 150° vacuum transmission line, where a failed bellows allowed the inner conductor to shift off axis and disabled the port. Elevated temperatures were also detected in the 210° elbow, indicating excess RF losses. In addition, a waveguide switching assembly intended to alternate launch configurations did not operate reliably.

Major repairs and upgrades were executed in FY24. The damaged inner conductor and bellows on the 150° line were replaced, and a boron nitride centering ring was installed to maintain alignment. A pressurized patch panel restored the capability for bidirectional power injection. To mitigate overheating observed previously, copper plating and diamond-like carbon coatings were applied to both elbows during FY24–25, lowering resistive losses and reducing multipactor susceptibility. Although these actions improved stability, extended RF conditioning remains necessary to prepare the in-vessel stripline for high-power operation. A redesigned waveguide section now enables switching between co- and counter-current launch. Plans for FY26 include replacing the arced boron nitride ring with a quartz component to further strengthen reliability.

Commissioning experiments in FY24–25 confirmed successful high-power operation. Central electron heating was observed in both L-mode and H-mode plasmas, producing on-axis temperature increases approaching 0.8 keV. Up to 700 kW was coupled for 2 s from the 150° feed, and motional Stark effect measurements indicated changes in the q-profile consistent with helicon-driven current. Overall, the recent upgrades have restored full operational capability of the DIII-D helicon system and demonstrate its potential as a tool for plasma heating and current drive.

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Development of NBI-Based ?N Feedback Control System with NBI-2 Power Modulation in KSTAR

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

As part of ongoing advancements in plasma performance control in KSTAR, the Neutral Beam Injection (NBI) control system has been continuously upgraded since 2021 to enable real-time actuator capability for advanced scenarios. In the 2024 campaign, the NBI feedback control system was successfully commissioned within the Plasma Control System (PCS), and real-time feedback experiments on normalized beta (βN), a key performance parameter, were performed. The feedback algorithm is based on an integral control scheme and achieves real-time regulation through continuous adjustment of beam power and energy. To minimize plasma perturbations and improve control robustness, additional functionalities such as source phase delay were incorporated into the control logic. The system was successfully applied to high-βN discharges, where feedback control at a target βN of 2.9 mitigated performance degradation caused by minor MHD activity. Furthermore, the application of a slope βN target effectively suppressed βN overshoot during the H-mode transition and reduced MHD activity that could otherwise lead to disruption, demonstrating the feasibility of NBI-based βN regulation in high-performance plasmas. Building on this foundation, the 2025 KSTAR campaign introduced NBI-2 arc power supply on-off modulation, enabling significantly faster and more flexible beam power control compared to previous continuous schemes. Dedicated experiments verified stable modulation operation and reliable integration with the PCS, achieving improved temporal response and enhanced actuator authority for plasma pressure control. Utilizing this high-speed NBI-2 modulation capability, active βN control experiments were conducted by dynamically adjusting beam power in response to plasma pressure evolution. Preliminary results indicate that fast NBI-2 modulation enables more effective regulation of βN evolution while maintaining discharge stability, reducing βN excursions and associated MHD activity. These results demonstrate the progressive development of NBI-based βN feedback control in KSTAR and establish a foundation for fully integrated, high-speed performance control in future high-βN operational scenarios.

Anode power supply development for triode type gyrotrons at TCV

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The Swiss Plasma Centre (SPC) has been involved for many years in the development and operation of high-power gyrotrons for applications in the field of fusion. At present, the contribution is focusing on two major activities. The first is the Electron Cyclotron Resonance Heating (ECRH) system of TCV, formed mainly of dual-frequency triode type gyrotron, rated at 1 MW for 2 s. The second is the FALCON test facility for ITER/DTT gyrotrons and associated RF components, including transmission lines and launchers.
Over the past 30 years, the SPC has operated several generations of anode power supplies (APS) for triode-type gyrotrons, referred to the cathode potential. A new APS (60 kV/50 mA), has been developed to supply the first dual-frequency gyrotron delivered at SPC (2018). The design is based primarily on off-the-shelves components, which have the advantages of being robust and cost effective. This home-made solution has been improved (2024) to leave the possibility to modulate (ON/OFF) the DC output voltage up to 100 Hz.
Recently, SPC launched a study for a new design integrating stringent requirements to comply with the ECRH operation on TCV. Namely a power supply (60 kV/50 mA), referred to the cathode potential (-85 kV), capable of ON/OFF square modulation up to 5 kHz, a partial square modulation on a DC voltage range of 10 kV, and controlling the output voltage for variation up to 1 kV/ms. The present paper describes in detail the design choices, taking into account the budget made available through special funding from the Swiss government for fusion technologies.

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Adaption of the ITER H&CD Workflow to MUSCLE3 for multi-scale integrated modelling

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

The heating and current drive (H&CD) workflow [1] has been developed within the ITER integrated modelling and analysis suite (IMAS) [2] as a modular component for predicting the performance of H&CD sources in Tokamaks, including all main H&CD sources: electron cyclotron resonance heating, ion cyclotron resonance heating, neutral beam injection , lower Hybrid current drive, and nuclear fusion reactions. While originally implemented using the iWrap orchestration capabilities [3], the H&CD workflow has now been adapted to the MUSCLE3 persistent actor framework [4] to enable advanced multiscale coupling and component-level parallel execution, while preserving the strong modularity that has been central to its design.

The migration to MUSCLE3 required a refactoring of the orchestration layer and the encapsulation of existing physics solvers as independent actors with standardized interfaces. The new architecture makes use of MUSCLE3's manager for coordinating multiple components in time and for running different models together in a single workflow. This allows H&CD modules to interact more easily with other physics workflows like ITER Pulse Design Simulator (PDS) [5] that operate on different time and spatial scales.

The modular architecture allows for flexible selection and iteration of physics actors, enabling studies of synergetic effects between multiple heating sources, as demonstrated in previous work for scenarios of the ITER old baseline [1]. Beyond preserving existing functionalities, MUSCLE3 provides a pathway to improved scalability through parallel execution and support for multiscale coupling in the H&CD modelling chain. It facilitates concurrent execution of coupled solvers, more flexible workflow reconfiguration, and supports hierarchical, runtime-configurable architectures. These features are essential for future integrated modelling studies in which H&CD calculations will be dynamically coupled to other physics workflows (transport solvers, MHD stability analyses, and synthetic diagnostics) operating at different scales, to build a comprehensive multi-scale plasma simulation environment for ITER and other devices.

This adaptation of the H&CD workflow is directly aligned with the development of the ITER PDS, as it enables its integration into the PDS and supports the long-term goal of predictive, whole-device modelling for scenario development and control studies. This work is also aligned with the development of the High-Fidelity Plasma Simulator based on the DINA-JINTRAC coupling, which now supports MUSCLE3 physics components and workflows.

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[2] F. Imbeaux et al, Nucl. Fusion 55 123006 (2015)
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[5] M. Schneider et al., this conference.

KIT’s integrated approach to progressive cooling of cavities in fusion MW-class gyrotrons

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Topics

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Communication type

Oral

Abstract

European ECRH gyrotrons currently use Raschig Rings (RR) for cavity cooling, but simulations indicate that mini-channels (MC) offer superior thermal performance, especially for the 170 GHz, 2 MW coaxial gyrotron design. To validate and optimize MC cooling, KIT is pursuing a comprehensive program combining experiments with automated multi-physics simulations. Significant progress has been made in both mechanical design and instrumentation of a dedicated cooling test facility, aiming to replicate realistic operating conditions of MW-class continuous-wave (CW) gyrotrons. An induction heating device produces cavity heat fluxes of the same order of magnitude and about 50% of nominal CW heat load levels. To improve experimental fidelity, the setup was redesigned for vertical orientation, enhancing coolant flow symmetry and alignment with real installations. In parallel, the control and data acquisition system is being upgraded, with an expected tenfold increase in sampling rate. This will improve the temporal resolution of thermal measurements and enable detailed analysis of transient effects, essential for validating thermal-hydraulic numerical models. Previous experiments revealed fluctuations in cooling-water flow rate, introducing uncertainty in calorimetric power balance and heat-flux estimates. To address this, the flow control system is being upgraded, including an electrically actuated valve for active flow stabilization during measurements.
Tested MC designs are being considered for implementation in a 170 GHz, 2 MW coaxial-tube cavity capable of pulses up to 1 s. A next-generation KIT mock-up featuring fully circular mini-channels has been designed to achieve stable gyrotron operation in multi-MW output RF power range. Key parameters under investigation include channel geometry, inner diameter and radial distance to the cavity wall. Furthermore, multi-physics simulations support interpretation of experimental results.
Current efforts focus on manufacturing the optimized cavity mock-up and commissioning the upgraded test stand. Experimental verification will provide critical data for validating and further developing advanced cooling concepts for high-power CW fusion gyrotrons.

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ITER TORUS DIAMOND DISCS: DIELECTRIC AND OPTICAL CHARACTERIZATION

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The electron cyclotron heating (ECH) systems in nuclear fusion devices require dielectric low-loss transmission windows that serve as confinement and/or vacuum barriers both at the microwave sources (gyrotron side) and at the plasma vessel (torus side). Window materials with outstanding properties, such as extremely high thermal conductivity and very low dielectric losses are needed, especially for high-power continuous wave operation. Chemical vapour deposition (CVD) polycrystalline diamond, manufactured in shape of discs in microwave plasma assisted (MPA) reactors, represents the only viable solution. The use of CVD diamond in nuclear fusion facilities, particularly as a tritium barrier in the torus window assemblies, requires a high level of quality-assurance measures. In the case of ITER torus windows, as part of a contract with Fusion for Energy (F4E), 62 diamond discs with diameters of 70 mm and thicknesses of 1.11 mm were tested for their dielectric and optical properties. This work shows the adopted procedures for the measurements, and the results are discussed. The dielectric losses were determined over the disc's surface by spatially resolved measurements in Fabry-Perot hemispherical resonator, and in the central region of the disc by spherical resonator. Dark microfeatures were detected across the entire surface of the discs by optical microscopy. The main conclusion is that all diamond discs exhibited consistently low dielectric losses. Although individual discs differ in quality, they were all qualified, meaning that they fulfilled the requirements for the statistical values D50 and D90 related to the loss tangent distribution measured in each disc.

Solid-State High-Power ICRH transmitter for the Divertor Tokamak Test facility: from design specifications to factory acceptance tests.

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

Vacuum-tube technology has long been the reference solution for high-power radio-frequency (RF) transmitters in ion cyclotron resonance heating (ICRH) systems, with major fusion devices relying on vacuum-tube amplification chains. Solid state technology, widely adopted in other high-power RF applications such as accelerators facilities and broadcasting, has been mainly limited to driver stages in ICRH systems. Recent progress in solid-state devices suggests that a fully solid-state approach can now be considered for the entire ICRH amplification chain.

Within this framework, two fully solid-state 1.2 MW transmitters have been commissioned for the Divertor Tokamak Test (DTT). Operating in the 60-90 MHz range, they are based on LDMOS technology and use multi-stage power combining for wideband performance. Each transmitter can deliver up to 1.2 MW at VSWR ≤ 1.5 in pulsed mode (1.4% duty cycle) for plasma experiments (50s per hour); up to 120 kW at VSWR ≤ 3 in CW mode for wall conditioning (1h every 2 hours); and low-power pulsed operation (10 ms-5s, duty cycle ≤ 1.5%) with a maximum RF output power ≤ 100 kW at VSWR ≤ 2 for antenna conditioning. The transmitter shall achieve an efficiency ≥ 0.52 under matched load conditions at nominal RF output power, with an overall gain ≈ 91 dB.

Factory acceptance tests (FAT) on the first RF module-one of 128 units forming the modular architecture of one transmitter-were completed in May 2025, demonstrating output power of approximately 11 kW at 60 MHz and 90 MHz with VSWR around 1.5 and efficiencies approaching 70%. Additional FAT activities at cabinet level were carried out in September 2025, reaching up to 2*44 kW under matched and mismatched conditions, including VSWR up to 3. The final stages of the power-combination chain were also tested by the end of 2025.

This contribution outlines the transmitter development from specification definition and design choices to factory acceptance testing of the first transmitter, planned for the second half of 2026. The fully solid-state approach adopted for DTT represents a first-of-its-kind application in ICRH and it will serve as a testbed to asses the feasibility and performance of solid-state technology in high-power ICRH systems, providing feedback for future developments.

#13035

Fluid models as a tool to investigate possible plasma non-uniformity compensation strategies in giant negative ion sources

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

ITER's Heating Neutral Beams (HNBs) are expected to play a key role in achieving the required operating conditions in the tokamak and in demonstrating the industrial feasibility of fusion energy production.
The HNB prototype, MITICA (Megavolt ITer Injector \& Concept Advancement), is currently being assembled at the Neutral Beam Test Facility in Padova and will soon start operations.
At full performance, it will deliver \SI{16.5}{\mega\watt} of heating power with a beam energy of \SI{1}{\mega\electronvolt}, obtained by neutralizing a current density of \SI{290}{\ampere\per\square\meter} of \ce{D-}.
The plasma source consists of four pairs of radio-frequency drivers facing a common expansion chamber, where the required negative ions are produced.
To maximize ion production, a transversal magnetic (filter) field is generated inside the expansion region to confine hot electrons away from the extraction region and thus reduce the \ce{D-} destruction rate.\par
However, the interaction between this magnetic field and the axial electric field induces a vertical plasma drift, leading to a significant vertical non-uniformity.
Mitigating this non-uniformity is essential to ensure uniform beam extraction and acceleration: the ITER requirement sets a target uniformity level of 90\%.
Several strategies have been proposed to address this issue, the most immediate being the adjustment of the input power and gas flux to the individual driver pairs, thereby introducing a controlled plasma density asymmetry to compensate for the drift-induced non-uniformity.
In this work, we present a numerical investigation of the compensation mechanisms based on a 2D fluid plasma model of the MITICA source, as a preparatory step toward the experimental campaigns of the coming years.
The simulations are performed with the FSFS2D (Fluid Solver For SPIDER 2D) code, developed at Consorzio RFX and extensively validated against experimental data, demonstrating its capability to reproduce, at least qualitatively, the trends of the main plasma parameters.
The code solves for the continuity and temperature equations for all the relevant plasma species (electrons, four species of hydrogen ions and two species of neutrals), as well as the Poisson equation for the electric potential, in a self-consistent way.
Hydrogen chemistry is included, both in the plasma bulk and at the surfaces.
The results reported here will guide the upcoming experimental campaigns, providing a preliminary assessment of the proposed compensation strategies and their expected impact on plasma parameters and beam uniformity.

Design and Structural Analysis of the Ion Cyclotron Travelling Wave Array for WEST

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

A new Ion Cyclotron Range of Frequency (ICRF) Travelling Wave Array (TWA) system was developed for the WEST facility. The TWA system aims to address the limitations of current ICRF launchers, focusing on improved coupling, reduced voltages and RF sheaths and enhanced reliability. This paper focuses on all the aspects of the structural analysis of the antenna. Electro-magnetic (EM) loads under disruption, static loads, dynamic analysis including WEST structure response. As well as the consequences on the antenna structure including external chassis with its radial positioning system.

The antenna design and its main parameters are briefly presented. The development complied with RCC-MRx, the French code for mechanical components in nuclear installations, particularly research reactors and fast neutron systems, to align with future fusion power plant requirements.

In WEST, EM load calculations are inferred from a 20° toroidal wide ANSYS model. Due to its large toroidal size (38°) and its toroidal cantilever structure inside the vessel, a larger 60° wide toroidal model had to be created to get accurate EM loads results.

As WEST vacuum vessel has a double shell structure, contrary to most tokamaks, it cannot withstand high mechanical loads from internal components, especially during disruptions. The support structure and the connecting elements of the TWA launchers was then designed accordingly, with special attention to dynamic EM loads.

The chassis have to serve several purposes. The first is to support the weight of external components and part of the antenna itself. The second is to allow the smooth and precise radial displacement of the antenna. Its design had to comply with the structural loads while fitting 4 RF lines (where current WEST ICRH antenna only has 2), 2 water loads or parts of the resonant ring (depending on the RF layout).

The analysis and results are discussed. First, care was taken to the load transfer onto WEST vessel through the “pavés rotule”, since these supports are critical for WEST integrity. Then the connections design and bolting. The main one between the two emitters and the beam is the most critical, because of the emitter cantilever, the strong EM loads and the tight space. Furthermore, for the manufacturing purpose, the emitters and the beam are bolted. A focus is made on the bolted junctions that include a RF heel in the emitters.

Design and Optimization of a Single-Channel cooling geometry for the ECRH front-steering mirror of DTT

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The full power configuration of the Divertor Tokamak Test (DTT) facility will include 32 Electron Cyclotron Resonance Heating (ECRH) front-steering launching mirrors, each one reflecting a 1 MW microwave beam. Despite the mirror surface high reflectivity, a small fraction of the incident microwave power is still absorbed, generating significant and localized elliptical Gaussian heat flux with a peak of 2.1 MW/m^2. At the same time, the in-vessel operating conditions also impose stringent electromagnetic and mechanical constraints, limiting the choice of mirror substrate materials to low-conductivity metals, such as stainless steel and Inconel alloys, or technical ceramics. Under these restrictions, the operating environment demands a water-cooling system capable of ensuring reliable heat removal, where a single uninterrupted channel is preferable from a safety and monitoring standpoint, simplifying fault detection. The present work aims to identify an optimal single-channel cooling geometry capable of ensuring acceptable operating temperatures in materials with low thermal conductivity, with a primary focus on Inconel 718. Following a review of the available optimization approaches capable of accommodating the single-channel topological constraint, the most suitable strategy was identified in a parametric shape optimization. This approach is guided by a simplified analytical cost function whose objective is to match the local average velocity along the channel with the surface heat flux distribution. Within this framework, a dedicated parametric geometry was defined to describe a spiral channel trajectory as well as its local width and depth distributions, together with a variable trapezoidal cross-section adopted to balance centrifugal forces and limit the generation of hotspots. This new configuration provides the required flexibility to explore design variations while preserving geometric continuity and respecting manufacturing constraints. Once optimized, the resulting geometry was evaluated through CFD simulations, showing significant cooling performance improvement with respect to the state of the art.

Stability of Beam Properties during Long Pulses at ITER NBI, for Hydrogen and Deuterium operation

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The international fusion experiment ITER will be equipped with powerful neutral beam injection (NBI) beamlines for heating and current drive (HNBI). The requirements for the HNBIs in terms of current extracted form the ion source are very challenging. Namely, 286 A/m2 for deuterium and 329 A/m2 hydrogen [1], for an extraction area of ≈0.2m2. The negative ions are extracted and accelerated using a multi-grid multi-aperture system [2]. To achieve optimal beamline transmission (above 78%), the extracted beam must satisfy two critical conditions: high uniformity (better than 90%) on the scale of individual beamlets, and a core divergence < 7 mrad [3]. The extracted current and beam properties must be stable over the entire beam duration (300 s - 500 s for the DT-1 phase [4]).

The BUG (1/8 scale) and ELISE (1/2 size ITER source) test facilities [5] are capable of long pulse operation. The presence of vertical plasma asymmetry, together with caesium dynamics during long pulses with pulsed extraction, have a potential impact on the beam properties. This study presents a comprehensive analysis of the extracted beam properties, both their absolute values and their temporal development, during long pulses for hydrogen and deuterium operation. The beam properties are compared to those obtained during short pulses (10 s). These results show the effects of different RF power levels, source pressures, surface biasing configurations, cesium conditioning statuses, and extraction conditions (such as extracted current density, extraction and acceleration potentials) on the extracted beam properties during long pulses.

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Prototype and testing of the actively-cooled flexure-pivot concept for the DEMO ECH steering mirror

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The Neoclassical Tearing Mode (NTM) control in the EU DEMO tokamak will be achieved by targeting millimeter-wave power to the q = 3/2 and q = 2 magnetic surfaces. To accomplish this, two steering antennas, each composed of a fixed mirror and a steering mirror and targeting two 2-MW beams, will be installed at the upper and lower positions of each one of the six Electron Cyclotron Heating (ECH) launchers planned for installation in the DEMO equatorial ports. The steering mirror is based on a mechanical assembly of rigid links connected through flexure pivots with specific structural compliance. These flexure pivots act as mechanical joints that convert the quasi-linear motion produced by a set of pneumatic actuators into a pure rotational movement of the steering mirror.
Recent design developments propose the integration of cooling channels directly within the flexure pivots. This solution enables direct dissipation of the high thermal loads deposited in these components while eliminating the need for external piping between elements. Several Inconel-718 flexure pivots were prototyped using additive manufacturing to demonstrate the feasibility of this concept. Tensile tests were conducted to characterize the mechanical properties of the samples, and thermal experiments were performed at the FALCON facility at EPFL, Switzerland, to evaluate their cooling performance under representative stray-radiation conditions.
This work has been carried out within the framework of the EUROfusion Consortium, partially funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No 101052200 — EUROfusion). The Swiss contribution to this work has been funded in part by the Swiss State Secretariat for Education, Research and Innovation (SERI). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union, the European Commission or SERI. Neither the European Union nor the European Commission nor SERI can be held responsible for them.

Fast Switchable Dual Harmonic Gyrotron Operation for Versatile Plasma Heating and Stabilization

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

In the frame of a EUROfusion Enabling Research Project, KIT and NKUA have been working on new theoretical and technical concepts aiming at the ability to quickly switch a high-power gyrotron between the usual first harmonic frequency operation and the second harmonic one. The new type of dual-harmonic operation allows the possibility to double - or to halve - the frequency rapidly with no changes or only minor adaptions to the applied magnetic field, which up to now is the bottleneck in the frequency switching speed. Therefore, this approach opens a new degree of flexibility in the power plant design by combining multiple functionalities, and can be utilized to save overall costs as well as gain space in the EC building. In particular, a gyrotron can be quickly switched to create different absorption profiles in the plasma. For example, switching between aiming at the plasma core (at the second harmonic frequency) and the plasma edge (at the fundamental frequency) would reduce or avoid dedicated power installation for the radiative instability mitigation, which would otherwise be mostly in standby.

In a first step, the basic principles and switching concepts have been elaborated, including the internal quasi-optical launcher system, which has to be capable to handle both harmonics and therefore significantly different resonator frequencies and modes. Then a DEMO-relevant dual-frequency / dual-harmonic coaxial cavity gyrotron has been designed, including the mentioned quasi-optical output system and a multi-stage depressed collector. The latter is necessary for electron energy recovery and thus serves to increase the overall efficiency of the gyrotron operated in the second harmonic, making it competitive with its counterparts operated in the fundamental harmonic. In the last part, MW-class second harmonic operation at 170 GHz of a TE34,19 coaxial cavity gyrotron will be experimentally demonstrated for the first time (planned for March 2026).

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Observation and Parametric Analysis of Lower Hybrid Fast Wave (LHFW) Coupling in KSTAR Plasmas

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The Lower Hybrid Fast Wave (LHFW) is an attractive candidate for heating and current drive in high-density fusion plasmas due to its deeper propagation and absorption characteristics compared to conventional slow waves. Building upon previous research that encompassed the design, fabrication, and successful installation of a travelling wave antenna (TWA) at the KSTAR P-port in August 2024, this study focuses on the first experimental results. We report the first successful observation of LHFW coupling in KSTAR plasmas during initial low-power experiments. Experimental data demonstrated that the reflection coefficient remained below 10% across a wide range of plasma densities. The coupling characteristics is investigated using data from specific experimental shots to identify the influence of three critical factors: electron density, gap distance at the mid-plane, and plasma elongation (kappa). Furthermore, plasma coupling simulations using COMSOL are performed to analyze the coupling fraction between the fast wave and slow wave branches. The results verify that the antenna's design effectively suppresses parasitic slow wave coupling, ensuring that the majority of the power is coupled into the fast wave branch.

Long-Pulse Operation Performance of the KSTAR NBI-2 System

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The KSTAR Neutral Beam Injection (NBI) system is a major auxiliary heating device developed to support high-performance plasma operation and long-pulse discharges in the KSTAR tokamak. The system consists of two beamlines, each equipped with three 100 keV ion sources. The Second Neutral Beam Injector (NBI-2) system, installed in 2018, was designed with enhanced engineering features to improve reliability, enable long-pulse capability, and provide real-time beam power controllability while maintaining output specifications comparable to the First Neutral Beam Injector (NBI-1)[1].

For long-pulse operation, significant engineering upgrades were implemented. Cooling water flow rates of the ion sources and beamline components were increased to improve thermal stability under extended operation. A cryogenic condensation pumping system was adopted to maintain a high-vacuum environment with a base pressure below 1.0×10^{-7} mbar[2], ensuring stable beam extraction and minimizing re-ionization losses.

The high-voltage DC power supply for beam extraction and acceleration is based on a modular architecture composed of 200 series-connected 600 Vdc power modules. A pulse-width modulation (PWM) control scheme generates a stable 100 kV output. To enhance operational reliability, 16 modules are configured as standby units operating at 0 V. In the event of a module failure, the control system replaces the faulty module within 10 ms, maintaining constant output voltage without interrupting beam operation[3]. This fault-tolerant design prevents system trips and enables continuous long-pulse operation.

Operational results demonstrate the engineering performance of the upgraded system, achieving a maximum beam power of 5.73 MW for 1 s and long-pulse operations of 1.17 MW for 102 s and 3.1 MW for 48.4 s. This paper presents the key engineering solutions adopted for long-pulse voltage control, describes commissioning and operational experiences, and discusses technical challenges and mitigation strategies relevant to reliable long-pulse neutral beam systems for future fusion devices.

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[2] K.P.Kim et al., Commissioning of Vacuum pumping devices of the second KSTAR neutral beam injection system. Fusion Eng. Design, 146(2019) 1786-1789
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Recent breakthroughs in long pulse operation of negative hydrogen ion sources for ITER NBI

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

The neutral beam injections systems for ITER are based on RF-driven negative hydrogen ion sources (H^-, D^-) and are designed to operate for up to one hour. This poses several major challenges compared to today's NBI systems, which routinely operate with positive ion sources on time scales of 10 seconds. The test facilities ELISE and BATMAN Upgrade are essential test facilities within the European step-ladder approach to address those issues and contribute to the success of ITER's NBI systems [1].
While more than 90% of the required current density for 600 s was successfully demonstrated in hydrogen with an RF generator power per driver that is only 75% of that at ITER's sources [2], stable long-pulse operation with deuterium is much more challenging. This is because the co-extracted electron current in deuterium is higher than in hydrogen and increases steadily over time until the power load on the extraction grid reaches it limits.
In principle, plasma operation is very stable over time, but the evaporation of caesium into the ion source adds a dynamic component to the plasma and extracted current densities. Caesium is needed to generate sufficient negative ions through the conversion of positive ions and atoms at the plasma-grid of the extractor.
In recent years, several control knobs have been identified, including the field strength and topology of the magnetic filter, the application of a bias voltage to additional surfaces to symmetrize and reduce the electron current flux to the grid, temperature variations of the surfaces to influence the cesium dynamics in the ion source, over-conditioning the source with cesium before a long pulse, etc. With the gained insights that the caesium flux to the converter surface is the limiting parameter during plasma phases, an additional caesium evaporation concept close to the converter surface was developed [3]. With this caesium shower, stable extracted negative ion currents representing over 90% of current density at the available RF generator power were achieved in deuterium for 1000 s. The co-extracted electrons are well below the critical limit and stabilized as well. This breakthrough was only recently achieved at BATMAN Upgrade and gives confidence that the ITER NBI target for deuterium will be achieved.

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[2] D. Wünderlich et al., Nucl. Fusion 65 (2025) 014001.
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Stray radiation minimization via real-time polarization control for DTT's scenarios

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The Divertor Tokamak Test (DTT) will be equipped with an Electron Cyclotron Resonance Heating (ECRH) system comprising a total of 32 gyrotrons (170 GHz, 1 MW, 100 s). The generated microwaves will be transported to the plasma launchers through an evacuated Quasi-Optical transmission line (TL). Each individual beam along the TL is equipped with a pair of polarizing mirrors. These grooved mirrors can be rotated about their normal axis to change the beam polarization, so to maximize the injected power absorption and prevent stray radiation from damaging in-vessel components. However, the relationship between mirror rotation angles and correct microwave polarization (for optimal plasma-wave coupling) is highly nonlinear. Furthermore, the uncertainties in the launching directions and in the plasma edge, to which the wave must be coupled, are difficult to model comprehensively. Consequently, the optimal coupling (necessary to reduce at the minimum the possibly harmful stray radiation) requires data-driven approaches. This work regards the design of real-time polarization controller, capable of regulating the microwave polarization to ensure minimal stray radiation levels. The control algorithms consist of a two-level structure, decoupling estimation from actuation. The first block, the estimator, computes the gradient of the stray radiation with respect to each polarizer rotation angle, exploiting perturbations on the polarizers' angular positions and using stray radiation measurements. The second block, the controller, is designed to compute the new references for the polarizers' angles given the gradient of the stray. Since stray radiation measurements are not always available in real-time, an alternative approach, using plasma electron temperature from the Electron Cyclotron Emission (ECE) emission diagnostic, has been proposed. Both approaches have been calibrated on TCV's data, and their validation is scheduled to take place on the TCV during the 2026 experimental campaign. The results of the aforementioned assessment are presented in this work.

#13145

Impact of caesium transport on the performance of the ITER Heating Neutral Beam Negative Ion Source SPIDER

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

SPIDER is the full-scale prototype of the ITER Heating Neutral Beam (HNB) negative ion source. To meet the required current densities, negative ions production (H^-/D^-) is significantly enhanced by coating the converter surfaces with caesium (Cs), which lowers their work function. In particular, achieving an adequate and uniform Cs coverage of the Plasma Grid (PG), the first electrode of the multi-grid accelerator facing the plasma, is essential for stable and efficient long pulse operation. SPIDER is equipped with three Cs ovens located at the rear of the source.
During the most recent 2024-2025 experimental campaign, the use of Cs enabled the extracted negative-ion current density to reach values up to 210 A/m2 with extracted electron-to-ion ratios of the order of unity. These results were obtained in a configuration where only one quarter of the ion source was operated, leading to non-standard caesiation conditions and modified redistribution dynamics.
Cs behaviour is monitored by an extensive diagnostic set, including Surface Ionisation Detectors (SID) to measure the flux of Cs out of the ovens, Laser Absorption Spectroscopy (LAS) measuring the Cs density in front of the PG, and Optical Emission Spectroscopy (OES) interpreted with dedicated collisional radiative models.
Cs transport in SPIDER was thoroughly studied in the vacuum phase, between plasma pulses, with numerical tools dedicated to gas transport in vacuum, in particular AVOCADO. This approach enabled optimization of the Cs injection geometry in SPIDER.
However, it is widely recognized that the steady-state Cs distribution is determined by the redistribution that occurs during plasma discharges, and that this contribution becomes more significant with respect to the evaporation in vacuum as the pulse duration increases. To provide a more comprehensive description of Cs dynamics, a Cs transport code was developed, that consistently treats both vacuum and plasma phases and includes surface processes affecting the redistribution.
This contribution presents an updated understanding of Cs transport in SPIDER, combining the analysis of experimental measurements with the results from the Cs transport and redistribution model, providing a deeper insight into the Cs dynamics observed during the most recent SPIDER campaigns. We demonstrate the strong sensitivity of Cs conditioning to the geometry of the injection point, identify the influence of the magnetic filter field direction on Cs redistribution, and assess the impact of variations in surface sticking probability. These results provide a physics-based framework for optimizing Cs management in upcoming full-power SPIDER operations with all beam segments active.

A conceptual design of ECCD launcher for JA DEMO

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The electron cyclotron current drive (ECCD) system design activities for JA-DEMO is on-going. The physical analysis of ECCD in JA-DEMO presented RF injection conditions and launcher design had started to realize high-efficient current drive operation aiming full non-inductive plasma current drive operation of tokamak.
In this study, we will present a conceptual design of the equatorial port launcher for engineering analysis modeling. To avoid the harmonic absorption region of ECCD in JA-DEMO plasma, an angle of more than 45 degrees in the toroidal direction is required for equatorial port injection and a fixed front mirror is utilized to produce the injection angle in the toroidal direction. For control of injection angle, a remote steering system with square waveguide is planned to utilize to adapt severe radiation condition in tokamak. 25 square waveguides with sides of 50 mm for remote steering, arranged in a 5x5 configuration, and to inject about 25 MW from a single port. The movable mirror mechanism for remote steering is installed behind of the port and the cryostat, which improve maintenance reliability of mechanics.
The quasi optical RF beam transmission is calculated to present the RF beam profile at tokamak inside for candidate operation frequency. The scalar combination of beam intensity profile represented the power deposition area is oval flat top shape with 0.75m x 0.4m and RF power intensity of 50 – 90 MW/m2 at flat top.
The structure of the launcher consists of waveguides surrounded by the shielding structure which is stainless steel with water piping for radiation shielding and cooling. The engineering model is developed for evaluation of the spatial radiation dose rate behind the port where the steering mirror system locates. In addition, maintenance planning considering access to the equipment in the device based on CAD modeling work will be reported.

Development of a 238 GHz Megawatt-Class Gyrotron within the BMFTR Project HeatQIS

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

The HeatQIS project addresses a key technological bottleneck for future high-field fusion power plants: the development of high-performance millimeter (mm)-wave microwave sources capable of meeting the demanding frequency and power requirements for plasma heating and control. Advanced quasi-isodynamic stellarator concepts such as Proxima Stellaris require operating magnetic fields exceeding 8 T. This magnetic field strengths directly determines the electron cyclotron resonance frequency, leading to a required heating frequency of approximately 240 GHz. Gyrotrons are high-power vacuum electronic devices that generate coherent mm-wave radiation and represent the only technologically mature sources capable of delivering continuous-wave mm-wave power in the megawatt range at sub-THz frequencies.

The overall objective of the project is the development of enabling technologies for a next-generation electron cyclotron resonance heating (ECRH) system. This includes not only the gyrotron tube itself and the investigation of a possible high-temperature superconducting magnet system for the gyrotron, but also associated key components such as mechanically robust diamond windows and launcher systems for mm-wave coupling into the plasma, as well as looking into the influence of stray magnetic fields. In combination, these components form the framework of the ECRH system for future high-field fusion power plants with magnetic plasma confinement.

The gyrotron work package of the project aims to demonstrate a device targeting 1.7 MW output power at 238 GHz. Scaling the gyrotron from current fusion-relevant frequencies (140–170 GHz) to 238 GHz while maintaining high efficiency and stable operation presents significant challenges in cavity design for the beam–wave interaction, electron beam shaping and thermal management. One promising approach is to use a coaxial-cavity gyrotron. The coaxial insert reduces voltage depression by space-charge and enables stable operation in very high-order modes by suppressing competing modes, thereby improving mode selectivity. This, in turn, allows for larger cavity radii and reduces wall loading, relaxing thermal stresses on the outer cavity wall.

Beyond the initial short-pulse prototype demonstration, the project aims to develop a continuous-wave gyrotron that is suitable for industrial-scale production, taking into account factors such as voltage concept and fabrication tolerances. This gyrotron, along with the knowledge and technical solutions generated within HeatQIS, will directly support the planning and engineering of the ECRH system for the first stellarator demonstration power plant, Proxima Alpha.

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Thermal and Structural Verification of the Polarizer Mirror Design for DTT ECRH System

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The Divertor Tokamak Test (DTT) facility aims to address the power exhaust challenge through the investigation of advanced divertor configurations and innovative solutions, supported by one of the largest Electron Cyclotron Resonance Heating (ECRH) systems currently under development. The ECRH architecture consists of four clusters, each including eight gyrotrons, a transmission line (TL), and eight independent launchers. The TL is organized into Single-Beam Transmission Line (SBTL) sections near the sources and launchers, and a central Multi-Beam Transmission Line (MBTL) employing large focusing and planar Multi-Beam Mirrors (MBMs) to simultaneously handle eight beams. At the end of each SBTL, a pair of polarisers is foreseen, with the crucial role of controlling the microwave polarisation according to plasma conditions to achieve maximum coupling of the launched beam with the desired plasma wave (ordinary or extraordinary polarised). Due to their corrugated geometry, these mirrors pose specific challenges from both electromagnetic and thermo-mechanical perspectives, particularly under high thermal loads. The present study aims to perform a Multiphysics analysis of the corrugated polarizer under high thermal loads, with specific emphasis placed on the spatially non-uniform heat distribution along the individual corrugations. In order to properly predict the overall thermo-mechanical response of the system, a detailed representation of the localized electromagnetic heat loads has to be taken into account. Electromagnetic simulations were coupled with dedicated mathematical modeling to derive localized heat load profiles, which were subsequently implemented in thermo-structural analyses. A direct comparison with the conventional assumption of uniformly distributed thermal loads was performed under identical boundary conditions. The mechanical structure has been optimized to reduce thermally induced corrugation structure deformation, as well as to maintain structural stiffness. A single-channel spiral cooling configuration has been considered to ensure thermal control, as it has been validated for transmission line mirrors to operate efficiently at high temperature levels. In addition, mass reduction has been considered to improve system dynamics to enable real-time mechanical control. The results show that there are significant differences between uniform and non-uniform thermal load distribution, indicating that non-uniform thermal load distribution must be considered to achieve precise prediction of thermomechanical response.

Optical design of EC antennas for a fusion power plant: method and concept

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

In view of future fusion power plants, Electron Cyclotron (EC) systems play a key role for several tasks, such as plasma start-up, sustainment and control during the discharge. Consequently, large amount of installed EC power is foreseen in near-future devices, such as DTT (32 MW) or ITER (72 MW), and above 100 MW for fusion pilot plants. EC Launchers, for the injection of the power into the plasma, must satisfy several physics and technological requirements. These include the capability to reach different plasma regions – requiring appropriate steering mechanism – and space limitations, which implies high power densities per launcher. Pending specific requirements for a fusion pilot plant, the design of EC launchers has to retain a degree of flexibility to cope with different potential physics needs in terms of wave frequency, total power and steering range for plasma heating and instability control. A trade-off between narrow apertures, for reducing neutron radiation escaping outward from the plasma, and thermal loads due to microwave stray radiation deposition in port ducts and optics housing boxes is also required. Building on the experience gained with the European DEMO reactor studies and its Low-Aspect-Ratio variant, the option of front-steering in-vessel multi-beam mirrors, combined with waveguide sections for matching with the transmission line, represents a reasonable starting point for a fusion pilot plant. This work describes the procedure adopted for the optical design of an EC launcher based on the requirements given by physics. Modelling of beam propagation into the plasma with the GRAY code provides indications on optimal launching angles and needed ranges to reach the required plasma regions with suitable driven current and deposition size. The needed beam parameters are obtained with the launcher optical design, defining mirrors position, size and shape. At last, electromagnetic simulations in GRASP provides an indication of the minimal duct aperture required for the passage of the EC beams and thermal loads on the mirrors and on port plug walls due to direct stray radiation.

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Integrated optiSLang workflow for the design and analysis of microwave launcher optical mirror systems

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

Future fusion devices, such as tokamaks and stellarators, will rely on non-inductive heating systems to maintain the high temperature plasma required for their operation. This is partly due to the inverse relationship between temperature and resistivity meaning that the heating effect from induced current is reduced as plasma temperature increases, requiring auxiliary heating and current drive methods to heat the plasma instead. Microwave frequency heating via electron cyclotron resonance heating (ECRH) or electron Bernstein wave (EBW) is an auxiliary heating technology that can both provide plasma heating and current drive.

A series of mirrors are required to shape and direct the input microwave beam towards the desired location; these mirrors and their surrounding structure are the microwave launcher. The design and analysis of these mirrors is a time-consuming, multi-discipline process, potentially requiring several iterations to reach an acceptable design.

This work has developed an optiSLang workflow to more efficiently design, analyse and optimise a microwave launcher system. The workflow first takes a specification file that sets key source term parameters and component locations. Next, it generates idealised geometry with different branches of the workflow corresponding to flat or focussing mirrors. The geometry is then put through a raytracing simulation using Zemax OpticStudio, resulting in the spatial irradiation data for each of the mirrors in the system and analysis of the resulting beam that exits the launcher. The spatial irradiation profiles are then used as inputs for an ANSYS Mechanical FEA simulation whereby the mirror thermal deformation is calculated, allowing the deformed system to be constructed and subject to raytracing. This enables analysis of the beam characteristics at the plasma interface, for both undeformed and deformed models, in order to compare key performance characteristics such as power delivered to the plasma, beam deflection and resulting beam dimensions. Finally, a robustness analysis has been performed to investigate which angular and positional tolerances have the largest effect on the resulting beam that is directed towards the plasma. This is with a view to understand tolerancing sensitivity which could then lead to optimisation with respect to cost.

#13361

Modernizing the FALCON Body Power Supply (BPS)

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The FALCON test facility, established by Switzerland and Fusion for Energy (F4E) at the Swiss Plasma Center (SPC) of EPFL in Lausanne, is a high-power experimental facility designed for testing Electron Cyclotron (EC) mm-wave components at high power. The Body Power Supply (BPS), one of the two main FALCON High Voltage power supplies, is a modular power supply capable of delivering up to 40 kV at 150 mA. This power supply, provides the voltage between the gyrotron's mm-wave generation section and the grounded depressed collector.

This paper details the upgrade of the BPS control system, replacing the original implementation with a modern solution using the Imperix(4) Hardware Platform. The upgrade retains all existing power hardware, including the transformer, power modules, mechanical structures, and output filter, ensuring minimal operational downtime and low cost while extending the system's operational life.

The new control system integrates a Beckhoff PLC to maintain backward compatibility with the existing Modbus-based Gyrotron control system interface, as well as power supply auxiliaries control, monitoring, and safety. The Imperix B-Box handles the power supply control, introducing Open Platform Communications Unified Architecture (OPC UA) protocol support for modern supervisory connectivity. Compatibility with existing power modules is preserved using Imperix optic interface boards, and the power supply protections interface directly with the B-Box RTD via a new interconnection board.

This in-house development enhances reliability and provides greater control over system design and performance, future-proofing the BPS for advanced monitoring and diagnostics. The approach demonstrates how legacy power supplies in high-power experimental facilities can achieve improved performance, reliability, scalability, and connectivity through targeted control system upgrades while preserving all existing power hardware.

(4) www.imperix.com

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Neither the European Union nor the European Commission nor SERI can be held responsible for them.

ECRH and ICRH studies for the Novatron magnetic mirror fusion device

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The Novatron magnetic mirror [1] [2] is a novel concept for a fusion reactor designed to reach fusion-relevant plasma parameters and competitive levelized cost of energy (LCOE). Its performance relies on long particle and energy confinement times along magnetic field lines—critical in open systems. Confinement is enhanced through the combined effects of favorable magnetic topology, ambipolar potential formation, and ponderomo- tive forces. Ambipolar plugging in the tandem configuration is achieved by establishing different plasma potentials in the central and tandem cells, which can be controlled by adjusting density or, primarily, electron temperature.

This paper presents a theoretical study of plasma heating in such a tandem config- uration, examining the applicability of Electron Cyclotron Resonance Heating (ECRH) and Ion Cyclotron Resonance Heating (ICRH) in both the central and tandem cells. The analysis includes system parameters, antenna design and placement, required power and frequencies, all determined by the magnetic configuration and outcomes from theory and ray tracing simulations. With the first experimental device, N1, already operating at the KTH Royal Institute of Technology, these studies directly support understanding system behavior and guiding the design of next-generation Novatron devices. This work is car- ried out within the framework of the Tau-E Breakthrough (TauEB) Consortium, which brings together interdisciplinary expertise under the coordination of KTH. This project has received funding from the European Union's Horizon Europe research and innovation programme, under Grant Agreement No: 101186012.

[1] J. Scheffel et al., 2025 Nucl. Fusion 65 066011 https://doi.org/10.1088/
1741-4326/add173.
[2] K. Lindvall et al., Phys. Plasmas 32, (2025) 072509 https://doi.org/10.1063/5. 0266976.

#13398

Beam power calibration and measurement of the beam properties of the MAST-U neutral beam long pulse upgrade.

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

MAST-U neutral beam heating has two high current tetrode PINI (Positive Ion Neutral Injector) beams. The beams are labelled SS and SW based on compass directions. Each PINI was designed to deliver a deuterium beam accelerated to 75 kV, 65 A, with a power of up to 2.5 MW for a duration of 5 s. In this paper we present the analysis used to determine the beam power. This is achieved by new accurate measurements of the neutralisation efficiency, including the losses in the accelerator determined by previous measurements, by modelling of the beamline transmission efficiency and by estimating the reionisation loss. Due to thermal limits, we found that it was not possible to measure the neutralisation above 65 kV and an extrapolation method is proposed to calibrate over the full range of operating voltage. The errors in these parameters are also determined. The neutralisation efficiency was used to determine the neutralisation target density, and this allowed determination of the powers in the three beam components. The determined power will be compared to that obtained from thermocouple measurements of the incident power density on calorimeter cooling elements. Finally, the determined component power fractions will be compared with spectroscopic measurements of the beam emission in the MAST-U plasma.

#13452

Recent Status and Upgrade Plan of Electron Cyclotron Resonance Heating and Current Drive System on HL-3

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

At the end of 2022, a 7.5MW ECRH system has been developed on HL-3 tokamak, which consists five 105GHz/1MW/3s subsystems, two 140/GHz/1MW/3s subsystems and one 68GHz/500kW/1.5s subsystem. The 7.5MW high-power microwaves are produced by seven 1MW and one 500kW GYCOM gyrotrons, transported by five ⌀63.5mm evacuated and one ⌀80mm un-evacuated over-mode corrugated transmission lines (TLs) and injected into plasma by three fast steerable launchers. The mode purity of ⌀40-meter TLs is about 94% which is tested on test bed. For the three launchers, the 4MW mid-plane, 2MW 1# upper and 1MW 2# upper launchers, the dynamic response time of mechanism for the full scan range in the poloidal direction is less than 100ms. During the 2025 spring experimental campaign, 1.5MA H-mode was obtained by NBI and EC, 14keV electron temperature achieved by 2MW 105GHz ECW and NTM was supressed by real-time feedback control 400kW/68GHz ECW deposition position in 50ms. After this experimental campaign, the HL-3 device is on the way to be upgraded till the end of 2026 for burning plasma experiments, including ECRH system. The designed power of the upgraded ECRH system will be up to 13MW and the frequency will cover 105GHz, 140GHz and 175GHz. This system will contain thirteen gyrotrons, eleven TLs and six launchers. Based on this design, some special EC heating and current drive experiments could be carried out on HL-3, for example, different frequency ECWs deposited at different position in one shot to watch the synergy heating and current drive effects. For the power sources, four kinds of gyrotrons were selected, original 140GHz or 105GHz single-frequency gyrotrons, new 140/175GHz dual-frequency and triple-frequency 105/140/175GHz gyrotrons, to make sure enough power injected into plasma at three different frequencies. Eleven ⌀63.5mm evacuated TLs were used to transport thirteen wave beams and the transport routing of wave beams could be changed by switch. Each TL is about 80 meters. Eleven wave beams will be injected into plasma by six launchers, one 4MW mid-plane, two 2MW upper, one 2MW bottom, one 1MW upper and one 1MW top launcher. The top launcher is remote-steering antenna and others will be front-steering antenna. The wave beam injection angles of all of these six launchers, except 1MW upper launcher, could be controlled both in the toroidal and in the poloidal direction. The functions of ECRH/ECCD will be testified for burning plasma on HL-3 in 2027.

KIT Development on Gyrotron Multistage Depressed Collectors

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

Magnetic-confinement fusion devices such as tokamaks and stellarators rely on high-power Electron Cyclotron Resonance Heating (ECRH) to sustain fusion-relevant plasma temperatures and for plasma control. The required microwave power at frequencies above 100 GHz and power levels above 1 MW is generated by vacuum-electron tubes known as gyrotrons. Large facilities operate many gyrotrons in parallel for increased ECRH power. To increase system efficiency, modern gyrotrons use depressed collectors that recover part of the spent-beam energy. Conventional Single-stage Depressed Collectors (SDCs) raise the typical gyrotron efficiency from ~35 % to ~50 %. However, implementing more efficient Multistage Depressed Collectors (MDCs), well established in other vacuum-electron devices, is challenging for gyrotrons because the strong magnetic field suppresses conventional energy-based electron separation.

The E×B-drift concept offers a viable path toward realizing MDCs in gyrotrons. At KIT, the cylindrical MDC approach has been advanced to achieve compact geometry, efficient electron separation, and low reflected current. Improvements to the helical-gap structure, electrode shaping, and drift configuration enable stable operation across a broad range of gyrotron frequencies and magnetic systems. The KIT short-pulse prototype applies an azimuthal electric field to induce a controlled radial drift, enabling two-stage energy separation in the collector. As the first MDC developed for fusion-relevant gyrotrons, it demonstrates a clear route toward efficiencies beyond 60 %, while showing strong robustness to geometric and magnetic tolerances.

These developments establish the basis for scalable MDC layouts, including a Continuous-Wave (CW) version with adapted dimensions, optimized operating voltages, and provisions for cooling and thermal load management. Current work focuses on industrialization of the CW MDC using an improved design that employs a compact conical–cylindrical first electrode to shorten the axial length and to simplify the additional collector coils. Integrated cooling beneath the helical first-stage surface is validated through thermal analyses, and the mechanical design will be compatible with ultra-high-vacuum and cooling technologies.

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Deploying the WEST ECRH System: Performance, Challenges and Lessons Learned

Lena DELPECH (CEA)

Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

WEST experiments aim to master long plasma pulses (1000 s range) and to expose an ITER-like tungsten divertor components to power fluxes up to 10 MW/m2. To increase the access to improved confinement regimes (in particular H-Mode) and control plasma core W-impurities in the plasma, the WEST Electron Cyclotron Resonance Heating (ECRH) system is being upgraded to 3 MW/1000 s at the frequency of 105 GHz.

Since 2022, the ECRH platform has undergone significant modifications to accommodate the new 105 GHz gyrotron and its auxiliary equipment. Alongside the upgrade of the ECRH generator, the RF design of the new gyrotron has been validated in 2024 in the KIT FULGOR test bench demonstrating 1 MW/5 s at 105 GHz. The commissioning of the ECRH generator is a critical phase of the deployment of high-power microwave systems on a tokamak. The installation of the first gyrotron on the WEST platform enabled its qualification with the validation of the interlock systems ensuring a rapid shutdown in the case of defaults, the tests of the mode configurations required for the WEST ECRH generator, and the design validation of the High Voltage Power Supply (for both Cathod and Body of the gyrotron). Finally, the control software (CODAC) was successfully commissioned in April 2025, enabling an accurate synchronization with plasma operations.

Following validation of the ECRH generator, a second gyrotron was installed and fully commissioned on the WEST platform. The short pulse operation was carried out to qualify the beam pattern of the gyrotron and to align the Matching Optic Unit box with the transmission line. The gyrotron demonstrated an RF power exceeding 800 kW/5 s on a dummy load and 1 MW/1000 s at the collector in diode mode regime. The duration of the RF pulse was limited by the absence of the transmission line cooling system. The gyrotron system will be gradually brought into service to full operation.

This paper describes the successful commissioning of the WEST ECRH platform, including the installation and the qualification of the gyrotrons as well as the results obtained on dummy load and during plasma operation in a short period of time. Key lessons learned are discussed for the commissioning of ECRH system such as the early validation of the interfaces, interlocks, and the involvement of the operational team from the initial stages of the gyrotron validation. The experience gained is expected to be transferable to larger scale systems.

Final design of the WEST ion cyclotron travelling wave heating system

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

Due to its demonstrated experience in a variety of scenarios, including plasma heating (including dominant ion heating), wall-conditioning, or even assisted breakdown, as well as readily available technology for continuous wave (CW) operation, Ion Cyclotron Range of Frequency (ICRF) heating has proven to be a useful tool in fusion experiments. However, the current ICRF launchers are plagued by low coupling conditions and the production of metallic impurities.

The Travelling Wave Array (TWA) antenna, an innovative and remarkably distinct ICRF antenna, has the potential to address all of these challenges simultaneously. A scale one mock-up of TWA antenna has been successfully tested at IRFM in the TITAN test-bed, up to 2 MW in vacuum. The next step is then to demonstrate the performance of this concept on plasma. Hence, an ICRF Travelling Wave Array (TWA) system for the WEST facility has been designed within an international collaboration.

The WEST TWA system aims to demonstrate the operational advantages of the TWA concept for ICRF and address the limitations of current ICRF launchers, focusing on improved coupling, reduced voltages and heavy impurities production, and enhanced reliability. Because WEST is a superconducting and long-pulse tokamak, the launchers must be actively water-cooled, which brings new challenges with respect to the mock-up. The project's first phase, which uses water loads and a direct feeding scheme, is detailed. The system specifications (3 MW/30s and 1 MW CW) and the key design decisions (such as cantilevered launchers) are described. A resonant ring feeding scheme, improving the overall system efficiency, is planned for a second phase. The RF and mechanical designs are summarized. The future commissioning and experimental program for the WEST TWA system is also detailed in this paper.

#13542

CIGALE: An innovative high efficiency neutral beam injector concept for future fusion power plants

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

High-density spherical tokamaks are regarded as promising candidates for future cost-effective fusion power plants. At plasma densities twice that of ITER (neITER~1.10exp(20) m-3), Neutral Beam Injection (NBI) is essential to ensure plant operation. However, this implies meeting stringent requirements (i.e., nuclear safety and RAMI constraints) while delivering powerful high-energy neutral beams (~20 MW, 1MeV D0 per injector) with high wall-plug efficiency (above 60%) over long-duration operation. To this end, a new injector concept CIGALE (Concept of a neutral beam Injector based on GAs neutraLization and Energy recovery) based on pragmatic solutions, is proposed.

The ion source is the key element of the beamline. It is connected to many external auxiliary systems (power supplies, gas, water) to generate the plasma and requires regular maintenance for caesium management, which is essential for the production of negative ions (D^-). In current NBI systems, the ion source is maintained at high voltage (−1 MV on ITER), leading to long and complex robotic maintenance operations as well as an invasive and complicated 1 MV electrical setup. In CIGALE, the ion source facing the accelerator is referenced to ground potential in order to improve accessibility and maintainability. Moreover, to ensure high injector availability, CIGALE is based on independent ion sources, each connected to independent power supplies. These features allow easy maintenance of one source while keeping the others—and the injector—operational. The plasma source and primary electrons are confined by a specific magnetic configuration based on Samarium-Cobalt magnets. A 1/5-scale prototype source was developed in 2025. Preliminary plasma characterization using Langmuir probes meets the prerequisite conditions for a high-performance negative ion source, revealing a high and uniformly distributed electron density ($n_e \approx 2 \times 10^{18}$ m^{-3}) at low operating pressure (P? ≈ 0.1 Pa) and low input power (The CIGALE accelerator is designed to produce thin, laminar D^- beams in order to minimize the amount of gas injected downstream into the neutralizer duct. In addition, the implementation of an energy recovery system at the neutralizer exit makes it possible to decelerate the energetic (1 MeV) non-neutralized D^- ions down to low energy (~50 keV) and collect them on an actively cooled target. This approach reduces thermal power losses and significantly increases injector efficiency.

The combination of these essential features results in the CIGALE injector concept, which offers easy maintainability, high neutral power (~20 MW), high wall-plug efficiency (~63%), and long-duration operation.

Static and Dynamic Effects Influencing the Performance of the ELISE Negative Ion Source in H and D

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The ELISE (Extraction from a Large Ion Source Experiment) test facility, with its half ITER-size negative ion source relies on the surface conversion of negative ions, favored by evaporating Cs onto the source's surfaces. It is developed to meet the requirements of the ITER neutral beam injection system regarding the high extracted negative ion current density jext of hydrogen (H) and deuterium (D) for pulse durations of up to one hour, while maintaining low co-extracted electron current density je, high beam uniformity, and low divergence. The achievable pulse length is typically limited by an increase of je, in particular in D.

It is known that both static (set by source parameters) and dynamic (introduced by the use of Cs) effects affect je. The study of these effects is so far based on datasets obtained over multiple experimental campaigns, each characterized by different source configurations, caesium conditioning status (Cs distribution on surfaces), and diagnostic availability.

To establish a coherent and operation-oriented dataset, a dedicated experimental campaign was carried out with full diagnostic coverage, fixed source configuration and stabilized caesiation (attested with very reproducible performance). Systematic variations of key source parameters -including RF power, filling

pressure, magnetic filter field, electrical biasing of different surfaces- were performed for both short (10 s) and long (300 s) extraction pulses. Peak extracted negative ion current densities of up to 150 A/m2 were achieved, representing a substantial fraction of the ITER target level (≈ 50 %).

The results show that for matched plasma densities in the plasma generation volumes, H and D exhibit mostly similar dependencies of the co-extracted electron current density on the varied parameters. With increasing RF power, isotope-dependent differences in performance become more pronounced and are attributed to a stronger degradation of caesium conditioning in deuterium operation. Biasing the first grid of the extraction system helps reducing je in H but (for that campaign) not in D. While further investigation is required, these differences are likewise believed to be linked to caesium-related effects. The temporal stability of je over long pulses is influenced by these parameters, but is dominated by the caesium dynamics.

These findings provide a consolidated, operation-oriented overview of the factors influencing the co-extracted electron current density in large negative ion sources and support the development of active caesium control concepts, such as the caesium shower approach currently under investigation, to improve performance stability in long-pulse operation.

WEST Tokamak 3D FEM Electromagnetic Analysis

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

WEST, acronym for tungsten (W) Environment in Steady-state Tokamak, is a superconducting, actively cooled fusion machine, actually operating in CEA Cadarache, in the South of France. Its mission is to explore long pulse operation in a full tungsten environment in support for next-step fusion devices (as ITER and DEMO).
This article presents the main features of the Electromagnetic (EM) studies carried out on this machine, starting from an upgrade of the 3D FEM (Finite Element Method) EM model. Specifically, the analysed environment consists of 3 of the 20° WEST sectors, modelled using the Hypermesh v2021 software and analysed with the ANSYS APDL v2023R1 code.
The choice of this particular angular span has allowed fully taking into account the actual 60° symmetry of the machine (coming from the equatorial ports alternate positioning), which cannot be completely represented with a classic single sector model. Moreover, this has also allowed adding an Ion Cyclotron Range of Frequency (ICRF) Travelling Wave Array (TWA), with a rather large toroidal size (38°) and whose design is currently under development. Finally, other important, representative components have obviously been included in the model (like vacuum vessel, ports, limiters, baffle, internal protections).
A reference WEST plasma disruption has been implemented, starting from a 1 MA reference plasma that undergoes a hot upward VDE (Vertical Displacement Event) with a thermal quench (τ_th = 0.5 ms) followed by a current quench (τ = 3 ms), simulated by means of the plasma equilibrium code FEEQS. This plasma input has been converted using a set of current filaments plus a fictitious toroidal solenoid in order to effectively take into account, respectively, the Plasma Poloidal and Toroidal Field Variation.
Toroidal Field coils (providing Btor = 3.866 T at R0 = 2.36 m), plus Poloidal Field and Divertor Coils have also been introduced, and equivalent electrical resistivities have been defined for the components materials when significant simplifications have been brought in.
Finally, cyclic boundary conditions have been applied to the limit surfaces of the model so as to effectively simulate the full toroidal extent of the machine.
All the implemented features have allowed obtaining a thorough and upgraded FEM EM model of WEST tokamak. This model is going to be extensively employed not only to assess the global behaviour of the machine, but also to evaluate the mechanical EM loads inside its current and future in-vessel components (potentially recurring to more advanced techniques like submodeling).

#13589

Dynamic modelling of the Remote Centre of Motion compliant steering mechanism of the EU-DEMO ECH Mid Steering Antenna

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

Within the EUROfusion Engineering Grant (EEG) project “ReactECH: design of Electron Cyclotron Heating launchers for the future fusion Reactors”, the first milestone targets compliant mechanisms for future ECH launchers, motivated by reactor-relevant constraints such as vacuum tribology, motion repeatability, and lifetime maximization. The EU-DEMO ECH Mid Steering Antenna (MSA), conceived for neoclassical tearing mode (NTM) control, is a prominent example: a pantograph-derived Remote Centre of Motion (RCM) six-bar linkage steers the plasma-facing mirror (M2) over a ±3° range about a virtual axis passing through the mirror optical centre, while Crossed Flexure Pivot (CFP) joints replace conventional bearings.

This contribution presents a control-oriented dynamic model of the MSA Top Steering Mirror Assembly, developed to quantify how gravity, actuation and disruption loads are transmitted to the Crossed Flexure Pivots (CFPs) that realise the linkage joints. A Simulink/Simscape Multibody model is generated from the CAD assembly and includes the six rigid links and the redundant revolute joints implemented by multiple CFPs in parallel. Pneumatic actuation with pressurised Helium, the gas line and servo-valve dynamics are integrated to reproduce the plant bandwidth and disturbance transmission. CFPs are represented through a pseudo-rigid-body approach, enabling fast evaluation of transient joint forces/torques together with CFP-level buckling indicators.

Simulations indicate that during reference tracking (steps, ramps and modulation cycles at nominal speed and acceleration) CFP loads remain well below the buckling instability threshold. In contrast, disruption-driven EM torques concentrate critical loads in specific joints, with CFPs approaching the buckling limit for upward Vertical Disruption Events. Potential dynamic amplification justifies design revisions, either by reducing disruption loads at source via mirror geometry/material optimization, or by re-sizing/re-distributing the CFP sets to increase stability margins.

Finally, the work discusses the pros and cons of this type of RCM compliant architecture over alternative concepts for a protected virtual pivot (mechanism kept in the mirror shadow) with frictionless, backlash-free motion suited to reactor-grade operation.

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Engineering based model for ECRH heating system integration

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

In the context of the future fusion machines, the Electron Cyclotron (EC) heating systems are good candidates to increase the plasma performance. To access the operational space in H-mode in WEST, a new 3 MW ECRH system comprising three gyrotrons at a frequency of 105 GHz and a water cooled antenna with three mirrors is being installed [1,2].
During the installation and commissioning, from gyrotrons to antenna, the significant technical challenges, highlighting the urgent need to simplify and streamline the tools that support this complex integration process for the future fusion plants.
In this work, we discuss a Model-Based Systems Engineering (MBSE) approach aimed at improving the clarity, consistency, and accessibility of system specifications and existing interfaces for the teams responsible for commissioning, operating, and maintaining the system throughout its lifecycle. There are several aspects to take into account, the model will be composed of physical and functional description, how the system operates, the functions it performs, and the relationships between its different subsystems, requirements that the system needs to satisfy, system interface and interaction with its environment.
Currently, the ECRH specifications are written in a document-centric format rather than in a digital, model-based form. To support this approach, a dedicated tool is being developed which is capable of processing an existing requirement or specification document, including its sections, subsections, and tabular data, and automatically extracting, organizing, and structuring the information. The tool converts unstructured or semi-structured textual content into a formalized system model, enabling clearer visualization of requirements, interfaces, and relationships among the system components. The proposed solution facilitates improved requirement traceability, impact analysis, and interface management. It also supports collaboration among engineering teams by providing a unified and structured view of system data. Instead of modeling individual objects independently, the methodology could integrate detailed numerical models and system-level codes in order to represent the complete physical behavior of the installation. This would allow the integration of full system dimensions, operational constraints, and performance specifications directly into the engineering environment. Moreover, this work could be coupled with system codes, in order to integrate more than simplified object into program but at final the whole systems dimension and specification constraints. This method has the potential to be generalized to other types of heating systems beyond the specific case studied in this work.

Simulations estimating the space-charge neutralization level in the 1.5 MW & 1 MW 140 GHz gyrotrons for the W7-X stellarator

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

In this work, we investigate the space-charge neutralization level in the existing 1.5 MW and 1 MW gyrotrons operating at 140 GHz in the W7-X stellarator [1, 2]. This study aims to revise the theoretical predictions of the neutralization effect [3], which is a critical factor in the design and development of a future 2 MW gyrotron for W7-X [4]. The neutralization level is estimated by comparing output power curves, experimentally measured at different accelerating voltages near the point of operating mode loss [5], with corresponding simulations performed using the interaction code EURIDICE [6]. Specifically, the beam optics code TRAK [7] is used to generate a set of electron beams for the simulations, with neutralization levels varying from 0% to 70%. The objective is to identify the neutralization level at which the interaction simulations align with the experimental curves, with respect to the RF power and to the accelerating voltage where mode loss occurs. The simulations indicate that the experimental neutralization levels can differ from those predicted by the potential wells formed by the electron beam.

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Stripping losses in the negative ion source ELISE: analysis, simulation and design for their reduction

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

For the ITER neutral beam injection system, stripping losses are expected to reach approximately 30% within the seven-grid system at a filling pressure of 0.3 Pa [1]. A relevant loss mechanism for negative ions in the extraction grid system is stripping – unwanted ionization of negative ions caused by collisions with background gas molecules along the beam path before fully accelerated.
In the ITER-relevant neutral beam injection test facility ELISE (Extraction from a Large Ion Source Experiment), a three-dimensional numerical simulation model of the molecular fluid region has been developed using the open-source Monte Carlo code MolFlow+ (CERN) [2]. Similar to the approach employed by RFX using the AVOCADO code [3], the model incorporates the source, three-stage extraction system and vacuum chamber as essential components, that mainly determine the gas conductance [4].
The molecular flow model is validated by comparison with measured gas flow rates and the pressure of the ELISE source and vacuum chamber. Based on the validated conductance, the density distribution of the background gas molecules is calculated and resolved inside the two stages, serving as an input for the stripping loss simulation of the current design. A modified grid design is proposed to reduce gas density downstream of the plasma grid, thereby reducing stripping losses. Simulation studies of the revised design demonstrate a reduction in stripping losses.

#13640

Expanding the DIII-D ECH system to 10 gyrotrons

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

Electron cyclotron heating (ECH) and current drive (ECCD) play a large role in the operation of the DIII-D tokamak. Microwaves injected into the plasma can provide core heating and current drive as well as impurity control, neoclassical tearing mode mitigation, and breakdown assistance. Total power injection by ECH systems is relatively scalable because many gyrotron sources can be located far from the tokamak while compact, low-loss waveguide transmission lines deliver high power to the plasma without requiring as much space near the tokamak as some other heating methods like neutral beam injection. DIII-D is presently expanding its ECH system from six to ten MW-class gyrotron sources. The existing system routinely uses six 110 GHz Microwave Power Products (MPP) gyrotrons with 31.75 mm diameter corrugated waveguide and quasi-optical launchers to inject up to 3.4 MW of power to the tokamak. The expansion will integrate two new 117.5 GHz gyrotrons from Thales Group into the existing waveguide and launcher systems with the 31.75 mm waveguides, and it will install two new multi-frequency (104/137/170 GHz) gyrotrons from Kyoto Fusioneering with new 60.3 mm diameter corrugated waveguide and a new dual launcher. This poster presents the up-to-date status of the installation and commissioning of the expanded system, and it compares the operational requirements of diverse gyrotron set on the facility's resources.
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#13662

Operational Experience and Upgrade of the ECRH System at Wendelstein 7-X

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

W7-X is a steady-state capable optimised stellarator. The main heating system is ECRH with currently 11 gyrotrons operating at 140GHz providing 8.5MW heating power into the plasma vessel. In the past years we continuously increased the reliability and availability of the ECRH in order to provide sufficient heating power for the W7-X experimental programme. The availability of heating power for high performance plasma operation was in the past restricted by the number of available high-voltage supplies, which are shared between ECRH, NBI and ICRH. In order to overcome this problem, we installed manual switches to operate two gyrotrons in parallel on one cathode supply. They are capable of driving two gyrotrons in parallel for about 15s, which is sufficient for most of the W7-X physics programme.

Reliability of ECRH is still an issue for long pulse operation. In the past we installed several measures to increase the reliability of individual gyrotrons. This included hardening of the transmission line against arcing and control techniques aiming at an automatic recovery in case of a failure. In the last campaign (year 2024/2025), a feedback control system stabilising the total power was implemented in the ECRH control system. In case of gyrotron failure the controller increases the output power of all remaining gyrotrons within less than 30ms. Ideally, the tripped gyrotron resumes operation latest after a few seconds so that the increased output power is only a temporary intervention. The successful application of this control technique enabled us to achieve an energy turnaround of 1.8GJ in an experiment programme with 4.9MW ECRH power for 360s.

By achieving this milestone we hit a new operational limit which is full power operation in very long pulses. Arcing on the transmission line, which we believed to be reduced to an acceptable level, becomes again a significant problem after ~5 minutes. We identified critical components and present strategies to overcome this problem.

The total available ECRH power is currently being upgraded by installation of an additional gyrotron on a spare 12th position. In addition, our weakest gyrotrons will be replaced by more powerful ones. In the near future, we are aiming at 2MW gyrotrons, which are currently under development.

In addition, a new W7-X operating scenario was established at 1.8T with multipath heating at the third harmonic resonance of the 140GHz gyrotron frequency. For plasma breakdown, one of the 11 gyrotrons was tuned to a frequency of 101GHz.

Hot Isostatic Pressing (HIP) for the manufacturing of the Blanket Shield Module (BSM) of the ITER Electron Cyclotron Upper Launcher (ECUL)

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The Electron Cyclotron Heating (ECH) system of ITER plays a key role in achieving and maintaining the high temperatures required for nuclear fusion within the tokamak, providing plasma heating and current drive, and magnetohydrodynamics stabilization. The ECH system is increasingly becoming the reference mechanism for plasma heating for fusion devices. In ITER, the ECH will inject 40 MW to 67 MW of electromagnetic continuous waves into the plasma. This power is transmitted to the plasma by 1 MW 170 GHz mm-wave beams that propagates through a network of corrugated waveguides.

The EC Upper Launchers (UL) are located in dedicated ITER Vacuum Vessel upper ports, of the Tokamak system, and include the front-end quasi-optical components of the ECH system (four mirrors and eight in-vessel waveguides). The primary UL functions are directing the beams to the appropriate target position in the plasma, providing shielding and vacuum and primary confinement of radioactive and hazardous substances.

A key component of each UL is the Blanket Shield Module (BSM), directly facing the plasma. The BSM is formed by 3 subcomponents named: Flange, Shell and Front Wall Panel (FWP). The FWP comprises several cooling channels and various manifolds responsible of distributing the internal cooling flow. An important feature of the BSM is the CuCrZr heatsink, protected by a stainless-steel layer, and actively cooled through the embedded stainless steel cooling network. Due to its complex design, the BSM is not suitable for conventional manufacturing.

Fusion for Energy (F4E) has acquired significant experience, together with industrial partners such as Alsymex, in the use of Hot Isostatic Pressing (HIP) for the manufacturing of ITER prototypes, including one BSM prototype.

HIP is a diffusion bonding technique that simultaneously applies high temperature, inert gas pressure in an isostatic manner to achieve permanent bonding between different materials. As part of the qualification activity performed for using HIP to manufacture BSM series, an extensive testing campaign (including both destructive and non-destructive testing and metallographic tests) has been

performed to qualify HIP joints diffusion bonding of different configurations, such as 316L(N)-IG to 316L(N)-IG or 316L(N)-IG to CuCrZr.

The results presented here confirm that the HIPped joints have better mechanical properties than the weakest material in the joint, confirming that HIP is a reliable option for manufacturing such components.

MISTRAL assessment of stray radiation shielded enclosures

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

Electron Cyclotron Resonance Heating and Electron Cyclotron Current Drive (ECRH/ECCD) are key actuators on present-day fusion magnetic confinement devices. Megawatt-level power at frequencies in the 100–200 GHz range is injected as Gaussian beams and absorbed at the electron cyclotron resonance layer, producing localized electron heating. For a variety of reasons, not all injected power is fully absorbed and microwave stray radiation inside the vessel result. This power is incident on all in-vessel structures and penetrates all sub-volumes in the vessel. The interior inside the sub-volume and its internal components are not all good electrical conductors, and microwave absorption can lead to overheating or damage. Examples include the cryogenic divertor and the ECRH launcher enclosure. In this work we report on two case studies: the shielding of the steering mirror assembly mechanism of the ITER Upper Launcher and the baffles of the JT-60SA cryo pumps. Both have been designed to reduce microwave power entering the enclosure. Both were tested inside the MISTRAL facility to assess the ratio of stray radiation outside the enclosure to that inside it. For the ITER launcher, stray radiation levels were also measured as a function of the polarisation of a collimated beam incident on enclosure slits, together with resulting thermal loads on internal components. The experimental arrangement, test methodology, and diagnostics are described. Simple models are used to estimate stray-radiation attenuation and are compared with measurement.

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The Divertor Test Tokamak, DTT, aims at studying possible solutions to power exhaust issue to pave the path to DEMO. To fulfil its mission, DTT shall host experiments where the flowing power is at DEMO relevant values, i.e., PSep/R ≈ 15 MW/m, requiring a reliable coupled power in the range of 45 MW. Moreover, DTT will have extreme flexibility to allow testing different divertor scenarios. The request of having reliable power in scenarios that can vary significantly, also in terms of plasma shaping and edge conditions, imposes severe conditions for safely coupling Ion Cyclotron power to the plasma.
Significant effort was devoted to the water-cooling design of the antenna's main plasma-facing components: the Central Strap, Lateral Straps, and Faraday Screen (FS). For the Central Strap, successive CFD-driven design iterations demonstrated that a reduced number of cooling channels with higher flow velocity provides a more uniform temperature distribution and substantially lowers peak temperatures, while maintaining acceptable pressure drops. The Lateral Strap cooling design, based on the same AM-compatible channel concept, was shown to meet temperature limits for the main strap body.
The Faraday Screen represents the most challenging component from both thermal–hydraulic and structural perspectives. Different sets of cooling layouts were studied adopting detailed CAD model and assessed through CFD and FEM analyses.
Coupled electromagnetic–structural analyses were performed to assess the mechanical integrity of the ICH antenna under conservative disruption scenarios. Mechanical design activities also focused on the main supporting structure that was conceived relying on commercially available structural steel beams, connected using bolted joints. A preliminary FEM analysis confirmed the viability of proposed structure

Mechanical design and functional testing of a high-vacuum compatible motion system for a retractable diagnostic calorimeter for MITICA

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

MITICA is the full-scale prototype of the ITER Neutral Beam Injector, developed to validate the design and performance of the negative ion source, beam extraction, acceleration, and neutralization systems. The extractor and accelerator of the MITICA Beam Source (BS) are designed to produce a beam with stringent aiming requirements to ensure proper injection of the beam.
Correct beam aiming must be experimentally verified using a dedicated diagnostic capable of measuring the beam spatial distribution with resolution sufficient to resolve individual beamlets. For this purpose, an inertially cooled, retractable calorimeter is foreseen, installed directly on the BS to achieve the highest possible resolution and accuracy. The mechanical system responsible for insertion and retraction of the calorimeter must guarantee extremely high reliability, as access for servicing is only possible during scheduled BS maintenance, and any failure leading to loss of position control is unacceptable. Consequently, the motion system is driven from outside the vacuum vessel, with motion transmitted to the BS through a high-vacuum-compatible mechanical transmission. The system is designed to accommodate BS alignment and tilting movements and to withstand exposure to stray particle and heat fluxes during operation. This paper reports the mechanical design, integration constraints, and functional testing of the proposed motion system.

A new control system for the TCV Diagnostic Neutral Beam Injector

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The ion temperature, toroidal and poloidal velocity and carbon impurity density profiles have been measured on the TCV tokamak by Charge eXchange Recombination Spectroscopy (CXRS) diagnostic, using a diagnostic neutral beam injector (DNBI) since 2000. Over the years, the plasma source and the ion optics system have undergone several upgrades, while the system's power supplies have remained largely unchanged since its last major revision in 2006. The operation of the DNBI System is currently limited by the obsolescence of both its power components and control infrastructure. To ensure efficient and reliable future operations, the power supplies and control system are being replaced with modern, high-performance solutions.

The DNBI Control system manages the operation and synchronization of its nine power supplies. It also ensures the correct DNBI operation in case of breakdowns in the ion optical system. The controller of the arc-discharge plasma source (ARC) located close to the DNBI is the master. It is connected to the high-voltage modulator (HVM) control, located outside the TCV area, through RealSync, an Imperix proprietary protocol, resulting in a single distributed control system. The DNBI ancillary subsystems (vacuum, cooling, gate valves, safety, etc.) are controlled by Backhoff based CS with interlocks routed to the ARC controller. Both the interface with the TCV control system and the graphical user interface (GUI) are implemented using OPC UA, which simplifies the monitoring and control of the system.

This paper begins with a brief introduction to the DNBI operation principle, followed by a detailed description of its control infrastructure, including each subsystem. Particular emphasis is placed on the synchronized operation of the power supplies at the microsecond scale, addressing the control loop design and programming code assessment. Additionally, the paper discusses the interactions of the DNBI control system with other systems, its operation modes, and the results of commissioning tests. Significant effort has been dedicated to integrating the new DNBI control system into the TCV synchronization network (White Rabbit) and the centralized TCV data acquisition system (MDS+). The ARC controller serves as the interface for both purposes, synchronizing the RealSync clock system with the White Rabbit network, An MDS client installed on one of its CPUs eliminates the need for additional DNBI data acquisition devices. This integration modernizes the interface and enhances control, monitoring, and diagnostics capabilities.

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Alternative caesiation method for a half size ITER Negative Ion Source

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The negative ion sources of the Neutral Beam Injector (NBI) for ITER shall deliver 286 A/m^2 over pulse length of up to 3600 s in deuterium and 330 A/m^2 over up to 600 s in hydrogen. Operative at IPP Garching are the two tests facilities BATMAN Upgrade and ELISE (respectively 1/8 and 1/2 size of the ITER ion source). The production of hydrogen negative ions takes place on caesiated low work-function surfaces, mainly the plasma grid (PG), the first grid of the extraction system. Conventionally, caesium is injected into the ion source by means of one or more ovens located at the backplate or side walls of the ion source. This concept relies on the re-distribution of caesium by the plasma. Usually the negative ion current density extracted during long pulses is limited by the pronounced increase of the co-extracted electron current, particularly in deuterium operation. Thus, keeping the low surface work-function stable and homogeneous during long pulses is crucial. For that, the caesium flux onto the PG surface needs to be sufficient in order to counteract the degradation of the caesium layer caused by the interaction with the plasma. The sole increase of evaporation rates using the conventional method can lead to hampered plasma production in the drivers and to excessive caesium in the extraction system causing break-downs. Moreover, caesium is extensively ionized in the plasma and can hardly reach the positive biased PG. In order to address this issue, an alternative caesiation device called caesium shower was designed to evaporate caesium directly over the PG, allowing a more direct control of the caesium flux onto the grid. Tests at BATMAN Upgrade demonstrated that the caesium shower stabilizes the co-extracted electron current during long deuterium pulses, enabling it to achieve 90% of the ITER requirement for the negative ion current density during a 1000 s pulse. Based on these positive results, a similar device is being developed for the ELISE test facility. The design process is based on a molecular flow simulation in MolFlow+. Which is validated by comparing simulation results for BATMAN Upgrade with measured caesium densities using a Tunable Diode Laser Absorption Spectroscopy diagnostic. BATMAN Upgrade has only one beamlet group while ELISE eight of them, the main challenge for ELISE is to achieve similar functionality over a much larger area. The final design of the Cs shower for ELISE is presented in this contribution.

Development of a hybrid transmission line concept for a power plant EC system

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

In future fusion power plants, the Electron Cyclotron (EC) Heating system could play a fundamental role in starting, shaping and controlling the plasma from start-up to the burn phase until plasma termination, also providing a means for instability suppression. Many MW of power should be provided with high reliability at different times and plasma locations during pulses. While gyrotron sources are being developed in collaboration with research Institutes and Industry, and EC launchers are specifically tailored to the machine port size and the power to be injected into the vessel, Transmission Lines (TL) are parts in common to all EC systems that can be developed independently of the machine architecture. Industrial standard solutions composed of corrugated evacuated waveguides will prove their manageability in ITER. For the EU-DEMO, designed within the EUROfusion consortium, a hybrid TL, adopting both waveguides and a modular multi-beam (MB) concept based on the W7-X quasi-optical transmission line (but under vacuum) was proposed. The concept, organized in modules of 7 to 9 beamlines, may simplify the layout and the maintenance of a TL with respect to a full waveguide one, and could be easily extended to a Fusion Power Plant when the space for a straight line from the gyrotron hall to the reactor hall is allocated from the beginning. The developed TL transmits multiple frequencies at the same time and adapts to different requests of power. Recently, the layout of the transmission line described in [1] has been further developed and revised, introducing the needed changes, defining mirror envelopes and mountings and considering well-proven and new options for mirror and waveguide alignment. Further studies on mirror cooling, deformation effects and reliability of a system including more than a hundred sources give a better understanding on what to expect from a hybrid Waveguide and MB TL.

[1] A.Bruschi et al., Fusion Engineering and Design 219 (2025) 115241

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Speeding up beam simulations with Physics-Informed Neural Networks: a proof-of-concept approach to solve the Poisson equation for negative ion beam simulations

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

Achieving efficient plasma heating in ITER relies on Neutral-Beam-Injectors (NBIs) capable of delivering particles at energies close to one MeV. At such high energies, negative ions are accelerated to ensure effective neutralization. Their production and extraction are sensitive to the plasma source, and their trajectories are determined by the electric field in the accelerator grid and their own space-charge. Therefore, numerical modelling is an essential tool for predicting beam properties and relies on an iterative solution of the Poisson equation coupled with particle tracing methods. For this purpose, codes such as IBsimu couple the ion trajectories tracing solver with the electrostatic potential obtained from the Poisson equation solver. This approach is computationally demanding, with runtimes of approximately an hour for SPIDER, the 1:1 prototype of the negative ion source for the ITER NBI, further increasing on the larger meshes used in full scale devices such as MITICA, the prototype of the ITER neutral beam injector.
This paper investigates machine learning strategies to reduce the computational burden needed to estimate the electrostatic potential solution of the Poisson equation. For this task, Physics-Informed Neural-Networks (PINNs) are explored. A dataset has been constructed from an IBsimu simulation on the SPIDER geometry consisting of 25 iterations of the Poisson solver, each one with a different charge distribution. The PINN is implemented as a three-dimensional Convolutional Neural-Network that takes as input the charge distribution at the previous iteration and the electrode voltage map, and outputs the estimated electrostatic potential distribution (Vpinn). The CNN architecture is optimized by minimizing a loss function, which takes into account the mean squared error of Vpinn with respect to both the IBsimu computed potential in each point of the accelerator geometry and the imposed electrode potentials, together with the physics residual of Poisson equation and the error on the corrected charge distribution using the Vpinn. Indeed, the Vpinn values must be coherent with the global charge neutrality condition in the plasma source region. A preliminary version of the model was trained to reconstruct the potential corresponding to iteration #2 of the dataset. The PINN well reconstructs the potential distribution, with an average normalized error of 0.25% on the mesh, and the residual indicates results consistent with the Poisson equation solution. The results are encouraging and motivate the extension of the approach to multiple operating scenarios, aiming to integrate a PINN-based solver into full beam simulation workflows.

Fastening and alignment concept of the structural components of the EC heating and control launcher for the EU DEMO LAR fusion power plant.

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

For the EU DEMO Tokamak fusion power plant in its latest LAR (Low Aspect Ratio) configuration, an EC (Electron Cyclotron) heating and current drive system is currently in its conceptual design phase. Six EC launching antennas will be installed into equatorial ports of the machine for injection of up to 130 MW millimeter-wave power into the plasma for NTM-control, bulk heating and radiative instability suppression. The millimeter-waves, being generated in Gyrotrons outside the Tokamak building will be transmitted through dedicated beam lines into the launchers, where an optical system, composed from waveguides and mirrors is installed into massive structural components to guarantee precise deposition of the millimeter wave power at dedicated locations in the plasma.
The structural components of the launcher include two port plug mirror modules, an additional shielding module, and the port closure plate, which also serves as first containment structure of the plasma chamber. Due to significant exposure to electromagnetic forces, all structural components must meet high requirements regarding secure fastening while at the same time high demands are required on the adjustment of the optical system. The need for robot-controlled maintenance concepts increases the complexity on the design of the fastening elements.
This paper presents the layout and the fastening and alignment concept of the EC launcher structural components. Taking into account selected electromagnetic and operational load cases, the mechanical compatibility of the fastening devices is demonstrated by structural analyses.

This work has been carried out within the framework of the EUROfusion Consortium, via the Euratom Research and Training Programme (Grant Agreement No 101052200 — EUROfusion) and funded by the Swiss State Secretariat for Education, Research and Innovation (SERI). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union, the European Commission, or SERI. Neither the European Union nor the European Commission nor SERI can be held responsible for them.

#13887

Integrated multi-physics simulations of a gyrotron cavity equipped with TPMS-based cooling structures

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

In nuclear fusion devices, Electron Cyclotron Heating (ECH) Systems rely on gyrotrons to generate high-frequency electromagnetic waves, which are then injected into the plasma. The internal surface of the gyrotron resonance cavity is exposed to extremely high thermal loads, with operating conditions resulting in peaks of up to 20 MW/m^2. Such severe conditions demand advanced cooling solutions to prevent excessive temperature rise and mechanical deformation of the cavity. This would result in a frequency shift and a reduction in the efficiency of the generated waves, as well as causing plastic deformations that would require costly maintenance.
Triply Periodic Minimal Surface (TPMS) lattices are a promising solution for this application thanks to their outstanding geometric versatility. Their porous architecture provides a high surface-to-volume ratio to enhance heat transfer, and their smooth, continuous morphology helps to limit pressure losses. Furthermore, additive manufacturing provides the flexibility to exercise precise control over important design parameters such as porosity, unit cell size, wall thickness and lattice orientation. This enables the cooling structure to be customized to meet specific thermo-mechanical requirements.
In this work, TPMS-based cooling structures employing Gyroid and SplitP topologies were investigated for a 1 MW, 170 GHz gyrotron cavity. Several configurations were designed by varying the main morphological parameters, including porosity and cell size, and their performance was assessed through thermal-hydraulic and thermo-mechanical numerical simulations using subcooled water as coolant. Two selected configurations were analyzed through the iterative process of MUCCA (MUlti-physiCs tool for the integrated simulation of the Cavity), which couples thermal-hydraulic, thermo-mechanical, and electromagnetic simulations to account for the mutual interaction between thermal loads, structural deformation, and cavity frequency response. The results were compared with those of a conventional reference configuration (RR), demonstrating the high effectiveness and flexibility of TPMS structures, and confirming their strong potential as an advanced cooling solution for gyrotron cavities.

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Progress on DTT ECRH: from gyrotron testing to load integration

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The Divertor Tokamak Test (DTT) facility, currently under construction in Frascati (Rome), will be equipped with an ECRH system designed to help achieving its power exhaust objectives. In its first phase, the system will rely on 16 gyrotrons, organized in clusters of eight units each and powered in pairs by four main high-voltage power supplies (MHVPS) under procurement.
The pre-series gyrotron, developed within the joint procurement with Fusion for Energy for the European gyrotrons supply to ITER and DTT, was manufactured by THALES and successfully passed acceptance tests at the FALCON facility.
The THALES gyrotron delivered 1 MW of RF power at 170 GHz with a pulse duration of 100 seconds. A reliability of 95% was also demonstrated by achieving 19 consecutive successful pulses out of 20 attempts.
Procurement of the first four RF loads for the first two clusters is underway and is scheduled for completion in 2026. These compact spherical matched loads are designed for high-vacuum operation and are suitable for both short and long pulses (1.2 MW, 100 s) at the gyrotron frequency of 170 GHz, with the same input Gaussian beam size. The first load has been tested at nominal power at the FALCON facility using the first DTT gyrotron and the results will be presented in this paper.
An evacuated Quasi-Optical Connection Line (CL) connecting the gyrotron to the RF load has been designed and manufactured. Procurement and assembly of the steering mirror mechanism and base support have also been completed. The mirror, made in CuCrZr and produced by additive manufacturing, successfully passed hydraulic testing, and its steering mechanism was verified under vacuum at a pressure of 6×10^{-4} Pa. The CL and the mirror will be validated at high power at FALCON using the pre-series DTT gyrotron.

The RoX (Return of Experience) of the MITICA beam source and beam line components manufacturing and assembly – link to the Technology Readiness Level (TRL).

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

The MITICA experiment, a full-size prototype of the ITER Neutral Beam (NB) heating injector, is key in the development, test and optimization of the Beam Source (BS) and Beam Line Components (BLC) designs. This is the last phase in the prototype's roadmap towards the manufacturing of the ITER NBI, with its stringent requirements and the capability to deliver 16.5MW of power to the plasma. The BS includes an advanced multi-aperture, multi-grid accelerator and an ion source based on the development of an RF driven negative ion, with modifications associated to the experience gained from the full-size ion source prototype being tested at the SPIDER facility in Padua, Italy. The BLCs are comprising the Neutraliser and the Electron Dump (NED), the Electrostatic Residual Ion Dump (ERID) and the Calorimeter (CAL). As first-of-a-kind components, the activities related to the BS and of the BLC expanded over multi-years contracts in stages from manufacturability and assembly studies up to components fabrication and acceptance testing. This paper summarizes eight years of contract follow-up, detailing the design, manufacturing processes and challenges in developing the BS and BLCs, connecting this FOAK components manufacturing to continuous evolution of their associated Critical Technologies and components TRL

Application of electron fence for reduction of co-extracted electron for the negative ion source of JT-60SA

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

The Negative-ion-based Neutral Beam Injector (N-NBI) for JT-60SA is required to inject neutral beams of 10 MW from 2 negative ion sources for 100 seconds, for which a beam acceleration of 500 keV and 22 A from each negative ion source is required. To achieve such high-current beam, it is necessary to extract and accelerate negative ion beams with current density of 130 A/m2 through 1080 apertures in large-area grids (1.2 m × 0.5 m).

To suppress co-extracted electrons extracted with the negative ions, a magnetic filter is generally used in negative ion sources. For the JT-60U, a PG filter was formed by a few kA currents through the Plasma Grid (PG). However, in the PG filter, the direction of the B×∇B drift in the source plasma is unidirectional, causing the high-energy electrons emitted from the filament cathode to be spatially biased in one direction, which causes an issue of a spatial non-uniformity of negative ion production at the PG surface, and a non-uniformity of negative ion beams extracted from the large-area grids. Therefore, for the JT-60SA, the magnetic filer was changed to the tent filter, which is formed by setting strong magnets at the top and the bottom of the ion source. In the tent filter, the magnetic field lines run from the top to the bottom of the ion source, and the direction of the B×∇B drift is reversed between the ends of the ion source. By these drifts, the high-speed electrons can circulate in the source plasma, which improved the spatial uniformity.

However, compared to the PG filter, the field strength near the PG is lower in the tent filter, which increases heat load on the extraction grid by the co-extracted electron. For the deuterium operation of JT-60SA, it was required to reduce co-extracted electron by 40 %. Therefore, an electron fence was installed to the top surface of the PG to shield electrons before approaching the vicinity of the PG. With the JT-60SA ion source, the effect was investigated by extracting negative ion with and without electron fence at the center and the edge of the ion source with varying thickness. As a result, electron current was reduced with the electron fence. By extrapolating the relation between electron currents with and without electron fence, under conditions corresponding to the deuterium operation, electron current is to be decreased by 43%, which satisfies the requirement in JT-60SA.

Towards NIO2, a 5 kW RF compact H- ion source

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Topics

C. Plasma Heating and Current Drive

Communication type

Oral

Abstract

Neutral beam injectors (NBI) are a fundamental component of fusion reactors and tokamak experiments, providing both steady plasma heating and current drive; direct fusion of beam against reactor ion nuclei (proven in JET) may offer some insight about advanced concepts as colliding beam fusion and suprathermal particle effects. The negative ion sources of neutral beam injectors (NBI) must be carefully designed and optimized for low beam divergence, minimal Caesium use, and stable HV acceleration, fostering the need of advanced plasma modeling of beam extraction. In this context long term operation (LTO) of the source is challenging, because not only plasma but also evolving surface wall conditions are involved in H- or D- production. Therefore operation of reduced size or compact ion sources are very convenient for studying the underlying physics as they provide the flexibility to test innovations, not available in much larger sources as SPIDER. The NIO1 (negative ion optimization phase 1) source, with 9 beamlets of H- ions and about 2 kW RF power, has operated in LTO demonstrating evidences of slow evolution and of fast transients, which can be stabilized through conditioning procedures . Based on these experiences, the design of a companion source NIO2 is proposed, with a larger expansion chamber and at least 25 beamlet H- extraction, for better beam uniformity studies. Generalization of the seamless match of magnetic filter and multipole confinement introduced in NIO1 is discussed, considering optimization of expansion backplate magnets. The spacing between source rear plate and RF coil and the large aspect ratio length/radius of this coil (which were introduced as design rules in NIO1) are maintained. Results and issues with Langmuir probe RF compensation measured in MetAlice LNL test-stand are also summarized. Improved gas and Cs injection are also proposed. The application of additive manufacturing at least to Faraday Shields (as recently build) and to the extraction grid is fostered.

An upgrade of the electrode-discharge system for coaxial helicity injection on the QUEST spherical tokamak

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

Non-inductive plasma current startup without relying on a central solenoid is a critical issue for spherical tokamaks. Transient Coaxial Helicity Injection (CHI) is a powerful method for non-inductively starting up a spherical tokamak. In CHI, a transient high voltage is applied to electrodes installed inside the vacuum vessel to initiate a discharge. The initial discharge path is formed along open magnetic field lines (the injector flux) connecting the electrodes, which are produced by the externally applied toroidal and poloidal magnetic fields.

On the QUEST spherical tokamak, a cylindrical electrode is installed coaxially with the central post. The discharge occurs between the electrode and the central post, with the vacuum vessel grounded. This configuration is being investigated as a reactor-relevant CHI startup system.

The electrode-discharge system has undergone regular upgrades. The electrode-discharge system was upgraded by reducing the capacitor bank capacitance from 5 mF to 3 mF using three 1 mF capacitors connected in parallel. The operating charging-voltage was increased from −2 kV to −3.5 kV, and the charging power supply is designed to operate up to −5 kV for future experiments. With this modification, closed-flux plasma currents exceeding 125 kA were generated after the externally injected current was reduced to zero. Furthermore, to enable operation at lower density, the hydrogen gas injection line for CHI was modified to increase its conductance. The improved conductance reduced excess gas accumulation in the vessel and enabled discharges at lower gas pressure. As a result, the electron density decreased to the 10^18 m^−3 range in CHI plasmas, suggesting that transient CHI plasmas can be more effectively coupled to additional heating methods.

As a future upgrade, a transient high-current power supply is being developed to increase the injector flux by 50%, since it is proportional to the closed-flux current. This would increase the current in the injector flux coil from the present 8 kA to 12 kA.

As a next step, a modification of the cylindrical electrode is planned. The new electrode is designed with a narrow surface to share the bottom divertor area with other plasma-facing walls. Since CHI is just used in the tokamak start-up phase, the compactness is critical to a reactor-relevant electrode. To reduce impurities in plasma, the redundant insulator ceramics are removed, and tungsten may be used for the part of the new electrode.

#13985

Implementation of an 8.56-GHz polarization-selected electron cyclotron heating system on the QUEST spherical tokamak

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

Non-inductive electron cyclotron (EC) plasma current ramp-up experiments using 28-GHz second-harmonic extraordinary-mode (2-X) waves have been conducted on the QUEST spherical tokamak [1]. The major and minor radii of the QUEST are 0.64 m and 0.41 m, respectively. By controlling the EC wave injection angle, a high plasma current of 80 kA or a relatively high bulk electron temperature of ~500 eV was obtained. However, since the 28-GHz second-harmonic resonance layer is located at a major radius R^2nd of 0.32 m on the high-field side at the center magnetic field B_0 of 0.25 T, the ramped-up plasma with the high electron temperature is restricted to a small region around R^2nd.

To extend EC heating capability on QUEST, a polarization (mode)-selective 8.56 GHz ECH system with a 250 kW klystron has been developed for ordinary(O)- and extra-ordinary(X)-mode wave injections [2]. At B_0=0.25 T, the fundamental resonance layer is located at a major radius R of 0.52 m, which is expected to attain bulk electron heating in a larger plasma. Four transmission line system was adopted to avoid damage at fragile vacuum windows for the high-power transmission. The four transmission lines are combined into the mode-selected one waveguide line in vacuum after the vacuum windows. A quasi-optical mirror was developed to focus the wave launched from the mode-selected waveguide.

This study focuses two subjects on implementation of the polarization-selected electron cyclotron heating system on the QUEST. The first is power combining from the four transmission lines to one mode-selected vacuum waveguide. The selected mode is the O-mode for a main experimental scenario of the fundamental resonance heating. The transmission line from the klystron to the QUEST vessel was carefully set-up with the power combining using H- and E- plane tees and a magic tee. The O-mode waves amplified by the klystron were successfully transmitted and injected into QUEST. A combining efficiency of a 98 % was achieved at a 10 W / 10 ms injection.

The second is the achievement of non-inductive EC tokamak formation by using the 8.56-GHz EC system. As a first trial of the non-inductive EC tokamak formation, the same experimental scenario with B_0=0.139 T as that used for the long-pulse discharge [3] at a relatively low power level is adopted. The tokamak formation was successfully achieved with stable 10-kW power injection for 10 s.

Numerical study on square corrugated waveguide FADIS properties in dual-frequency (110/138 GHz) operations based on beam propagation and coupling analyses

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Topics

C. Plasma Heating and Current Drive

Communication type

Poster

Abstract

In nuclear fusion research, high-power millimeter-wave electron cyclotron heating and current drive is a promising method for localized plasma heating and current drive. Specifically, in tokamaks, synchronized injection into rotating magnetic islands of the neoclassical tearing mode (NTM) suppresses instabilities more effectively than continuous injection. While conventional synchronized injection via gyrotron power modulation suffers from 50% lost opportunity, high-speed switching using a fast directional switch (FADIS) offers a more efficient alternative. A previous study [1] proposed a diplexer system using a square corrugated waveguide (SCW) splitter as a FADIS for the dual-frequency (110/138 GHz) gyrotron in the JT-60SA project. This approach has advantages to handle dual-frequency cross-polarized fields without requiring fragile materials. Although SCW-based dual-loop resonant rings showed sufficiently sharp resonance, it was evaluated using ideal coupling without considering the actual split field distributions.

In this study, the electric field distributions of the radiated beams from the SCW splitter were evaluated using a newly developed code. The quasi-optical mirrors were designed for one split beam to be focused and coupled into an adjacent SCW while matching its phase distribution to a Gaussian beam with a waist point at the adjacent SCW input aperture. The fields reflected by the mirrors were evaluated using the in-house extended Kirchhoff integral code, which was developed to incorporate curved mirror boundary conditions, surpassing conventional planar-aperture limits. Finally, resonant and non-resonant FADIS outputs were evaluated by superposing the electric fields of split beams with and without coupling to the 2nd SCW for each round trip of the resonant ring.

Proper waveguide length (L) and incident angle (θ) are searched to achieve the sharpest resonant property with minimum loss in the dual-frequency use. A primary figure of merit (FOM) was defined as the matching coefficient between the output SCW aperture distribution and the theoretically derived ideal aperture distribution for specific L and θ. One split beam could be nearly a Gaussian beam whereas the secondary split beam was significantly deformed from a Gaussian beam when maximizing the matching coefficient. Therefore, a second FOM was introduced to search L and θ that provide highly symmetric split beams with fairly high matching coefficients. By referencing these two metrics, four (L-θ) candidates were selected. The optimal parameters for the dual-frequency (110/138 GHz) FADIS are finally proposed based on and the beam propagation analysis and the resulting resonant properties.

[1] H.Idei, et all.: J. Infrared Milli Terahz Waves 36 (2015) 662.