Plasma boundary reconstruction of ITER plasmas using physics-informed neural networks

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Equilibrium control in fusion devices relies on real-time plasma boundary reconstruction to feedback shape descriptors. One approach solves the MHD plasma equilibrium online during discharge using codes such as LIUQE [1] and PEFIT [2]. These algorithms achieve their best accuracy in iterative mode, which is suitable offline. In real time, they run in streaming mode, computing the current equilibrium in a single iteration from the previous one, meeting sampling times of 500 μs–1 ms while maintaining good accuracy.
Alternative methods rely on simplified plasma representations—parametrized current distributions or local poloidal-flux approximations—used only to compute the Last Closed Flux Surface (LCFS), whereas streamed equilibrium codes reconstruct the full poloidal flux map. Streamed equilibrium codes are the current choice for ITER's planned real time plasma boundary reconstruction, while JET has long used local approximations (XLOC [3]), and JT 60SA currently adopts the Cauchy Condition Surface method, which imposes Cauchy conditions on a surface inside the plasma region [4].
CARONTE is a ML-based algorithm based on a single physics-informed Extreme Learning Machine recently proposed in [5]. Similarly to existing simplified plasma boundary reconstruction methods, such as XLOC and CCS, it computes the LCFS. Thanks to the network simplicity and to the associated algorithm low computational cost, the ELM used by CARONTE is suitable for online training, and therefore adapts in real-time to different plasma scenarios. CARONTE proved to be comparable or even outperform approaches such as XLOC, hence can be considered as alternative for real-time redundancy at ITER.
This paper aims at assessing CARONTE performance against ITER scenarios, by including the effect of the envisaged noise on the magnetic measurements. The reconstruction accuracy will be evaluated using the CREATE-NL equilibria as ground-truth.

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Improved coil decoupling at TCV through the ALIGN algorithm

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

In tokamak magnetic control systems, reference traces for the coil currents are typically provided either by the shot preparation workflow or by outer control loops that regulate plasma current, position, and/or shape. In this context, accurate coil current decoupling is a fundamental component of tokamak magnetic control, enabling precise regulation of plasma current, position, and shape. Traditionally, decoupling is achieved using the mutual inductance matrix of the active coils, which provides ideal performance at steady state. However, during transients or highly dynamic plasma phases, interactions with passive structures such as the vacuum vessel can degrade decoupling effectiveness, limiting control performance.

In this study, we apply the ALIGN algorithm (Edmunds and Kouvaritakis, 1979) to compute optimal, real-valued decoupling matrices at specified frequencies based on the system's transfer function. The identification of real-valued matrices simplifies controller implementation and often requires minimal modification to existing tokamak control architectures. Using the vacuum circuits and vessel model generated by the FGE code of the MEQ suite (C. Heiss, arXiv:2512.06847; Carpanese, PhD thesis), this algorithm was used to obtain decoupling matrices at given frequencies of interest. The method was validated through simulations and dedicated experiments performed on the TCV tokamak, including a configuration in which multiple matrices were combined via band-pass filters to address different dynamical regimes.

Experimental results show a measurable improvement in coil-to-coil coupling suppression across relevant frequency ranges, demonstrating the practical benefits of frequency-dependent decoupling for tokamak control. These findings highlight a promising pathway toward enhanced dynamic control, and future work will extend this approach to decoupling of the shape control channels, further supporting robust and precise plasma operations in advanced tokamak scenarios.

US Contributions to the ITER Disruption Mitigation System Toward Reliable Pellet Launching and In-Flight Stabilization

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

Shattered pellet injection (SPI) is the chosen technology for the disruption mitigation system (DMS) of ITER. SPI utilizes cryogenic cooling to desublimate low pressure (

Error Field Correction Strategies applied to the DEMO LAR Tokamak

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Manufacturing and assembly deviations in the Magnetic System (MS) of large tokamaks inevitably generate non-axisymmetric magnetic perturbations, commonly referred to as Error Fields (EFs), which can significantly impact plasma stability and performance. For DEMO-class devices, the definition of acceptable tolerances for the Toroidal Field, Poloidal Field and Central Solenoid coils represents a critical trade-off between engineering feasibility, cost and the effectiveness of Error Field Correction Coils (EFCCs). In this work, a comprehensive assessment of MS misalignments EFs and correction capabilities is performed for the EU-DEMO, Low Aspect Ratio (LAR) baseline where the tolerance on the coil's alignment has been relaxed w.r.t. ITER (i.e. from millimiters to centimeters). A suitable method based on a filament–stick representation of the coils is employed to model MS misalignments. The method presented, valid for sufficiently small misalignments, computes a first-order Taylor expansion of the related magnetic field deviations, enabling large-scale Monte Carlo analyses involving several million random realizations of MS misalignments. This approach is used to derive statistical distributions and identify worst-case scenarios. The impact of EFs is quantified using two reference metrics: the Three-Mode Error Index, representing the "vacuum response", and the Overlap Field, which accounts for "plasma response" through coupling with the linear MHD stability code MARS-F.
Correction strategies are also investigated by optimizing the EFCC currents under realistic engineering constraints for different coil configurations, including upper–equatorial–lower, upper–lower, and equatorial-only arrays. The study demonstrates the importance of integrating tolerance definition, EF estimation, plasma response modelling, and correction system design within a unified framework. The proposed methodology provides a robust and computationally efficient tool to support the optimization of assembly and manufacturing tolerances of tokamak MSs, as well as EFCC architectures, contributing to a cost-effective and technically viable design for any tokamak device.

Topology measures for the study of the complexity changes in time series related to disruptions

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

On JET with a metallic wall, forthcoming disruptions start leaving a signature in various diagnostic signals, such as e.g. locked mode and plasma current, about ~300 ms before the beginning of the configuration collapse [1]. A significantly increased complexity of these time series non-linear dynamics is revealed by the sudden growth of the embedding dimension of the phase space, profound changes in the recurrence properties and the onset of chaotic behavior [1]. Modifications in the topology of the attractor shape are studied by means of the Veitoris-Rips filtration and by using topological classifying measures such as Betti numbers and barcodes [2]. The evolution of the dynamical systems leads to the creation of attractors with fractal properties. The complexity of the fractal structure is characterized in terms of an extremely simple estimator of the correlation dimension [3]. The dynamical changes involve also interactions across scales, and the rise of complexity can be evaluated with the multiscale entropy [4]. The methods are computationally efficient and they have been tested on a set of JET discharges including representative experiments of the DT campaigns, using the locked mode signal.

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Status and plans of the ITER Pulse Design Simulator

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

The ITER Pulse Design Simulator (PDS) is an open-source software package being developed to support ITER operation by offering the capability to design and validate pulses before operating them on the machine. The ITER PDS uses the IMAS standard and the Muscle3 Persistent Actor Framework to facilitate the coupling of codes for the development of multi-language workflows. Such technical choices are aligned with those of the WEST flight simulator, NEST, the development of which is made in parallel.
The ITER PDS consists of two parts: 1) a pulse editor, providing a user-friendly interface to facilitate the configuration of the pulse; 2) a pulse simulation tool to check the feasibility of the configured scenario and create references for plasma operation. The pulse editor includes a waveform editor, developed in python, to create and manage 1D time-dependent data based on human-readable text-based file format where waveforms are described as tendencies. It also includes a plasma shape editor to configure the desired plasma shape for the key phases of a pulse. This shape editor relies on the execution of the NICE free-boundary equilibrium code used both in static inverse and direct modes, to identify a set of poloidal field coil currents fitting the desired plasma shape. The pulse simulator, in its first version, is based on self-consistent modelling of the plasma equilibrium, calculated with NICE code in static inverse or evolutive direct mode, with plasma transport, calculated with either METIS or TORAX, associated with magnetic controllers developed within the ITER Plasma Control System Simulation Platform (PCSSP). Note that NICE-METIS is the code combination also adopted for the development of the JT-60SA PDS.
The paper will present the status and development plans of the PDS, together with the first results obtained with NICE-METIS and NICE-TORAX, and a comparison to the DINA code for a few scenarios of the ITER Research Plan. The integration of an NTM workflow within the ITER PDS and its first results are also presented. This workflow uses additional models such as TORBEAM as actuator for Electron Cyclotron Heating and Current Drive, associated with PCSSP NTM controllers and the Co-MRE code to simulate the dynamics of magnetic islands. Forthcoming PDS development plans, also presented here, will include the introduction of segments and the integration of the heating and current drive workflow.

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The JET Operation Instruction System and its relevance for Next Gen fusion facilities

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

Like other complex, high stakes scientific projects, the operation of JET could only be a success if personnel and investment risks were carefully managed, without un-necessarily restricting the machine capabilities while ensuring the staff feels part of the process.
JET was not unique in having a set of Instructions on how to operate the device and ensuring the personnel and investment protection. However, it's important to recognize how the JET Ops Instructions (JOI) system has been such an integral part of the successful exploitation of the largest, and D-T capable, tokamak. Having a clear system governing what can, cannot or must be done gave the JET project the necessary solid basis to manage tokamak operations and engineering activities.
The paper will give an overview of the JOI system, covering both personnel safety and investment protection aspects, its management processes, and how the system was framed around the concepts of "why", "how to" rather than just "thou shall not", thus providing constructive support to the operational and scientific staff. Another important aspect of the JOI system was the continuous review process. The Instructions were living documents, with revisions initiated either by an emergence of a new threat, for example the expected failure of the IR protection cameras with high D-T neutron flux, or by modifications or upgrades to a tokamak sub-system, like the increase in NBI power and pulse duration. As part of the management process, exceptions to JOIs were allowed for limited scope, specific experiments, for example to exploit safety margins or measurement redundancy or to accept the increased risk of machine damage against the very high scientific value of an experiment, as it was the case for the Runaway Electron studies in direct support of ITER.
The paper will, also, describe how JOIs were implemented throughout the experiment preparation and execution process, including a unique pre-pulse digital validation system which was extremely advanced for its time.
The combination of clear documentation and knowledge transfer, a rigorous update process, staff training, and transparent implementation has made the JOIs an extremely powerful system, suitable to ensure the engineering and scientific success of such a high-risk facility through its dynamical, evolving lifetime. The JOIs, and their management processes, provide one of the most relevant examples of safety and investment protection systems from which we can learn for Next Gen fusion facilities.

Physics Informed Neural Network for Real-Time Toroidal Reconstruction of Magnetic Perturbation in RFX-mod2

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

In preparation for the forthcoming operation of RFX-mod2 [1, 2] in a reversed-field pinch configuration, a major upgrade of the experiment's real-time control system involves transitioning from a simplified cylindrical approximation to a more accurate toroidal description of the plasma equilibrium and magnetic perturbations. While a full toroidal reconstruction framework is already available [3], its computational cost currently limits its use to post-shot analysis, preventing direct integration into the real-time control loop and motivating the development of fast surrogate models.

In this work, we propose a surrogate modeling strategy based on a modular topology, based on the composition of several neural-network models designed to enable real-time toroidal reconstruction of the entire radial profile of the magnetic perturbations in RFX-mod2, matching the edge field measurements.
The approach follows a two-step pipeline architecture in which, at first, the equilibrium magnetic configuration is approximated by a dedicated neural network, while additional networks are introduced to model the non-axisymmetric perturbations. Each perturbation model is associated with a specific toroidal mode number n, with initial emphasis on the most relevant modes for RFX-mod2 operation n=7,8,9. For each toroidal mode, the model accounts for multiple coupled poloidal harmonics m=0,1, including the corresponding sidebands m=−1,2, which naturally arise from toroidal
coupling effects.

The surrogate models are trained within a physics-informed neural network (PINN) framework [4], exploiting the knowledge of the governing equations of the underlying physical model. In particular, the magnetic perturbations are described by a parametric system of coupled ordinary differential equations in the radial coordinate. These equations are incorporated into the training process through additional physics-based loss terms, acting as regularizers that constrain the solution space and improve generalization capabilities, especially in regimes that are weakly represented in the training data. The presence of multiple coupled modes and parameters poses additional challenges, which are addressed through tailored network architectures and training strategies.

The proposed approach achieves reasonable reconstruction accuracy while significantly reducing computational complexity compared to the full physics-based solver. Moreover, the use of neural networks enables further optimization for low-latency execution and hardware deployment, making the method particularly suitable for implementation on FPGA platforms. These results demonstrate the potential of physics-informed surrogate models as a viable path toward real-time toroidal reconstruction and control of magnetic perturbations in RFX-mod2.

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First steps towards the fast protection of the ICRH system for DTT

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

HotRIO is a fast, low-latency architecture with a proprietary communication protocol based on fiber GbE (HotRIO protocol) which is designed for real-time diagnostics and machine protection systems. It has a portfolio of electronic boards that handle specific acquisition tasks, along with processing units that perform real-time computations and data aggregation, strongly relying on Field Programmable Gate Array (FPGA) technology. The focus of this work is the exploitation of such FPGA capabilities to enable the fast protection of the Ion Cyclotron Resonance Heating (ICRH) system of the Divertor Tokamak Test facility (DTT). The ICRH system is part of the auxiliary heating of DTT, required to reach a DEMO-relevant condition of the edge plasma. Its first two radiofrequency generators of 1.2 MW, i.e., the transmitters, are under procurement along with all necessary equipment to set up an ICRH testbed. The protection of such equipment is of paramount importance and typically requires detecting a dangerous situation and shutoff the transmitters in a few μs. An FPGA is a device capable of realizing written code as a digital circuit, allowing operations at very high frequencies and making this technology a viable alternative to legacy ICRH machine protection systems based on analogue and digital electronics, which feature limited flexibility and re-configurability. In this work we focus on the local machine fast protection architecture, which will be a part of a distributed machine protection system coordinated by the central machine protection of DTT. The current architecture consists of electronics boards equipped with ADCs connected to the field and interfaced with an on-board micro-controller (MCU) and FPGA. The MCU and FPGA collect and preprocess the acquired data before transmitting it through HotRIO protocol to a central board in charge of aggregating all information. The central board executes fast protection functions via its multiple I/Os and implements a state machine based on robust protection logic. This architecture has been validated through a hardware-in-the-loop methodology using a STM32 board.

Overview of the DTT ECH&CD Control System Architecture

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

The Divertor Tokamak Test (DTT) facility is currently under construction at ENEA, Frascati (Italy), to contribute to the study, design, and assessment of plasma heat exhaust handling systems. The facility will be equipped with auxiliary heating systems capable of delivering power fluxes to the divertor in an integrated environment under Fusion Plant-relevant conditions, by investigating solutions in terms of materials and techniques at the limits of current technology. Among these systems, the most powerful is the Electron Cyclotron Heating (ECH), which will provide 32 MW of installed power in its final configuration. The ECH system comprises a total of 32 gyrotrons (each rated at 170 GHz, 1 MW, 100 s) divided into four clusters, each containing eight gyrotrons with associated power supplies and auxiliaries. For each cluster, power transmission from the gyrotrons to the plasma is achieved through a quasi-optical evacuated transmission line connecting to eight launching modules installed in the equatorial and upper ports of the vacuum vessel.

A dedicated control system has been designed to ensure maximum operational flexibility during DTT plasma operations while maintaining optimized power availability. The control architecture is organized as a network of Plant System Units (PSUs), which serve as intermediaries between the central Control & Data Acquisition System (CODAS), the Plasma Control System (PCS), and the individual plant components. The PSUs standardize interactions by abstracting low-level implementation details and providing a uniform control interface across all components. Furthermore, the ECH control system implements plant protection functions and enables real-time operation to execute requests from PCS. This work presents an overview of the organization of the ECH control system and its subcomponents.

Simulation of MHD plasma equilibria and evolutions by APEC-M code

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Tokamaks are toroidal, pulsed machines designed for magnetic confinement of thermonuclear plasma. Plasma disruptions during operation can induce high electromagnetic loads on tokamak components, making accurate plasma modeling crucial for design. APEC is a FEM (Finite Element Method) code developed at Fusion for Energy using ANSYS APDL to simulate plasma equilibrium and evolution in magnetic confinement fusion machines. The main difference with other codes, which solve completely axisymmetric problems, is that APEC allows direct coupling of the axisymmetric plasma equilibrium equations with the full 3D eddy currents equations, describing the three-dimensional passive structures surrounding the plasma. While leveraging ANSYS features like geometry setup, meshing, A, V-A nodal formulation elements and post-processing, APEC's dependence on ANSYS introduces performance issues, including iteration restarts and repeated matrix factorization, limiting scalability for large models. Additionally, reliance on costly
HPC licenses restricts its usability with large models. This work aims to address these challenges by developing an independent, free, scalable, and customizable code, called APEC-M, to replicate APEC's algorithm and overcome its limitations.

Development of a Propellant Valve Prototype for the ITER Disruption Mitigation System

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

Disruptions pose a severe risk on tokamak type fusion reactors, due to their heat and mechanical loads on the machine's first wall and components well above engineering limits. A few grams of cryogenic pellets are planned to be injected into the torus, which are accelerated by high-pressure gas and then shattered into smaller pieces, so that hundreds of smaller pellet fragments enter the plasma [1]. The device that creates this is called the Shattered Pellet Injector [2], and the entire system is called the ITER Disruption Mitigation System (ITER DMS).

Based on the existing propellant valve of the ITER DMS Support Laboratory in Budapest [3][4], an eddy-current driven Propellant Valve Prototype (PVP) has been developed by the Advanced Valve Technologies Consortium following an ITER tender. The valve must achieve ≤1 ms opening times, satisfy stringent ITER vacuum requirements with 60 bar hydrogen fill, withstand up to 1 MGy radiation dose under 14 MeV neutron irradiation, and remain unaffected by ambient magnetic fields up to ~400 mT even after 10,000 cycles of operation.

The development was carried out by five parallel, closely interacting work packages. A team established a fully coupled system model which includes all aspects of the valve design such as primary circuit electronics, electromagnetic actuation, moving-mass dynamics, compressible choked flow, breech pressurization and pellet acceleration to meet the specifications [5]. The proposed design was checked in parallel for flow stability, supersonic conditions and pellet loading by a dedicated fluid simulation team. A structural simulation team assessed pressure vessel integrity and cyclic loading. Large effort was dedicated to find the best closing-element design with experimental validation to ensure sealing reliability [6]. The mechanical design team integrated all subsystems into a manufacturable and ITER-compliant mechanical design.[SJ1.1]

This contribution provides a system-level overview of the design [SJ2.1]and demonstrates that the proposed PVP satisfies ITER's stringent requirements on materials, radiation hardness, magnetic field robustness, vacuum compatibility, pressure vessel integrity, high-voltage operation, and achieves reproducible performance with high confidence.

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Preliminary results of integrated assessments of the ITER Plasma Control System

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

The ITER Plasma Control System (PCS) will make simultaneous use, during pulsed operation of the machine, of multiple function blocks to control and monitor all the relevant quantities. Considering the complexity of ITER and the costs of its operation, a system engineering and model-based approach for the PCS design is fundamental. The design of each function is considered complete when they are assessed, i.e. it is proved that they satisfy the associated set of performance requirements.
Each function is first assessed independently. However, this is not sufficient to demonstrate that the overall PCS performs correctly when several functions are integrated. This process of verification is defined as integrated assessment.
To better manage a large number of assessments, they do not necessarily cover an entire pulse. Instead, they focus only on a specific phase of the pulse that is particularly challenging for the functional blocks of interest.
The process of building the integrated assessments follows a bottom-up approach, progressively combining individually assessed functional blocks to increase the level of integration. A list of integrated assessments has been identified and prepared. It is reasonable to think that as the integration of functional blocks increases, the number of required assessments is reduced.
Each assessment, independently of its level of integration, can make use of models to simulate physics phenomena. Models are typically reduced in fidelity to just capture all the aspects of the assessments. In the case of integrated assessments, the models must simultaneously represent all the involved phenomena interconnected with each other. For the integrated assessments of the PCS for the first Start of Research Operation (SRO) phase of ITER operation in the 2024 re-baseline, it has been decided to use ASTRA/Fenix as main plasma magnetic and kinetic modeling code.
In this paper, we present results from the first set of integrated assessments which have been performed, providing a preliminary flavour of the complexity and challenging nature of the task faced in order to ensure a complete and robust PCS design. In particular, the paper shows results of assessments of magnetic and density controllers during the plasma ramp-up phase of a low current, hydrogen L-mode SRO scenario. All the necessary steps taken to build the integrated assessments from the individual assessments are described. It is important to note that, considering the amount of functionality involved, this case is considered as a complex and essential backbone for all other integrated assessments.

#12993

Tokamak plasma initiation: a systematic, predictive approach to scenario design demonstrated on MAST-Upgrade

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

The initiation (plasma breakdown and burn-through) phase of a tokamak discharge is a crucial element of scenario development. Present experimental devices can use a process of shot-to-shot iteration to achieve successful initiation, but future large tokamaks will be both more difficult to start and have significantly fewer shots available, motivating predictive simulation for design and optimisation of the initiation scenario. This contribution describes a workflow for achieving successful initiation on MAST-U with efficient use of plasma shots, and exploits it to map out the operating space for successful initiation.

First, the electromagnetic model of the passive conducting structures is calibrated by fitting its conductivity distribution to best match magnetic diagnostic data from vacuum shots. These model improvements are essential to accurately predicting the magnetic field structure during the highly dynamic initiation phase.

Second, the poloidal field coil current trajectories are optimised to produce a desired field evolution using a new tool called yfactory. This seeks to produce a requested loop voltage, vertical field, field curvature, etc. as functions of time, whilst obeying engineering limits such as those on power supply voltage and current.

Third, the scenario is tested in DYON, which solves for the plasma and impurity particle and energy balances to predict plasma breakdown, burn-through, and current ramp-up. An iteration between DYON and yfactory then converges upon a successful scenario which is reasonably self-consistent, for example in terms of radial force balance.

Finally, this scenario is implemented on MAST-U and the experimental measurements compared to the yfactory and DYON predictions; we demonstrate good agreement between the designed scenario and its manifestation. We showcase the unique capability to not only predict boundaries in operating space which are more subtle than the standard dependence on prefill pressure and peak loop voltage, but to systematically design and implement initiation scenarios which avoid them.

This work has been funded by the EPSRC Energy Programme [grant number EP/W006839/1]

#13000

Feedback Control of Magnetic Topology for Alternative Divertor Configurations on ASDEX Upgrade

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

During the 2024-25 ASDEX Upgrade experimental campaign, several magnetic configurations of Alternative Divertor Configurations (ADCs), such as
snow flakes (SF), X divertors (XD) or super compact radiative divertors (sCRD),
were explored. These configurations are characterized by a poloidal field topology that enhances flux expansion and spreads the exhaust power over the divertor target plates.
While the ADC magnetic topology showed promise in terms of heat-load
mitigation, it demonstrated limited robustness, particularly with regard to variations in poloidal beta (βpol), which complicates the investigation of various
ADCs under different core plasma parameters. This necessitates feedback control stabilizing the ADC topology.
To choose the appropriate feedback signal, we investigated the behavior
of the magnetic field-line geometry using principal component analysis (PCA).
This analysis shows that each ADC configuration is located in a well-defined region of a low dimensional space. Then, as a first step, we implemented the
dimensionality-reduction algorithm for real-time execution on the Discharge
Control System (DCS), exploiting the JANET++ magnetic equilibrium solver.
As the next step, we implemented the controller, which was validated offline
using the Fenix Flight Simulator and subsequently integrated for the first plasma
experiments.
This paper reports the methodology and the corresponding experimental
results on ASDEX Upgrade.

Progress on ICH control and exception handling in ITER PCS design

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

In the new ITER baseline, tungsten replaces beryllium as the first-wall material, improving resilience to disruptions but increasing the risk of core W accumulation, thus significantly degrading plasma performance. To control W accumulation, auxiliary heating is significantly upgraded: ECH power increased from 20 MW to 40 MW and new 10 MW of ICH implemented in the first operational phase, Start of Research Operation (SRO). In particular, ICH-induced temperature anisotropy of resonant species directly affects W core transport, and also supports fuel removal and wall conditioning together with ECH.

The ITER Plasma Control System (PCS), which manages all aspects of the ITER plant to achieve the desired pulse scenario, must accommodate the integration of the new ICH module supporting plasma heating, ion cyclotron wall conditioning (ICWC), and sawtooth control. Owing to the system engineering approach adopted from the outset of PCS development [1]—ensuring design coherence and a streamlined process for design, verification, and validation—the ICH control module is integrated seamlessly into the existing PCS framework. The established design procedure is readily applied to ICH plant modeling, control, and exception handling, consistent with the previously developed modules (e.g. ECH module [2]).

This work presents an overview of the ICH module design within the ITER PCS, including simplified control-oriented plant model and exception handling, and illustrates the results through representative use cases executed on the ITER plasma control simulation platform (PCSSP), such as ICH power control, gas-coupling control, and associated fault management.

Simulating PID Control of the Total Radiation in the Perspective of Next Generation Tokamaks

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

As reactor dimensions increase, the precise measurement and control of the plasma total emitted radiation becomes more important. This quantity is indeed essential to keep the power loads in the divertor within reasonable limits, not exceeding the mechanical properties of present day materials. To this end, various radiative scenarios have been recently proposed and tested. The most credited solutions require detachment of the divertor and involve increasing the plasma density and injecting light impurities to induce high radiation emission, thereby reducing the heat load on the plasma facing components. However careful control of these plasma conditions is not easy, because the injected impurities must not contaminate the core, with the consequent dilution and excessive cooling of the reactive plasma center. In this work, a first modelling framework for plasma radiation and density is presented to simulate the control of highly radiative scenarios using a set of PID controllers. The model has been developed in Simulink with the aim of studying and testing different plasma and controller dynamics. Even if preliminary, the designed controllers allow identifying the main delicate aspects of such feedback schemes in terms of measurement requirements, parameter tuning and characteristics of the scenarios. A first training and validation of the simulated model is performed using experimental data from both JET and TCV.

Design of the ITER DMS Sequence Manager

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

In modern experimental tokamaks and future fusion devices, the energy content of a fusion plasma is at a level where a disruption can lead to large thermal and electromagnetic loads being deposited on the tokamak wall, potentially causing significant damage. It is therefore important to prevent or avoid disruptions whenever possible. Failing this, disruption mitigation is required to reduce the loads on the first wall to tolerable levels that minimise the risk of damage. In recent years Shattered Pellet Injectors (SPI) have been successfully used to mitigate disruptions. ITER will have a set of 27 SPI around the torus for its mitigation strategy. A key challenge is to drive an appropriate subset of these injectors to successfully mitigate a disruption, while maintaining a sufficient set of SPI as backups should others become unavailable.

The Disruption Mitigation System (DMS) Sequence Manager is part of the machine protection system. Its purpose is to determine in real-time, according to a pre-defined set of rules, which SPI are required to fire in order to mitigate a detected disruption or to extinguish a runaway electron beam. This ordering of SPI triggers is called a sequence. To determine what the sequence of SPI to trigger should be, the rules to select an appropriate sequence depend on the plasma state and phase of the disruption at trigger time, ensuring the most adequate selection of SPI under the current conditions. The system is also designed to cope with exceptional circumstances such as SPI unavailability, trigger failures, and missing or unreliable inputs. In these cases, the sequence is adapted or an alternative sequence is selected based on the type of failure that occurred, ensuring a viable mitigation strategy.

This paper presents the design of the DMS Sequence Manager. It details the logic implemented in the system, from the definition of the requirements to the development of a MATLAB/Simulink model suitable for deployment on a real-time target. Finally, it also presents some simulation examples from the suite of test cases developed.

Improved electron density profile estimation on ASDEX Upgrade for pellet-fuelled discharges

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

Real-time (RT) monitoring of the electron density profile is a key requirement for plasma control in high-performance tokamak operation, particularly during pellet-fuelled H-mode discharges where rapid and localized density perturbations occur during the ablation of the pellet. These perturbations may corrupt RT interferometry signals, causing fringe jumps. The RT estimation of the density profile can be performed using RAPDENS, which combines a control-oriented particle balance model with an Extended Kalman Filter. Heterogeneous diagnostic information can be integrated within this framework into a consistent, time-resolved estimate of the radial electron density profile.

We apply the improved RAPDENS code to ASDEX Upgrade, explicitly accounting for the separatrix electron density [1] to improve RT estimation. The framework integrates line-integrated interferometry measurements with separatrix density inferred from divertor pressure gauges. If available, local RT Thomson scattering measurements can also be incorporated, as shown in [2]. Following [1], the separatrix density is treated as a dynamic boundary condition, thereby improving the edge and near-separatrix representations. Including the estimated injected pellet size [3] as a predictive input enhances the system's ability to forecast core profile evolution during missed or partial pellet ablation.

Application to ASDEX Upgrade H-mode plasmas demonstrates an improved estimation of the electron density profile. The resulting profiles provide continuous monitoring even in the presence of diagnostic failures or limited spatial diagnostics coverage. These capabilities support RT reconstruction of density evolution and form a basis for future feedback control applications and integrated state estimation schemes in various tokamak scenarios.

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Plasma-AURA: A conversational AI tool for tokamak operational data analysis at WEST

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

The analysis of tokamak operational data between pulses requires rapid event identification and data access to guide subsequent experiments. Traditional retrieval methods based on SQL queries and specialized scripts demand significant expertise, limiting accessibility during the critical inter-pulse window. Recent progress in Multimodal Large Language Models (MLLMs) and agentic workflows has opened new possibilities for fusion data analysis. However, deploying such systems for interactive, multi-signal exploration during operations presents challenges that have not been fully addressed.
We present Plasma-AURA (Agentic Unified Retrieval and Analysis), an AI conversational tool that enables natural language interaction with tokamak operational data. The tool supports tasks such as signal retrieval, post-pulse event identification, and cross-shot comparison during experimental sessions.

The system employs a code-generation architecture where an MLLM interprets user queries, generates executable Python code, and reasons over computed results.
This approach overcomes tokenization limitations encountered when processing long, high-frequency diagnostic signals while maintaining numerical precision.
Plasma-AURA has been deployed as a question-answering assistant using Mistral-Small 3.2 as the core model. Following a proof of concept on the C7(2023) campaign (1,331 shots, 31 operational signals identified by Session Leaders), the system was scaled to a larger WEST archive (C7-C11, 2023-2025) and deployed in the control room during the C12 (Nov-2025) campaign. Moreover, the data ingestion is fully automated and triggered after each pulse. The infrastructure includes data ingestion from the WEST PostgreSQL database and IMAS (HDF5) sources, vLLM serving on two A40 GPUs, and MLflow-based conversation logging for user feedback collection.
During C12 deployment, operators used the system for inter-pulse analysis with diverse objectives, including comparison of diagnostic signals across shots, evaluation of electron temperature during plasma current flat-top phases, and automatic generation of diagnostic plots. This operational use provided valuable feedback on control-room requirements. User feedback highlights the value of the conversational interface while revealing the importance of encoding operational context, such as distinguishing stable plateau values from raw signal transients. The experience also identified requirements for physics aware data validation and the integration of session-level metadata to support operational queries.
Ongoing work focuses on integrating fusion domain knowledge, incorporating semi-automatic event classification from anomaly detection models, and refinement based on collected operator feedback.
This work has been carried out within the framework of the EUROfusion Consortium
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#13088

Post DTE2 Tritium clean up analysis and recommendations for clean-up post DTE3

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

After DTE2, JET had sufficient neutron budget remaining to allow for a DTE3 campaign (a period of D-T fuelled experiments and a clean-up phase). To support the DTE3 preparations, a review was performed to clarify the requirements of the DTE3 clean-up based on the available DTE3 scope and timescale. Based on this information the timescale for DTE3 clean-up was estimated. This showed that clean-up could be performed with a short non-tritium operational period available before the planned operational closure of JET and its planned decommissioning.

Performing DTE3 required a commitment to clean-up the machine after D-T operations to recover as much tritium as possible and to leave the machine in a non-tritium operational state. This paper summarises the assessments made to define these DTE3 clean-up requirements. This involved reviewing the DTE2 clean-up in terms of efficacy, neutron budget and timescale and suggesting alterations to the DTE2 clean-up procedure to ensure it could be done in the time available. It was also checked that all of the required machine systems would be available to support the DTE3 clean-up.

The proposed clean-up programme for DTE3 was similar in concept to the DTE2 clean-up, starting with machine baking and using ICWC to access loosely bound tritium in vessel followed by a period of operation dedicated to clean-up only pulses. The aim was to increase the pulse repetition rate, target specific areas where tritium was known to reside and shorten the clean-up duration. The machine systems and diagnostics to be used to monitor the clean-up were also reviewed. The main diagnostic proposed to monitor the clean-up was the KM7/KN1 neutron diagnostic ratio approach, the simplest of the neutron measurement methodologies for determining low T fuel ratios.

During tritium operation, all hydrogenic gases are collected and processed through the AGHS system at JET. In contrast to the DTE2 clean-up, there was less time available for processing this gas, and appropriate gas storage was the main issue identified. The major issues identified by the post tritium clean-up review are highlighted.

DTE3 clean-up took six weeks to complete including He calibration pulses, LIDS-QMS measurements and included 5 days of campaign pulses. A clean-up ratio of 0.02 vol % T/D was obtained. This compares to a clean-up ratio of 0.08

Towards AI-based surrogate models for pulse design

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

Magnetic-confinement fusion experiments are expensive, and machine time is limited [1]. Large facilities require substantial infrastructure, specialized operation teams, and dedicated diagnostics, while experimental campaigns run under formal proposal selection and restricted pulse allocation. As a result, many potentially interesting scenarios may remain unexplored. This creates a strong incentive to evaluate candidate discharges before they are executed.

This work proposes the development of AI-based tools that can estimate the likely evolution of a discharge before its execution. Using information available in advance, such as heating settings, magnetic configuration, gas injection, wall conditioning, and other programmed inputs, the developed tools aim to predict plasma behavior. In practical terms, this would help researchers compare candidate pulses in silico, identify promising operating windows, and support pulse design before committing machine time. The objective is not to deliver a full digital twin, but to build a fast and useful predictive assistant for experimental planning.

The present proof of concept is grounded in TJ-II, where a curated database for statistical modelling is already available and where more than 50,000 experiments provide a valuable basis for data-driven studies [2]. Previous AI-based work in TJ-II has explored plasma-density estimation and forecasting of impurity-related signals from experimental data [3, 4]. These studies represent important first steps, although their predictive scope remains limited. They nevertheless support the feasibility of progressing towards a practical AI-assisted surrogate models for fusion experiments.

Although the work completed so far has been carried out in TJ-II, the broader goal is to extend and test the methodology with data from W7-X, JET, and potentially other devices. In the longer term, the aim is to develop transferable models that may also prove useful for future machines lacking extensive historical databases, such as ITER.

Toward ITER Operation: Recipes for Minimum Threshold Power H-Mode Access Without ELMs in KSTAR

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Recent KSTAR experiments [1,2] have revealed critical insights into the L-H transition power threshold. Systematic studies confirmed that non-resonant magnetic fields have no measurable impact, while resonant low-n fields significantly affect the threshold. Furthermore, divertor material plays a decisive role: the transition threshold increases with tungsten divertors, but boronization effectively suppresses tungsten-related impurities, lowering the threshold. These findings suggest that optimized wall conditioning can minimize the required power for H-mode access.
KSTAR has demonstrated machine learning (ML)-based suppression of nearly all ELM crashes, including initial ELMs, without raising the H-mode threshold power [3]. The integration of ML into the plasma control system enables real-time detection and prediction of L-H transitions and ELM events, providing a pathway to fully automated control.
Taken together, these results indicate that KSTAR can propose a practical recipe for ITER: achieving H-mode access at the minimum threshold power while suppressing ELMs. This recipe, combining optimized wall conditioning and ML-driven control, offers essential guidance for ITER's stable, economic, and long-term steady-state operation.

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#13129

Validation of ITER PCS-Based PF Coil Control in KSTAR: From Simulation to Plasma Startup

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

The KSTAR Plasma Control System (PCS) [1-3], introduced as a branch of the DIII-D PCS, has been successfully operating since 2008. However, operating next-generation fusion devices for Korea's demonstration reactor requires a decentralized and flexible PCS capable of efficiently handling numerous parallel algorithms for long-term maintenance. To address this, KSTAR is collaborating with ITER to incrementally introduce the ITER PCS [4, 5] to KSTAR and validate it through experiments. In our previous work [6], we implemented the basic functionalities of the ITER Real-Time Framework (RTF), integrated it with KSTAR infrastructure, and successfully performed a dummy load test for poloidal field (PF) coils.

In this study, we conducted three preparatory tasks for validating the operational performance of the ITER PCS through PF coil real load testing and plasma startup during the 2025 and 2026 KSTAR campaigns. First, we extended the RTF application to the KSTAR tokamak environment, developed the PCS Provisioning System (PPS) to operate PCS configurations created by ITER Pulse Schedule Preparation System (PSPS), integrated PPS with the KSTAR Pulse Automation and Scheduling System (PASS), and established interfaces connecting the ITER PCS with KSTAR fueling devices and Fast Interlock System (FIS). Second, we developed a SIMULINK-based simulator to compare the control performance of the existing KSTAR PCS PF controller and the ITER PCS PF controller (KCURR). The simulator includes power supply models, a state-space model for KSTAR, and diagnostic and controller blocks with noise signal models. Since KCURR does not include signal pre-processing, we added a low-pass filter before KCURR to ensure the controller does not act on high-frequency noise content. Third, we designed and executed PF coil real load tests and plasma startup experiments using both PCS under identical scenarios. For plasma startup, prefill gas fueling was applied, and modeled PF coil current waveforms were applied after startup to attempt plasma current ramp-up.

Simulation results confirmed comparable performance of both controllers after adding the low-pass filter. In PF real load tests with DC commands, the ITER PCS showed a time delay of approximately 1.5 ms, comparable to the existing KSTAR PCS. Feedback control and stable ramp-down during PF overcurrent and network disconnection scenarios operated properly. Plasma startup experiments successfully achieved prefill gas fueling, plasma startup, and plasma current ramp-up over several hundred milliseconds,

validating the operational performance of the ITER PCS in KSTAR.

#13159

Assessment of Linearized Plasma Response Models and Vertical Instability Growth Rates

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Linearized response models are widely and successfully applied to describe the plasma behaviour in the neighbourhood of an equilibrium configuration. Due to limited set of degrees of freedom involved, they are ideal for the real-time control of plasma current and shape. They are also essential for analyzing and controlling the vertical instability of elongated plasmas by computing key quantities such as the growth rate. While their effectiveness and reliability is demonstrated by several experimental tests, there are cases where experimental verification is not possible, either because these tests would risk the integrity of the device or because predictions are required during the design phase.

Different linearized models may be based on diverse assumptions leading to possible contradictory results. These discrepancies may be surprisingly large in case of critical values of elongation, poloidal beta, or internal inductance, as well as steep or irregular current density profiles. In such cases, the computation of the unstable plasma vertical evolution is numerically challenging, and results are sensitive to small changes in input data.

Consequently, a robust technique to validate their predictions is required.
This work proposes an efficient method, applied to different tokamak devices, to assess the growth rate obtained by different linearized models, based on two free boundary equilibrium computations. Typically, the validation of the linear model matrices (including modified inductance Lstar and resistance R) is performed by adding small current perturbations DI to each circuit. The resulting differences of output quantities Dy (e.g., sensor measurements, or plasma-wall gaps) between the perturbed and initial free boundary equilibria are compared with the corresponding values dy predicted by the linearized model.

The assessment of the vertical instability growth rate is obtained in a similar way. However, in this case, all the (active and passive) currents are perturbed with variations DI_U corresponding to the unstable mode of the A=-inv(Lstar)*R matrix with eigenvalue gamma_U>0. The variations of the fluxes linked to the circuits DPsi_U are then used as follows:
Comparison: They can be compared to the corresponding predicted values, namely dPsi_U=-R*DI_U/gamma_U.
Virtual energy: They are used to evaluate the virtual energy provided by DI_U'*DPsi_U, and compare this value to the corresponding quadratic form provided by the linearized model DI_U'*inv(Lstar)*DI_U, which is negative if unstable.
Growth rate verification: They are used to get the quantity (-DI_U'*R*DI_U)/(DI_U'*DPsi_U), to be compared with the growth rate gamma_U provided by the linear model, where a positive sign clearly indicates instability.

Upgrading gas flow control on MAST-U for improved density ramp-up and detachment control

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

The Gas Introduction Modules on modern Tokamaks have two main functions: (a) To fuel the plasma; (b) To timely inject appropriate impurities for plasma detachment control. Without accurate control of the gas flow, controlling the density profile and the detachment state reliably is difficult. Uncertainty and variability persist in the experiments. This paper presents the design of a real-time gas flow control system on MAST-U for improved repeatability and robustness. This upgraded gas flow control system reduces engineering uncertainty thus improving extrapolation to future tokamak devices.

Experimental data from JET was analysed and non-linear flow models were identified deterministically and probabilistically. This model predicted gas introduction flow and enabled quantification of flow uncertainty. The limited JET gas introduction sensing capability introduced variability in flow calculation and transient gas pressure distribution.

A prototype bench test control system for the MAST-U gas introduction systems was constructed to simulate the process at vacuum reference pressure. With improved pressure and flow sensing capabilities, a new model was identified. It estimated flow and transient pressure distribution through the gas line with greater certainty. Additionally, the bench test successfully demonstrated accurate flow control using the piezo valves.

Recently installed flow and pressure sensors on a MAST-U gas introduction line collected experimental data from MU05. The identified models were applied to this data, demonstrating flow prediction for control with uncertainty quantification for MAST-U implementation. If proven to be successful during the MU06 campaign, we aim to expand this new gas flow control system to all 52 MAST-U gas lines, improving detachment and fuelling control.

Multiphysics Model for Optimization of the Propellant Valve Prototype design for the ITER Disruption Mitigation System

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

The ITER Disruption Mitigation System (DMS) [1] is a machine-protection system intended to mitigate plasma disruptions by injecting cryogenic pellet fragments into the plasma via Shattered Pellet Injection (SPI) [2]. A critical requirement for the SPI is the reproducible and synchronized arrival of pellet fragments from multiple injectors to ensure minimum toroidally asymmetric radiation. This necessitates a highly predictable and consistent acceleration of the cryogenic pellets, which is governed by the performance of the high-pressure propellant valves, namely the expelled gas amount reproducibility. The valve actuation is based on eddy current principle: high transient current is driven in a primary coil which induces current in a secondary ring thus repelling each other [3]. The design and operation of these valves present a complex engineering challenge due to the strong coupling between electromagnetic forces, high-speed compressible gas dynamics, and mechanical response, all while operating under significant external magnetic fields.

This contribution presents a comprehensive, coupled multi-physics model developed for finding the optimum design parameters of the ITER DMS Propellant Valve Prototype [4]. The simulation framework integrates an equivalent circuit network for electromagnetics, a thermodynamic model and force calculation for gas dynamics, and rigid body dynamics for the valve and pellet that also accounts for external magnetically induced torques and friction. The simulation tool was validated against experimental data from the ITER DMS Support Laboratory [5] and CFD simulations, showing excellent agreement in valve piston displacement and pressure drop characteristics.

The model was utilized to perform a robust optimization of the valve geometry and coil configuration with constraints on closing time and structural integrity [6]. The primary objective was to minimize the variance of the pellet's time of arrival in the torus. The analysis demonstrates that the optimized design achieves reproducible pellet delivery. Overall, the framework advances SPI reliability by minimizing performance uncertainties.

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Overview of Improvements to the Capabilities of the Plasma Control System of the MAST Upgrade Tokamak

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

The plasma control system (PCS) of the MAST Upgrade (MAST-U) tokamak has been in operational use during the first five experimental campaigns. We present an overview of the recent improvements to the capabilities of the control system, key experimental results that were enabled by the improved control capabilities and the recent hardware platform upgrade that paves the way to building on these capabilities for future experimental campaigns.

Versatile and reliable core and divertor shape controllers and a magnetic reconstruction step for gap feedback control have been developed collaboratively and enabled the development of plasma scenarios for the exploitation of conventional and alternative divertor configurations and studies of plasma exhaust power load mitigation. Improvements in control of the shape of the plasma boundary have enabled detailed studies of the effects of shaping on confinement and stability in the core and edge pedestal regions, and investigations of time-dependent behaviour during vertical oscillations around a connected double null topology. MAST-U achieved access to negative triangularity for the first time on a spherical tokamak, which will significantly contribute to studies into stability boundaries and ELM-free operation.

Plasma electron density control further facilitated repeatable plasma scenarios and enabled reactor-relevant studies into locked mode mitigation and L-H power threshold. A collaboratively developed detachment front control was achieved and demonstrated the feasibility of controlling the spatial extent of the detached neutral buffer. Building on the unique character of MAST-U being a double-null device, recently this capability was extended to allow the simultaneous control of core density and the detachment fronts independently in the lower and upper divertor chambers, which opens avenues for studying up-down exhaust symmetries.

To support future capabilities, significant infrastructure upgrades have been undertaken. The PCS hardware and software infrastructure have been upgraded to a 64-bit architecture during the 5th experimental campaign, increasing the number of available processing cores from 4 to 32. The hardware has built-in support for low-latency data transfer over User Datagram Protocol (UDP) based protocols to enable communication over a Real-time Data Network built from commercial, off-the-shelf components. These enhancements will enable the integration of machine learning for shape control, continued exploration of advanced control techniques for density and detachment control and will provide capacity for the growing control capability demands of the MAST-U experimental programme, such as plasma current profile control involving Electron Bernstein Wave (EBW) heating that is scheduled to be installed in the upcoming engineering break.

Real-time pellet fuel flux inventory accounting for advanced density control in ASDEX Upgrade

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

Fuelling a future fusion reactor will require pellets – mm sized bodies produced from frozen hydrogenic fuel. Injection with sufficient speed allows penetration beyond the plasma edge, resulting in efficient particle deposition inside the separatrix. Hence, pellets will become the main actuator for core density and hence burn power control. In ASDEX Upgrade, the plasma density profile in real-time is estimated by RAPDENS, an Extended Kalman Filter algorithm combining available measurements with a predictive model. In the past, it was assumed that the pellet actuator always delivers the requested fuel flux. However, pellets are fragile objects and their arriving mass showing significant variations. Thus, advanced control requires real-time pellet mass monitoring for flux inventory accounting. Our chosen approach is to monitor the strong radiation emitted during pellet ablation in the plasma while applying a dedicated real-time diagnostic that analyses this radiation in real-time. Initially, the diagnostics recognized missed-out pellets and did allow timely substitution.

The present work extends the approach by deriving on the fly the pellet mass from the integrated ablation radiation. For this it becomes essential to establish the relationship between the pellet particles upon arrival and the total ablation radiation A. We make the ansatz $A = C(t)\, F(X)\, \Delta N_e$, with ΔN_e the pellet induced electron inventory increase in the plasma calculated from offline full density profile reconstruction by Integrated Data Analysis. C(t) considers a potential detection system sensitivity degradation over time determined from regularly repeated injections with fixed parameters. F(X) is a scaling expressing the dependence of A on pellet parameters and the plasma state characterised by e.g. toroidal magnetic field, shape and temperature. In order to determine F(X), a dataset is compiled from pellet injection events that cover a wide range of pellet and plasma parameters. Once available, the pellet mass estimator will be implemented in the control system to improve the accounting of the delivered pellet particle flow.

Shattered Pellet Injection Technology Development for Large Tokamak Experiments

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

Plasma disruption represents a major investment risk for tokamak devices close to reactor scale. These events may cause damage to the machine through several effects: mechanical loads from eddy currents resulting from too fast plasma current quench or halo currents resulting from vertical displacement of the plasma, thermal overload of the wall in case of too high local radiation or plasma contact and localized wall melting by uncontrolled runaway electron beam contact. In order to avoid damage the plasma thermal and magnetic energy should be extracted in a controlled way. Injection of cryogenic pellet fragments generated by shattering cryogenic pellets close to the plasma is currently the most promising method and has been tested on present-day devices. However, future large tokamaks need pellet sizes and number of pellets well exceeding the ones used up to now. The Disruption Mitigation System (DMS) design by the ITER Organization has initiated the development of Shattered Pellet Injection (SPI) technology.

To develop and test specific technological solutions for SPI a DMS Support Laboratory has been set up in 2020 at the HUN-REN Centre for Energy Research, Budapest, Hungary. This laboratory intends to test many key elements of SPI technology: formation and launch of protium, deuterium, neon and mixture pellets, propellant gas suppression, pellet flight control, optical pellet diagnostic and shattering. Since the first successful launch of 19 mm diameter pellets in 2021 the laboratory performed more than one thousand 19 mm and 28.5 mm diameter pellet launches and shattering. The concept of a propellant valve, a fast gas shutter and a real-time capable

Optical Pellet Diagnostic (OPD) have been demonstrated, and prototypes for ITER are being developed. The fragment size distribution was measured for various pellets using a laser curtain diagnostic. Reproducible pellet quality and velocity could be achieved. Preventing the propellant gas from entering the plasma before the pellet fragments and controlling pellet trajectory and rotation at the same time proved to be a very difficult task; final solutions are still being developed.

The talk will present the technical solutions and physics understanding of processes developed in the laboratory, together with the achieved results and components for SPI as it will be used in the ITER DMS.

#13269

Advances in Artificial Intelligence for Tokamak Control, Diagnostics and Actuators Resilience

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

Artificial intelligence (AI) is transforming tokamak operation by enabling high-performance plasma control, resilient diagnostics, and data-driven scenario optimization. As fusion devices move toward reactor-scale conditions, reliable operation under limited diagnostic access, harsh environments, and tight stability margins becomes essential. Recent AI-driven developments demonstrate that machine learning can evolve from an analysis tool into a core component of real-time plasma supervision and control.

A key breakthrough is the development of multimodal diagnostic reconstruction frameworks such as Diag2Diag, which exploit cross-diagnostic correlations to enhance measurement fidelity and robustness. At DIII-D, Diag2Diag increased the effective temporal resolution of Thomson Scattering from ~200 Hz to up to 1 MHz, enabling direct observation of pedestal dynamics, individual ELM cycles, and resonant magnetic perturbation (RMP)-induced profile flattening. Crucially, this framework also mitigates diagnostic failure: missing or degraded measurements can be reconstructed from complementary diagnostics with high fidelity, maintaining continuity of critical state variables. Such AI-based virtual diagnostics provide fault tolerance without hardware redundancy, an essential capability for ITER and future power plants where maintenance is constrained.

AI-driven resilience extends beyond reconstruction to cross-device adaptability. In radiation front-based detachment control on DIII-D and KSTAR, machine learning enabled real-time estimation of detachment metrics from alternative diagnostics, allowing robust feedback even when direct measurements differ or are unavailable. This demonstrates that AI can unify heterogeneous diagnostic systems while sustaining control authority during evolving equilibria and partial diagnostic limitations.

In parallel, AI strengthens plasma control architectures. Reinforcement learning, real-time kinetic equilibrium reconstruction, and 3D field optimization have demonstrated improved actuator coordination for tearing mode suppression, gyrotron power regulation, and ELM control in DIII-D and KSTAR. These tools extend operational boundaries in ITER-relevant regimes while maintaining stability.

AI also mitigates actuator failures: during gyrotron loss, rt-TORBEAM re-optimized remaining ECH actuators to maintain target deposition. Its surrogate, TORBEAM-NN, runs 100 times faster, enabling scalable real-time ECH control.

Together, these advances position AI as a foundational technology for reactor-relevant tokamak operation, integrating super-resolution diagnostics, failure-tolerant monitoring, adaptive control, and scenario optimization into a unified, data-driven framework that bridges experimental constraints and physics discovery.

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#13300

Reachability and Controllability Gramian Analysis of Coupled Electromagnetic–Thermal Plasma Profile Dynamics in a Port-Hamiltonian Framework

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

Controlling internal tokamak plasma profiles, in particular poloidal magnetic flux and electron temperature, is important for advanced scenarios with improved confinement and MHD stability. While many model-based controllers have been proposed, it is still unclear how reachable and controllable these coupled profiles are under realistic actuator constraints. Here we use finite-horizon Gramian analysis in a port-Hamiltonian (PH) framework to quantify these limits.

We use finite-horizon reachability and controllability Gramians to quantify actuator authority. The reachability Gramian defines the set of profile states reachable from rest within a time window. The controllability Gramian defines the set of initial perturbations that can be steered back to zero. Eigenvalues show which directions are easy or hard to control, and log-determinants summarize the overall volume of controllable directions. This analysis identifies which profile directions are well or poorly actuated, supporting actuator placement optimization and feasibility checks.

In prior work, we applied this approach to the resistive diffusion equation governing poloidal magnetic flux evolution. Using RAPTOR predictive simulations, we constructed a linear time-varying PH model with transport and input profiles that vary in time and radius, taken from the simulation outputs. We computed finite-horizon reachability and controllability Gramians and evaluated their minimum eigenvalues and log-determinants as scalar summaries. By sweeping the electron cyclotron current drive (ECCD) deposition location, we showed that these metrics provide a quantitative ranking of actuator placement for electromagnetic profile shaping.

The present contribution extends the analysis to coupled electromagnetic and thermal dynamics. Plasma resistivity depends strongly on temperature, while thermal transport depends on magnetic configuration, creating coupling between the two subsystems. In EC-heated tokamaks, current drive and heating are delivered by the same launcher, so ECCD and ECH power cannot be set independently. This shared-input constraint means that electromagnetic and thermal controllability cannot be assessed separately. We formulate a coupled PH model using structure-preserving discretization and compute finite-horizon reachability and controllability Gramians for the combined electromagnetic–thermal state vector. Since thermal transport equilibrates on millisecond timescales while resistive diffusion evolves over hundreds of milliseconds, this timescale separation is expected to appear in the Gramian eigenvalue spectrum, indicating which directions are reachable or controllable within a given horizon. The implementation builds on our existing RAPTOR-based code. Preliminary results show how the shared-input constraint reshapes the controllable and reachable subspaces. These findings will inform actuator placement and profile control design in ITER and future fusion devices.

#13344

Real-time prediction of NTMs onset at TCV

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

We present a model to forecast tearability [1] in TCV, i.e. the probability of triggering an [N]TM within a time horizon, which we assume of the order of a confinement time for a typical TCV plasma.
Tearing Modes and Neoclassical Tearing Modes ([N]TMs) can degrade the confinement performance of the plasma energy and may cause plasma disruptions in Tokamaks, which can lead to machine damage in high-power tokamaks, e.g. ITER [2]. These MHD modes need to be avoided/prevented when possible and stabilized when unavoidable. The onset of these modes depends non-linearly on the plasma internal state, especially on the evolution of the kinetic and current density profiles, as well as on the seeding from other plasma events, such as ELMs or Sawtooth. The triggering conditions of [N]TMs across different plasma regimes, and the influence of various plasma parameters on their evolution, remain partially elusive due to complex non-linear dependencies. Consequently, Machine Learning (ML) models offer a promising approach to unraveling these intricate physical interdependencies.
The model has been trained and validated on multi-diagnostic signals from over 4’000 shots at TCV. The DEFUSE framework [3] has been used to generate and curate the dataset. The algorithm is integrated into the TCV Plasma Control System and it outputs to SAMONE [4], the supervisory control and actuator manager of the TCV PCS, where tearability is taken into account for event handling. As a next step, real-time experimental validation of the model on TCV is planned for upcoming campaigns.

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[2] I. Bandyopadhyay et al, Nucl. Fusion 65 (2025) 103001, Sec. 2.2, https://doi.org/10.1088/1741-4326/ade7a0
[3] A. Pau et al., “A Modern Framework to Support Disruption Studies: The Eurofusion Disruption Database”, 2023 29th IAEA Fusion Energy Conference (FEC 2023), London, United Kingdom, 2023
[4] T. Vu et al., "Integrated Real-Time Supervisory Management for Off-Normal-Event Handling and Feedback Control of Tokamak Plasmas," in IEEE Transactions on Nuclear Science, vol. 68, no. 8, pp. 1855-1861, Aug. 2021, doi: 10.1109/TNS.2021.3084410

Surrogate models for transport modelling in pulse design simulators

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

TORAX is an open-source, differentiable tokamak core transport solver predicting the heat, density and current profiles up to the last closed flux surface. It is based on Python using the JAX framework and is tightly coupled to turbulent transport surrogate models such as QualiKiz and TGLF based neural networks. It has recently been adapted to use the IMAS standard and the MUSCLE3 persistent actor framework to enable modular coupling with other integrated modelling components. The use of MUSCLE3 facilitates the management of the different space and timescales involved in the modelling framework. The goal of this implementation is to make TORAX one of the main components of the ITER Pulse Design Simulator (PDS), currently under development at the ITER organization.

TORAX provides enhanced coupling capabilities, making it easier to build on existing open-source physics models. Its architecture gives flexibility in the choice of the physics models to which TORAX can be coupled, regardless of their language. It can be coupled to an external equilibrium code, heating and current drive sources, plasma edge models for core-edge modelling, controllers or synthetic diagnostics. So far, the IMAS and MUSCLE3 adaptation of TORAX enables its coupling to the NICE free-boundary equilibrium code used in inverse or direct modes via the IMAS ‘equilibrium’ Interface Data Structure (IDS), and to Heating and Current Drive (H&CD) sources via the ‘core_sources’ IDS. It calculates the kinetic profiles and the so-called P’ and FF’ profiles used for self-consistent coupling with the equilibrium.

The present study reports on the status of the TORAX adaptation, together with the first results obtained in the context of its use in the ITER PDS. It focuses on verification and validation against the High-Fidelity Plasma Simulator (HFPS) , the European IMAS version of JINTRAC and the METIS transport solver, based on WEST cases. Self-consistent modelling of electron cyclotron H&CD, coupling TORAX with TORBEAM, applied to simulate scenarios from the first operational phase of the new ITER research plan, is also presented.

#13396

Reduced wall-heating protection during DT operation on JET: management of risks.

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

The JET Iter-like wall (ILW) was protected from overheating during normal operations by the “protection of the iter-like wall” (PIW) system: a number of infra-red (IR) cameras observing various regions of interest (ROI) of the walls and divertor. Should a hotspot be observed above a preset threshold, mitigating action was initiated such as reduction of input heating power and/or a change in plasma configuration to reduce plasma-wall interaction. In the recent Deuterium-Tritium campaigns on JET (DTE2 in 2021 and DTE3 in 2023), the much higher expected neutron flux compared to deuterium-only operation meant that many PIW cameras had to be removed from within the torus hall and only a subset of PIW cameras were retained by routing their optical pathways via fibre optics to areas outside of the neutron protection shielding. This then meant that the levels of protection available for PIW were reduced and administrative methods, based on extensive prior experience, were required to successfully operate high power experiments in DT whilst maintaining acceptably robust levels of machine protection.

In high power pulses, particular hazards to the ILW come from neutral beam injection (NBI) and ion-cyclotron resonance heating (ICRH). Both systems generate highly energetic particles that can cause high localised power densities on the vessel walls and protection tiles, ICRH via sheath effects, and NBI via shine-through and reionisation (where neutrals are ionised in the duct region before reaching the plasma). The NBI system on JET comprised two beamlines with identical injection geometry but separated 180° toroidally. Specific protection tiles were mounted in a carefully determined positions to intercept reionised power from each beamline that would otherwise cause damage to the plasma facing components. PIW cameras monitoring the temperature of the reionisation tiles then protected these tiles against potential damage from overheating.

In DT operations, due to the reduced number of available PIW cameras, only one of the two PIW cameras monitoring the reionisation tiles was retained, with full protection relying on the symmetry of the other beamline combined with administrative requirements based on prior experience. Reduced coverage of other areas of the ILW by

PIW protection also required best-practise techniques for safely operating high power ICRH.

This contribution will discuss the limitations imposed on experiments caused by the necessary reduction in available machine protection and techniques employed to overcome these to achieve full scientific exploitation and results with implications for design of future DT capable machines.

Reduced Disruption Modelling Workflow for the Design of New Components in View of ST40 Upgrade

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Plasma disruptions represent one of the most critical off-normal events in tokamak operation, generating severe electromagnetic and thermal loads that can challenge the integrity of both toroidally continuous structures and segmented plasma-facing components. A reduced disruption modelling workflow has been applied to provide fast, design-oriented load assessment suitable for engineering applications. The approach is based on 2D axisymmetric electromagnetic simulations used to reconstruct the disruption transient, including plasma current decay, induced eddy currents, halo current evolution, and plasma motion. Particular emphasis is placed on extending the applicability of reduced models applied to the vacuum vessel and main in-vessel components, such as divertor sectors, limiters, and plasma-facing components, through equivalent circuit representations and mesh-based electromagnetic coupling techniques. The resulting load datasets are structured to directly support a fast 2D-to-3D electromagnetic-to-structural assessment, enabling efficient transfer of global and local electromagnetic forces distributions into finite-element structural analyses. This integrated framework allows rapid design iteration while preserving conservative load envelopes appropriate for early and intermediate design phases. The reduced modelling predictions are benchmarked against available experimental measurements from ST40 disruption events, including plasma motion, current quench dynamics, and induced current signatures, providing validation of the methodology and increasing confidence in using it to support design efforts. Finally, to demonstrate the robustness of this workflow in supporting the design of both toroidal and non toroidal components for current and future devices, it has been applied to the modelling of a low field side limiter that will be installed on Tokamak Energy Ltd's high field spherical tokamak ST40 as part of the LEAPS (Lithium Evaporations to Advance PFCs in ST40) project.

#13408

Evaluating update paths for the input-output interface of the ASDEX Upgrade Discharge Control System

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

The Discharge Control System (DCS) of ASDEX Upgrade (AUG) requires reliable, real-time capable exchange of information with other subsystems of the experiment. Currently, the main input-output (IO) interface for signal exchange during plasma discharges consists of custom build plug-in cards for the DCS server and fan-out IO front-ends connected via optical fiber. Both analog and digital IO are in use to exchange actuator commands, measurements from diagnostic sensors, as well as status and interlock signals.

For newly developed systems, substituting hardware IO with network based real-time communication is an interesting option. Using network communication, existing fiber-optical point-to-point connections can be reused for critical communication. Where suitable, however, communication can also be moved to the available network infrastructure gaining flexibility. Instead of having to convert between physical quantities, e.g. requested coil currents, and voltages in the ADC/DAC domain, data can be exchanged describing those quantities directly. And in cases where the actuator controller or data acquisition system performs digital signal processing, too, removing the intermediate DA/AD conversion reduces inaccuracies and noise.

First systems to use network communication with DCS during plasma discharges are the currently deployed real-time diagnostics. Also, the renewed control system for the power supplies of the coils, which was successfully tested during the 2024/2025 campaign, switched to network communication.

Currently, a successor to the aging analog gas valve controllers of the gas inlet systems is under development. While the immediate application is the gas valve controller, the platform allows to explore a more general update path for the DCS IO interface, as well. Starting with the gas valve controller as an example application, this contribution will discuss the steps necessary to extend the concept into a flexible, network based IO interface for DCS. It will address the benefits of replacing the current IO hardware with such new IO hardware, discuss additional capabilities, that would make it more than a one for one replacement and show use cases, where such hardware could be deployed beyond DCS.

Physics-based real-time control of [neoclassical] tearing modes with co-MRE and TORBEAM at TCV

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

[Neoclassical] tearing modes ([N]TMs) are magnetohydrodynamic (MHD) instabilities that form magnetic islands at locations of rational safety factor q=m/n, where m and n refer to the poloidal and toroidal mode number, respectively. Once fully developed, they can significantly degrade the energy confinement and, consequently, the fusion gain. They often act as precursors to plasma disruptions and are thus a major concern for the safe and optimal operation of large fusion devices such as ITER [1].

One of the primary actuators for [N]TM control are electron cyclotron (EC) waves, which can provide localized heating and current drive. The corresponding absorption locations and current drive efficiencies can be computed in real-time by beam-tracing codes such as TORBEAM [2]. The impact of the EC on the [N]TM width and frequency can be best estimated by comparing experiments with modelling using the Modified-Rutherford equation (MRE), which is implemented in the comprehensive MRE solver co-MRE [3].

This contribution details the integration of [N]TM and EC observers with co-MRE and TORBEAM in different layers of the generic plasma control system of the tokamak à configuration variable (TCV). In particular we show how the inclusion of these observers is used to inform a supervisory controller on the optimal EC power for [N]TM prevention and stabilisation, as well as on the EC beam–to–[N]TM alignment. We discuss how this improves [N]TM control compared to standard [N]TM control schemes at TCV, and its relevance for the design of the ITER control system [4].

[1] I Bandyopadhyay et al Nucl. Fusion 65 103001 (2025)
[2] E Poli et al, Computer Physics Communications 225 36–46 (2018)
[3] M Kong et al, Plasma Phys. Control. Fusion 64 044008 (2022)
[4] A T Vu et al, Fusion Engineering and Design, 222:115501 (2026)

System Engineering Approach to Plasma Control System Design for Future Fusion Power Plants

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

The increasing complexity of future fusion power plants requires advanced methodologies for plasma control systems (PCS) beyond traditional control development approaches. This work presents a model-based systems engineering (MBSE) framework for the design, validation, and lifecycle management of PCS for next-generation fusion devices. The proposed methodology adopts a structured MBSE approach to manage design complexity and enable systematic integration of requirements, functional models, system architecture, and performance validation. A centralized PCS database is developed using SysML-based modeling to define and scaffold the system engineering elements of the project. This database captures stakeholder requirements, control functions, system interfaces, and architectural elements, providing a unified representation of the PCS design. The framework establishes explicit links between requirements, functional behavior, and verification activities, enabling full traceability from high-level stakeholder needs to test cases and validation procedures. To support model validation and performance assessment, the framework integrates TokSys simulation models within the MBSE workflow. This integration enables simulation-based verification of PCS functionality and provides a consistent environment for evaluating control performance under realistic operational scenarios. The approach also supports verification planning, change impact analysis, and design evolution, facilitating collaborative development across multidisciplinary teams. By transforming PCS development into a traceable and model-driven process, the proposed framework improves design robustness, reduces development risks, and enhances system reliability. The methodology also enables systematic management of interfaces between diagnostics, actuators, and control algorithms, which is critical for achieving stable and efficient plasma operation in future fusion reactors. Furthermore, the structured representation of system knowledge provides a foundation for knowledge transfer, long-term system maintenance, and scalable deployment across multiple reactor concepts. The implementation of a comprehensive system engineering management plan for PCS is expected to support future reactor development efforts and strengthen collaboration with international fusion research programs and industry partners. In addition, the proposed MBSE framework can interface with Product Lifecycle Management (PLM) environments to support configuration management and lifecycle governance of PCS models, enabling digital continuity from design through operation. The integrated simulation models also provide a foundation for developing Digital Twins of the PCS for predictive analysis and operational scenario evaluation. Ultimately, this work demonstrates how MBSE can provide a scalable and verifiable framework for addressing the control challenges of next-generation fusion power plants, enabling more efficient design processes and accelerating the realization of commercially viable fusion energy systems.

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#13500

Pellet trajectory optimisation studies for the ITER Disruption Mitigation System

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

A test laboratory for a Shattered Pellet Injection (SPI) system to support technology development for the ITER DMS has been constructed at the HUN-REN Centre for Energy Research (CER) in Budapest, Hungary. The device launches a solid hydrogen, or neon pellet, which is shattered on a shattering head. It is crucial that the pellets reach the shattering head intact to ensure that the resulting fragments are delivered reliably to the plasma for mitigating disruptions. Therefore, the study in this contribution focuses on the key survival aspect of the pellet flight.
Recent investigations indicate that various factors such as launch parameters, pellet guiding geometry, and propellant gas asymmetric flow around the pellet can induce significant angular tilt in pellets. The resulting unstable flight can lead to premature impact of large pellets, as foreseen for the ITER Disruption Mitigation System, with the restricting orifices in the flight line due to gate valves etc prior to reaching the shattering head. To avoid these collisions, the statistical dispersion of the pellet trajectory must remain within limits through targeted engineering solutions.
In this research, we are focusing on optimizing pellet trajectory, using the combination of pellet guide rail, funnels and chamber sizes. We have implemented also a converging rail system designed to reduce pellet oscillations and "bouncing" between guide rails. A funnel-shaped guiding concept was investigated to control rotational motion immediately after the gas pressure reduction phase. In addition, the projectile flight path was modified to evaluate the rotational behaviour of the pellet during free flight. Various guiding geometries were examined to characterize the influence of different guidance methods on pellet tilt. Throughout the tests, pressure data were recorded to assess the sensitivity of the system to the relevant parameters.

Real-Time GPU-Accelerated Equilibrium Quantity Reconstruction Using Deep Learning for Control at ASDEX Upgrade

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

The recent years have shown Deep Learning (DL) to be a versatile tool in real-time plasma control. Once trained, even complex DL models can be used to make predictions on new data at comparatively trivial computational expense. This makes them a natural fit for applications in which high-fidelity real-time calculations using traditional approaches are not feasible. GPU acceleration can compound this benefit by substantially increasing the available computational performance. GPUs are typically underrepresented present day in real-time applications because of their propensity for increasing latency.

Plasma control requires accurately reconstructed magnetic equilibrium quantities such as positional markers in real time. But the complexity of solving the Grad-Shafranov equation can limit the achievable fidelity and latency of real-time equilibria.

As a result, we created and tested a data-driven DL model as a possible substitute to ASDEX Upgrade's (AUG's) long serving function parameterization-based real-time control equilibrium [1]. The DL model was implemented using our standardized and GPU-accelerated DL integration pipeline [2] and integrated into AUG's Discharge Control system (DCS) [3].

The DCS used the DL model outputs for position control in three discharges of the 2024-2025 AUG campaign. The networks performed above expectations and proved the feasibility of using a DL model for this task in routine operation. Switching to DL allowed for more outputs at higher accuracy while reducing latency by almost one order of magnitude.

In the 2026 AUG campaign, we will improve the models further and introduce more complex outputs with the ultimate goal of providing a real-time-capable data-driven equilibrium on par with traditional Grad-Shafranov solvers in terms of accuracy and features.

This contribution will discuss the initial models and their results as well as the improvements and their impact during the 2026 AUG campaign.

1. P.J. McCarthy, PhD thesis, University College Cork, 1992
2. Illerhaus et al., Fusion Eng. Des., 2025
3. Treutterer et al., Fusion Eng. Des., 2014

Feasibility and Controllability Assessment of Negative Triangularity Plasmas in JT-60SA in view of OP2

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Negative triangularity (NT) plasma configurations have recently regained attention as a possible alternative shaping strategy for future tokamak reactors, motivated by experimental and modeling results suggesting improved core transport and edge behavior. TCV and DIII-D tokamak experiments show that NT plasmas combine H-mode–like core pressure with L-mode–like edge kinetic profiles, resulting in an ELM-free, high-performance regime. These observations motivate further investigations of NT configurations in larger devices, particularly from the perspective of equilibrium control and operational feasibility.

In this work, the feasibility and controllability of NT plasmas for JT-60SA during Operation Phase 2 (OP2) are assessed through an integrated equilibrium, control, and scenario-consistency analysis. A set of NT target shapes is generated by varying the upper triangularity and elongation while satisfying strike-point location constraints in single-null configurations. The analysis assumes a reduced central solenoid current (±15 kA) and a conservative estimate of flux consumption based on OP1 experience, with the target shape reached at the beginning of the flat-top phase.

Free-boundary equilibria are first computed using FBT/FGS and subsequently refined using the linearized plasma response model (FGE) to reduce boundary shape errors through active poloidal field coil current adjustments. The resulting equilibria are cross-checked against alternative free-boundary equilibrium solvers (CREATE-NL+/CREATE-L) to ensure consistency. Pressure and current profile compatibility and time evolution are verified using the RAPTOR code, assessing the consistency between the prescribed equilibria and achievable profile dynamics, while NT turbulence properties are assessed with GENE simulations.

The feasibility of the candidate equilibria is evaluated against OP2 operational constraints, including poloidal field coil current and voltage limits, strike-point positioning, and vertical stability. Stabilizability is assessed using a Best Achievable Performance analysis, in which the vertical instability is counteracted using the maximum available actuator voltages. Realistic power-supply models and saturation limits are adopted throughout the analysis. In addition, shape controllability is quantitatively evaluated using control-oriented metrics and minimum coil saturation margins.

The results indicate that a subset of NT shapes compatible with JT-60SA OP2 constraints can be achieved and stabilized with acceptable margins under conservative assumptions, supporting the feasibility of NT operation in JT-60SA OP2. More generally, the results motivate further assessments of NT configurations as a possible option for reactor-relevant test facilities such as

VNS, provided that control and stability requirements can be satisfied.

#13559

WEST PCS design to explore long pulse operation

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

The WEST (W Environment in Steady-state Tokamak) project has two primary objectives: preparing for the procurement and operation of the ITER actively cooled tungsten divertor, and mastering integrated plasma scenarios over relevant time scales in a metallic environment. To achieve these goals, WEST has developed specialized plasma controls to explore various operating regimes, particularly long pulses, while ensuring machine protection. Central to this effort is the WEST Plasma Control System (PCS), which computes actuator requests based on the pulse schedule and considers operational limits.

In 2025, WEST set a new plasma duration record for a tokamak, sustaining a discharge for over 22 minutes. This milestone validates the architecture of the CODAC (Control, Data Access, and Communication) systems and tools which were implemented a decade ago and have been continuously upgraded since then. The key components include the Pulse Schedule Editor (PSE), interfaced with the real-time Discharge Control System (DCS) via a custom program.

The PCS Gateway process manages the interface between DCS, originally developed for ASDEX-Upgrade, and WEST's infrastructure, which includes a Reflective Memory real-time network, timing system, and tokamak state machine. It handles multiple interface protocols, such as TIPC and Ethernet. Plasma controllers are developed and validated using MATLAB/Simulink, with an automatic code generation tool facilitating their integration into the PCS. To streamline development and deployment, a Jenkins Jobs-based graphical user interface automates the compilation of sources from various repositories, ensuring a consistent PCS configuration. A safety mechanism also prevents the use of unvalidated PCS releases for plasma discharges.

Initially, the PCS was designed to archive all internal data during a plasma discharge. However, for pulses exceeding 100 seconds, the data volume surpassed the PCS server's memory capacity. To address this, strategies for selecting subsets of data for archiving were implemented for long-duration operations.

The PSE, originally developed for the W7X stellarator and known as Xedit, was adapted for WEST through collaboration with the W7X CoDaC team. Xedit includes an event-handling capability that allows scenario changes, extensively used for long discharges. For example, the management of central solenoid flux consumption is used to trigger a plasma soft stop when available flux has been consumed, regardless of the scenario. This ensures the best use of the available flux and results in the longest possible discharge.

This paper will detail the capabilities and evolutions of the WEST PSE and PCS for performing long-duration discharge operations.

#13561

Advancement of control systems toward next SPIDER experiments and first MITICA operation

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

The ITER Neutral Beam Test Facility (NBTF) in Padua aims at developing MITICA, the full-size prototype of the ITER Neutral Beam Injector (NBI). To meet the strict requirements imposed by the physics of the additional heating for the ITER fusion experiment, NBTF also hosts SPIDER, the ITER's full-size beam source. SPIDER allows for in-depth study of plasma characteristics and beam quality using a wide range of diagnostic systems.

Over the last two years, SPIDER experimental campaigns and MITICA commissioning have been quite demanding from the COntrol and Data Acquisition System (CODAS) perspective. Indeed, in SPIDER the control systems and related strategies have been continuously evolving since 2015, and their optimization has been tested and consolidated in the last campaign. In MITICA, the latest high-oltage

tests allowed the control frameworks to be validated from both the Plant Units (PU) and data acquisition perspectives.

In view of the next SPIDER campaign, a new Plant Unit will be integrated, a new unsupervised control strategy will be implemented for the first time at NBTF to allow nightly operations without operator’s presence. The MARTe framework will be updated to its MARTe2 version with all its components to integrate the latest features and to comply with the MITICA control system.

In preparation for the first MITICA experimental campaign, the control of all Plant Units will be tested and integrated into MITICA CODAS and new strategies will be applied to address the high-demanding diagnostic systems such as MITICA cameras and the diagnostic calorimeter (MSTRIKE).

The paper highlights challenges faced over the last two years in terms of control systems for performing high-value scientific campaigns. In particular, with SPIDER we focused on optimizing the data acquisition systems, communication with Plant Units and developing new tools to improve the quality of the operation management. In MITICA we aimed at building and testing a control system capable of meeting the requirements of such an innovative system.

#13565

Implementation and Benchmarking of a Software?in?the?Loop Flight Simulator Embedding the Discharge Control System of ASDEX Upgrade

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Saving experimental time, reducing costs and mitigating undesired plasma behaviors are central challenges in present-day and future magnetic fusion campaigns. A pre-shot validation of the pulse schedule (the set of control instructions for a plasma discharge) and of the device control system is therefore important because it can improve experimental readiness and safety. These goals can be addressed effectively through flight simulators, which are software tools that predict the time evolution of plasma and machine variables starting from a prescribed pulse schedule and closing the intricate feedback loop between the plasma model and the device control system. Developing and operating flight simulators in the same environment as the control systems enables their use in routine validation pipelines, even on-the-fly during experimental days.

At ASDEX Upgrade (AUG), the flight simulator Fenix has been built on the ASTRA suite and on a MATLAB/Simulink model of the AUG Discharge Control System (DCS). ASTRA is a 1.5-D transport and equilibrium solver that includes simple plasma and tokamak models. The MATLAB/Simulink component incorporates models of the schedule interpreter, exception handler, observers, monitors, controllers, and actuator managers of the DCS, originally written in C++. While Fenix successfully reproduces a variety of AUG scenarios, its reliance on a model prevents direct testing of the DCS code that runs during experiments. Moreover, the translation from the native C++ implementation to MATLAB/Simulink may introduce hidden discrepancies.

To overcome these limitations, a Software-in-the-Loop (SIL) flight simulator that embeds the actual AUG DCS software has been developed. The DCS framework runs almost unchanged and no intermediary MATLAB/Simulink model is employed. Communication between the DCS and ASTRA is performed via a shared-memory mechanism similar to that used in Fenix. Synchronization of the two software tools is handled by the DCS, which has been programmed to adapt its internal time according to the execution of ASTRA simulation steps.

We present the implemented communication and synchronization mechanisms, together with benchmarks against the Fenix simulator on a representative set of ASDEX Upgrade discharges. The SIL-based simulator reproduces the reference shots with comparable accuracy, while requiring less wall-clock time per simulation. These first results demonstrate that a SIL flight simulator can provide a relevant framework for pre-shot validation.

#13583

Application of Alpha Heating Power Emulation on ASDEX Upgrade

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

The operating scenarios of present-day tokamaks, which are dominated by external heating, cannot be directly extrapolated to future generations of reactors, where alpha heating is expected to prevail. In such burning, self-heated plasmas, nonlinearities arise, and the plasma response to them is not yet fully understood.

In a burning plasma, the onset of a Neoclassical Tearing Mode (NTM) that significantly degrades energy confinement can lead to a reduction in alpha heating power. A key question is whether this decrease in self-heating provides a stabilizing feedback on the NTM or instead alters the plasma profiles in a way that allows the mode to persist or grow.

Similarly, as a burning plasma approaches the high-density limit, confinement degradation is expected, again reducing alpha heating. This reduction in self-heating may introduce additional feedback mechanisms that further degrade stability and narrow the available operational margins, necessitating dedicated feedback control to counteract these effects.

Understanding these nonlinear behaviours is essential for predicting and controlling burning plasma performance. To enable their investigation on existing devices, the Reactor Emulation Protocol presented in [Fable et al., 2023] is employed. Within this approach, the scenario of a selected future burning-plasma reactor is scaled to a present-day tokamak scenario, and the resulting emulated powers, including alpha heating and equipartition powers, are realized through feedback control using the available heating mix.

In this contribution, the Reactor Emulation Protocol is incorporated into the Discharge Control System of ASDEX Upgrade, including the real-time RAPTOR code used to provide the emulated quantities. The real-time application is then demonstrated on a selected ASDEX Upgrade scenario, and the impact of self-heating on plasma performance is discussed.

#13603

Modifications to the DIII-D Plasma Control System for Commercial use and Fusion Reactor Relevance

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Modifications to the DIII-D Plasma Control System for Commercial use and Fusion Reactor Relevance

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The DIII-D Plasma Control System (PCS) has proven itself over decades as a reliable foundation for experiment driven operations of the DIII-D Tokamak. With access to the most extensive diagnostic suite ever deployed on a tokamak, the PCS is continuously evolving to enable cutting-edge control experiments. From basic machine protection to advanced AI-driven real-time control, the PCS handles the entire range of operational demands. The flexibility of this system accelerates new feature development, with design choices that prioritize adaptability over absolute versatility. The PCS is well-positioned for reactor relevant applications; the main steps involve extending operational run-time and refactoring interfaces to clearly delineate its dual control functions. Basic machine protection and advanced physics algorithms must be fundamentally separated, and the “hard-coded” DIII-D specifics must be abstracted away in order for the PCS to be easily adopted onto a completely new device, whether that be for a future fusion reactor, or another industry entirely. The creation of this “General Control System”, is beneficial for any industry with need for a real-time control system. Rather than developing a new one from scratch, this product could be readily deployed on any device with actuators and sensors.

Keywords: Plasma Control, Software, Control System, Real-time

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NUMERICAL CFD MODELLING OF HYDROGEN AND NEON PELLET DE-SUBLIMATION AND OPTIMIZATION OF ITER DMS 28.5 mm and 18 mm COLD CELL DESIGNS

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

The ITER tokamak will be equipped with a machine protection system to avoid or mitigate damage to its in-vessel components and structures in case of plasma disruptions. The Disruption Mitigation System (DMS) will convert plasma thermal and magnetic energy into radiation while avoiding the formation or dissipating the energy of runaway electrons, by means of Shattered Pellet Injection (SPI) and assimilation into the plasma of hydrogen, neon or their mixtures.
Based on DMS modelling and return of experience from operating Tokamaks, the ITER DMS will rely on cylindrical pellets of two diameters, D = 28.5 mm and D = 18 mm, and pellet aspect ratio $1.5 < L/D < 2$, requiring two different designs for the de-sublimation Cold Cell (CC) heat exchanger, cooled by ~5 K supercritical helium. The design of the CC, the acceleration tube, the axial temperature gradient control and the gas pre-cooling were supported and optimized by 2D axial-symmetric and 3D numerical simulations, based on a validated de-sublimation Open-FOAM CFD model [1], in order to minimize the pellet formation time while ensuring a sufficient pellet aspect ratio, within the limits of the available cooling power.
The numerical models allowed us to define an optimal range of the following parameters for each CC diameter, which were later fixed considering integration and manufacturing constraints: the CC cold zone length, the location and temperature of the gradient control rings, the pre-cooled gas injection temperature, and the thickness of the acceleration tube.

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Disclaimer
The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

#13653

WEST scenario and control design based on the flight simulator NEST

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

To facilitate the development of plasma controllers and scenarios, physicists and engineers have relied for decades on a versatile set of codes. However, the gap between simulations and reality, or the computing time required to run the codes, has hindered such an approach, and the complementary trial-and-error method during commissioning sessions is still widely used. In such cases, simulation results were mostly used to tune the plasma parameters along the different tests.
To go a step further, the flight simulator NEST (Nice Environment for Simulated Tokamak) has been developed to prepare and facilitate the WEST operation. It aims to compute with sufficient accuracy the time evolution of the plasma and the different subsystems of the tokamak based solely on the pulse schedule as input. For this purpose, NEST is interfaced with the WEST pulse schedule editor Xedit and uses the same configuration files as the WEST Plasma Control System (PCS). NEST is based on the free boundary equilibrium code NICE and uses the MUSCLE3 library and IMAS to connect additional physics and control modules together. Such a structure allows NEST to be used on several other tokamaks like JT60-SA (for comparison with its own flight simulator [E. Joffrin, FEC 2025]), and also to test the architecture of the ITER Pulse Design Simulator [M. Schneider, SOFT 2026]. This approach makes NEST a real collaborative platform, allowing easy integration of contributions from other developers.
Based on MATLAB Simulink, the NEST PCS uses the same controllers as the WEST PCS. Thus, it can be used to assess control development and validate any modification before deployment on the real plant.
As a proof of concept, several use cases of NEST for the operation of WEST is presented in this contribution. First, the strategy to develop the negative triangularity configuration scenario is described. Then, based on the latest developments linked to the integration of the TORBEAM code to model the EC heating and current drive, the flight simulator is used to prepare the commissioning of this new system and check its operational domain. Finally, as an example of plasma control development, a new feedback control mode has been developed, tuned, and assessed using NEST to optimize the double null configuration. The last part is dedicated to the current limitations of the code and how to overcome them in anticipation of the intensive use of this type of simulator for the ITER operation.

#13669

Application of the ITER Real-Time Framework to a Two-Color Interferometer Diagnostic

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Developed primarily to manage the demanding operations of the Plasma Control System (PCS), the ITER Real-Time Framework (RTF) serves as a foundational software architecture for complex real-time tasks. To evaluate the feasibility of applying the RTF to diagnostic systems, an RTF-based application for a Two-Color Interferometer (TCI) was implemented and tested on the Korea Superconducting Tokamak Advanced Research (KSTAR). To assess the framework under operational conditions, the application was enhanced to support standard pulse operation and operator interaction by incorporating three major functions: execution under the standard ITER operation sequence, real-time signal publication and retrieval, and continuous live monitoring.

The implemented TCI application acquires processed phase data at 625 kS/s and processes it at a 1 kHz cycle to compensate for environmental disturbances and to calculate plasma density in real time. Demonstrating the high modularity of the framework, the application was efficiently constructed by reusing standard function blocks already utilized by the ITER PCS, alongside a specific device support block developed for data acquisition. Crucially, the system supports the Synchronous Data Network (SDN) interface, allowing the calculated density to be fed directly to the ITER PCS with low and bounded latency. This real-time density information is planned to be used for active plasma density control in future operations.

During pulse execution, the application operates under the integrated ITER operation sequence, managing state transitions robustly. Concurrently, processed signals and plant status data are published and retrieved via PVAccess. To support this operation, an operator interface (OPI) was utilized for real-time monitoring, enabling users to track raw waveforms, calculated outputs, and operation states without interrupting the real-time data processing loop.

The operational evaluation confirmed the stable execution of these newly applied features under continuous KSTAR tokamak operations. The integrated operation sequence governed state transitions consistently, and data communication via both SDN and PVAccess maintained consistent timing. The OPI confirmed stable continuous monitoring throughout the pulse. These results validate the integration of the RTF-based TCI application and demonstrate its capability to support complex diagnostic applications and future plasma control.

Control-oriented plasma magnetic models cross-validation with EDICAM

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Model-based design of equilibrium control systems in tokamaks relies on the availability of reliable linear models of the plasma response [1,2]. Although approximated, such a description of the plasma behaviour is sufficient for control design purpose, since control algorithms aim at regulating, i.e., keeping constant, the relevant plasma parameters over given time windows of the discharge, by rejecting disturbances and compensating for model uncertainty.

The main objective of this contribution is the assessment of the validity range for the CREATE control-oriented linear model [3], and therefore of its prediction capability. To this aim, we cross check the reconstructed poloidal flux map and plasma boundary with the images acquired by the EDICAM during the Integrated Commissioning and Operation 1 of JT-60SA [4]. In this paper we focus on experiments where limiter plasmas have been moved vertically during the flat-top, hence varying the limiter contact point on the inboard. In particular the 2D last closed flux surface (LCFS) estimated with the CREATE control-oriented models is projected to the corresponding EDICAM image to compare it with the radiating belt location. The location of the radiating belt derived directly from the EDICAM image by assuming a fixed poloidal plane is also compared to the LCFS estimated with the CREATE model. Moreover, EDICAM images are also used to cross-validate the reconstruction performed using the CARONTE code, which is ML-based algorithm based on a single physics-informed neural network recently proposed in [5].

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#13743

Geometric Optimization and Vertical Stability Analysis of Negative Triangularity-based C-shaped Plasmas for Minimizing Bad-curvature Regions

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

This study investigates the design and stability of C-shaped plasma configurations, specifically engineered to minimize the "bad curvature" regions inherent in Negative Triangularity (NT) D-shaped plasmas [1]. The research is motivated by the hypothesis that introducing an indentation on the outer region of a NT plasma can vertically stabilize for wide variety of setups, enabling high elongation operation that increase plasma confinement and fusion power, and geometrically improve the magnetic field line curvature, thereby suppressing pressure driven ideal MHD instabilities [2, 3]. To explore this, a series of C-shaped plasma equilibria were designed with a focus on high elongation and negative triangularity, specifically tailored to evaluate the feasibility of this unique configuration from both physics and engineering perspectives. The investigation utilized the TokaMaker code for free boundary equilibrium calculations, employing parametric curves for C-shape boundary identification [4]. The study focused on scans of key geometric parameters, including indentation, elongation, minor radius, and curvature, to quantify their impact on the plasma's vertical instability growth rate ($\gamma\tau w$). By systematically varying these constants, we identified the sensitivity of the plasma's vertical position control to its internal geometric properties. Following the characterization of vertical stability, we performed ideal MHD stability analysis to characterize the specific modes triggered within these unique configurations. MHD calculation results obtained from the MISHKA code using refined equilibria by the CHEASE code reveal a complex tradeoff between local and global stability [5, 6]. While increasing the indentation vertically stabilizes and physically expands the "good curvature" region—broadening the interchange stability window as confirmed by the Mercier criterion—it simultaneously exacerbates instability [7, 8]. This study identifies a viable operational window where vertical stability can be maintained through optimized geometric parameters. These findings provide engineering insights and design guidelines for sustaining high performance fusion devices utilizing nonconventional plasma shapes.

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Pellet formation of a dual shattered pellet injection system for enhanced operational versatility

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

The dual shattered pellet injection (SPI) systems installed on KSTAR (Korea Superconducting Tokamak Advanced Research) have been in operation since 2019 to allow disruption mitigation studies for the ITER Disruption Mitigation System (DMS), which will use multiple injectors. The KSTAR SPI system has the unique capability of simultaneous injection from two identical injectors located at toroidally opposite positions (180° apart). Each injector can form and launch up to three pellets with different diameters and lengths, enabling flexible operation for disruption mitigation experiments.

Pellet quality is monitored using a microwave cavity (MWC) system and an optical pellet diagnostic (OPD) system which provides detailed images of the injected pellets prior to their arrival in the plasma. The pellet formation process has been continuously optimized to ensure the production of intact pellets. The pellet formation recipes have been integrated into a new PLC-based fully automated state machine employing electronic pressure controllers (EPCs) to ensure pellet injection with maximum reproducibility, which is essential for the effective execution of mitigation experiments.

This paper presents the developed pellet formation process and the recent operational experience, including plasma experimental results for disruption mitigation with shattered pellet injection using D, Ne and mixtures thereof obtained during recent KSTAR campaigns.

Analysis of aerodynamics and mechanical factors affecting the pellet flight in shattered pellet injection for the ITER DMS

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Mitigating thermal and electromagnetic loads during a disruption is a key factor in preventing damage to a tokamak. Disruption Mitigation System is of major importance for this purpose. The system uses Shattered Pellet Injection system, which utilizes a high-pressure gas pulse to launch large cryogenic hydrogen/deuterium/neon pellets. These are shattered into smaller fragments on a shattering plate just before the tokamak's vacuum vessel. Intact arrival of the pellet at the shattering plate is of major importance therefore any premature breakage must be avoided.

Within the SPI test laboratory system at the HUN-REN Centre for Energy Research, the pellet exits the acceleration barrel and enters a propellant gas suppressor. In this chamber, the pellet is mechanically guided by rails over a short distance while the propellant gas can expand. Although the pellet subsequently travels through a predominantly rarefied gas environment, the propellant gas overtakes the pellet shortly after launch, introducing aerodynamic perturbations. These disturbances may induce minor impacts between the pellet and the guiding rails, potentially causing the pellet to exit the suppressor chamber in an unstable flight condition. Such instability can result in pellet rotation and undesirable collisions before reaching the shattering head.

To mitigate these effects, theoretical models are developed to quantify the influence of residual propellant gas flow and pellet–guiding structure interactions. The aim of this paper is to analyze the dominant perturbation mechanisms and to propose solutions that ensure stable pellet flight up to the shattering plate. In addition, the effects of proposed guiding and stabilizing mechanisms are evaluated to assess their effectiveness in reducing pellet instability.

Gain Tuning Based on Step Response for Multi-Variable PID Control of DEMO Plasma with Engineering Constraints

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Core plasma in a fusion reactor is a complex system in which various physical phenomena interact with one another. Multiple physical quantities must be controlled in a complex manner. In this study, we conducted control simulations of a JA-DEMO reactor using the integrated plasma transport code TOTAL, specifically focusing on the multi-variable control of fusion power and line average electron density by manipulating NB power and pellet injection rate. Feasibility of multi-variable PID (Proportional-Integrated-Derivative) control is studied.

In a multi-variable PID framework, it is essential to optimize the proportional gains, integral time constants, and derivative time constants for all combinations of manipulated and controlled variables. Conventionally, these parameters have been determined empirically, and establishing a general determination method has been a significant challenge.

In this study, we proposed two determination methods based on the step response of controlled variables. In the first method, a response matrix is constructed by assuming a linear relationship between manipulated and controlled variables. Each element of the matrix is the steady state change in each controlled variable relative to a step change in each manipulated variable, and the control gains are then determined by calculating its inverse matrix. This approach offers the advantage of determining gains regardless of the response sensitivity, though time constants cannot be determined. The gains determined by this method were relatively small. In the control test where the target value of electron density was changed while keeping the fusion power constant, the results showed relatively slow response of the controlled variables. Nevertheless, these gains allow the system to maintain stable behavior even under engineering constraints such as response delay of the actuators.

The second method is the Ziegler-Nichols' Process Reaction Curve method. In this approach, the gains and time constants are determined based on the maximum slope and the delay time of the response curve. This procedure is applied to each relationship between the manipulated and controlled variables to define their respective parameters. However, when the response is small, the gains may become excessively large. The results of the same control tests showed that this method yielded relatively large gains and achieved high control performance with small deviations from the target values. However, because the gains are large, it may not be suitable when practical engineering constraints are taken into account.

Influence of pellet formation and acceleration parameters on the fragmentation of solid hydrogen pellets for Disruption Mitigation System

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

Magnetically confined fusion reactors are exposed to disruption events resulting from sudden loss of plasma confinement. For high-energy plasmas, the Shattered Pellet Injection (SPI) is the ITER reference disruption mitigation technique to limit thermal and electromagnetic loads on first wall and mechanical structure. The SPI consists of injecting a large cryogenic solid hydrogen multiple pellet that is shattered against an inclined plate before entering the plasma vessel. The resulting shard cloud enables controlled radiative cooling, mitigates halo currents, and reduces runaway-electron generation. Achieving reliable disruption mitigation requires precise control of shard characteristics, including fragment size and velocity distribution, and injected mass. These properties strongly depend on pellet formation (temperature, pressure and gas flow rate) and impact conditions (velocity and impact angle). A detailed understanding of solid hydrogen fragmentation mechanisms under SPI-relevant conditions is therefore essential for the design and performance optimization of the SPI technique and will be presented.

This study presents an experimental and numerical investigation of the formation, propulsion, and fragmentation of solid hydrogen pellets under ITER-relevant SPI conditions. Firstly, the dynamic impact behavior (force response, fracturing process, and kinetic energy effect) is characterized using a Hopkinson bar apparatus. Secondly, a quantitative analysis of pellet fragment generation during acceleration and free flight is conducted for pellets produced under different formation parameters. Thirdly, controlled oblique impact experiments on a rigid plate are performed to characterize the fragmentation process and the resulting shattered fragment plume relevant to SPI operation. Finally, two numerical approaches are evaluated for their capability to reproduce experimental impact and fracture behaviors: i) a micromechanical damage model based on probabilistic defect activation, able to describe tensile damage and confined plastic strain, and ii) an isotropic, rate dependent, elastic–plastic material with separate tension and compression yield curves. The results obtained with the two models are outlined and their suitability for predictive simulations of SPI fragmentation dynamics for the Disruption Mitigation System in magnetically confined fusion reactors, is discussed.

Feedback Control of Plasma Density at ASDEX Upgrade using Magnetic Perturbation Coils

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

ASDEX Upgrade (AUG) is equipped with 16 independent magnetic perturbation coils [1] arranged in two rows in the upper and lower regions of the vacuum vessel. These coils can be configured to generate toroidal mode numbers n=1–4 with an arbitrary phase shift between the rows. They are used for several applications, including error field compensation and the application of magnetic perturbations that enhance particle transport in the pedestal, thereby reducing plasma density at a constant fueling rate [2].
Devices with full tungsten walls such as AUG require a minimum fuel injection rate to avoid excessive radiation caused by high concentrations of heavy impurities in the plasma. Consequently, AUG scenarios requiring low plasma density, such as ELM suppressed scenarios or dedicated transport studies, have typically relied on density pump-out produced by magnetic perturbation coils. Until recently, the perturbation amplitude was predefined before the discharge, meaning the achieved plasma density depended strongly on day-to-day machine conditions.
Real-time control of the current in each coil has now been implemented within the AUG Discharge Control System (DCS), enabling the magnetic perturbation coils to act as actuators for real-time plasma control. This contribution first presents the development of a plasma density controller that adjusts magnetic perturbation coil currents to achieve a desired density. The controller will then be applied to studies of scenarios with ELM suppression, which is possible only at very low plasma densities [3] that can be achieved using magnetic perturbation coils. The controller will be used to enable studies of ELM-suppressed regimes at a prescribed plasma density.

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Hybrid MPC–Optimisation Workflow for Long-Horizon RAPTOR Scenario Development

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

Optimisation-based scenario development using physics-based plasma models becomes increasingly challenging for long-duration Tokamak scenarios. In particular, non-inductive ramp-up phases extending over thousands of seconds require actuator trajectories, such as Electron Cyclotron Resonance Heating, with a high number of degrees of freedom (DOF) in order to satisfy physics and operational performance objectives under tight constraints. The resulting high-dimensional nonlinear optimisation problems are therefore prone to convergence failure or infeasibility due to strong nonlinearities and coupling between control variables and plasma states. A key difficulty is the construction of an initial trajectory that is sufficiently close to the desired target to allow the optimisation to converge, a process which is typically highly manual and time-consuming. In much of the applicable optimisation literature, these challenges are addressed by restricting the problem to low-DOF actuator parameterisations, an approach that is effective only for short-duration scenarios. For longer scenarios, an alternative strategy is to progressively increase the complexity of the objective function across multiple optimisation stages, but this requires further manual intervention between steps and does not guarantee convergence.

In this work, we present a systematic and automated method to generate high-quality initial trajectories for long-horizon, high-DOF optimisation problems using a closed-loop model predictive control (MPC) framework. Our plasma evolution is simulated using the RAPid Plasma Transport simulatOR (RAPTOR), which self-consistently solves four coupled one-dimensional PDEs for poloidal flux diffusion, electron and ion thermal transport, and electron particle transport. The nonlinear RAPTOR model is linearised continuously at
each simulation time step using state Jacobians obtained directly from the code. This yields a sequence of locally valid linear models which are used in an MPC formulation to solve reduced optimisation problems over a finite prediction horizon. By executing the controller in a receding-horizon fashion, starting from an arbitrary initial condition, a complete actuator trajectory is generated that achieves the desired objectives while respecting operational
constraints. The resulting trajectory provides a close-to-feasible initial guess, significantly simplifying the subsequent open-loop nonlinear optimisation. The approach is demonstrated for the non-inductive ramp-up and burn phases of the Spherical Tokamak for Energy Production (STEP) reactor, in which the objectives include plasma current, fusion power tracking, regulation of
the q-profile and li3, subject to both linear and nonlinear constraints on actuators and plasma states.

This hybrid MPC–open-loop optimisation workflow offers a robust and scalable route for long-pulse scenario development, reducing manual intervention and improving optimisation convergence for next-step Tokamak operations.

Cryogenic pellets production with supercritical helium cooling in a compact cold cell assembly for ITER Disruption Mitigation System

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

As a first step, CEA/DSBT designed and operated a liquid-helium (LHe) cooled pellet injection test bench to investigate the production and acceleration of large cryogenic pellets made of protium, deuterium, neon, and their mixtures for the ITER Disruption Mitigation System (DMS). Protium pellets with a diameter of 28.5 mm and a length of 57 mm were successfully formed in 20 minutes and accelerated to velocities exceeding 500 m/s. Following this first success, additional studies have shown that using gaseous helium (GHe) at 1.25 bara thermalized in LHe bath as a coolant, is also feasible to produce large cryogenic pellets, albeit at the cost of slightly longer formation times.

For ITER operation, the cooling system of the DMS pellet injection system relies on a supercritical helium (SHe) forced-flow circulation. Additionally, the integration requirements impose a compact cold cell assembly to produce large cryogenic pellets. To assess such operation and integration requirements, CEA/DSBT upgraded its test bench by integrating a new cryostat able to produce supercritical helium cooling flows up to 1 g/s at 5 K and 5 bara. In addition, a new compact Cold Cell Assembly was designed.

The main difference between SHe and LHe cooling lies in the temperature variation induced during desublimation heat removal. This temperature increase is a limitation during pellet formation, as the cell temperature must be maintained as low as possible to minimize the formation time.

This paper presents the new compact cold cell assembly and a comparative study on the pellet formation between SHe cooling performances and previous results obtained with GHe and LHe cooling. Key parameters investigated include the cold cell temperature, the desublimation pressure, the cooling flow rate, the shell thickness for pellets layered with different material (H2/Ne), and the mixture composition for the mixed pellets. Protium pellets were the main focus in this study, as they are the most extensively characterized under LHe and GHe cooling. Other types of DMS pellets were also briefly examined. The results obtained with SHe show that formation times, pellet aspects, and velocities are comparable to those achieved with LHe and GHe cooling. In addition to this comparison study, a new recipe was also developed to produce smoother pellet aspect and less debris for protium pellets. This proof of principle for SHe cooling in a compact cold cell assembly represents a significant milestone in the design study of the ITER-DMS cold cell.

Preliminary analysis of the formation cycle of cryogenic pellets for disruption mitigation in DTT

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Cryogenic pellets are widely used in many nuclear fusion experiments such as tokamaks. In magnetically confined plasmas, they serve multiple purposes, such as fueling, ELM pacing and disruption mitigation. Due to their widespread use, their composition varies with the intended use. They are typically made up of pure hydrogen, deuterium or noble gases, such as neon or argon, or a mixture of these gases. Their purpose also determines their dimensions. For example, smaller pellets are injected into the plasma for fueling, while larger ones are used for disruption mitigation.

Focusing on disruption mitigation purposes, cryogenic pellets are used in multiple experiments such as ASDEX, EAST and KSTAR. For others, such as ITER, the design of their injection systems is still ongoing. The Italian Divertor Tokamak Test (DTT) is evaluating the design of the injection system of cryogenic pellets for disruption mitigation, in full power scenarios. A pipe gun is being evaluated as the main technology, with a gas gun as a potential acceleration system for these pellets. This kind of technology generates a high-pressure peak at the pellet base, dislodging it from its formation point and accelerating it through the barrel. Due to the high-pressure peaks achieved by the pulses at the pellet base, these need to be able to withstand significant mechanical stress, without shattering or deforming prematurely.

The formation process of cryogenic pellets used for disruption mitigation usually involves injecting the feeding gas into a cooled barrel where it desublimates into a solid form. This phase is vital to ensure that the gas injected into the coldest part of the barrel forms a solid block of ice rather than a mixture of ice and snow. In this study, we investigate the key pellet formation parameters (pressure and temperature) to optimize pellet quality for disruption mitigation applications, such as the Shattered Pellet Injection technology, considering deuterium (D), hydrogen (H), neon (Ne), and their gas mixtures. Multiple pellets were formed at the ENEA laboratories in Frascati, to optimize the formation process. Clear, solid cryogenic pellets were observed at the end of the process.

This study analyses the formation cycle of these kinds of cryogenic pellets presenting detailed results for pure pellets. The mixed pellet production system is currently in its development phase.

Heat flux control for the ITER upper first wall

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

For the for the ITER Start of Research Operation (SRO) phase, real-time monitoring and control of thermal loads on inertially cooled tungsten first wall panels (FWP), is a fundamental requirement of the Plasma Control System (PCS) currently under design. The work reported here focuses on real-time heat load control on the upper first wall chamber, where the proximity of a secondary, non-active X-point will lead to some of the highest loads anywhere on the first wall during diverted plasma operation. The scope of this work is to propose a control scheme for real-time thermal load regulation on the upper FWPs through minimal variations of the reference plasma shape, while respecting magnetic control constraints and discharge goals.
In this controller design procedure, a static model derived from CREATE-L for the plasma–vessel–coil electromagnetic dynamics is coupled with a non-linear heat flux estimator for an axisymmetric first wall. This lightweight model enables rapid exploration of the effects of flat-top magnetic equilibrium perturbations (such as secondary X-point position, distance between primary and secondary separatrix, and plasma–wall gap) on the plasma heat load impinging on the upper FWPs. The impact of the full 3D first wall geometry, including magnetic shadowing effects from toroidally shaped FWPs, is then assessed using the SMITER framework. This magnetic field line tracing tool provides high-fidelity validation of the heat flux mitigation achievable through the plasma shape variations required by first wall heat load control (FWHLC) in the vicinity of the secondary X-point region. Subsequently a Matlab/Simulink model for control-oriented time-evolving simulations, featuring the characteristic dynamics of magnetic actuators, the thermal evolution of FWPs under plasma loads, and a simplified infrared synthetic diagnostic, is connected in closed-loop with FWHLC. In these simulations, we show the ability of FWHLC to constrain within predefined limits the temperature increase of the upper FWPs which is estimated in real-time by combining the infrared temperature measurement and its growth rate. Regulation of the upper FWP temperature at the end of a discharge is achieved through continuous variation of the reference plasma shape, imposing the newly developed perturbative patterns onto the reference equilibrium.
The paper will present an assessment of the ITER FWHLC for the upper FWPs in closed-loop dynamic simulations for extended SRO diverted plasma discharges, using the Plasma Control System Simulation Platform to demonstrate the effectiveness of the proposed strategy.

#13870

Exploring passive disruption mitigation techniques in the ARC-class reactor: the Liquid Sandwich Vacuum Vessel.

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Plasma disruptions remain a significant threat to the commercial viability of fusion energy, as they have the potential to damage reactor components and necessitate lengthy and costly repairs. To prevent vertical displacement events from evolving into disruptions, active plasma stabilization techniques are employed. If a disruption occurs, massive gas injections have been found to severely decrease damage to surrounding structures. However, these techniques need active systems to operate, which can potentially fail when they are most needed. Moreover, high-field tokamak designs, which will have higher plasma currents and thinner-walled vacuum vessels to enable heat transfer to the blanket, will be especially vulnerable to disruption damage.

The Liquid Sandwich Vacuum Vessel design employs non-conducting ceramics (SiC-SiC) composites as its main structural material, with the addition of liquid lead channels. Adjusting the channel orientation controls the direction of induced currents and can passively reduce disruption force loading during disruptions. The force magnitude depends on the angle between channel orientation and magnetic field lines. While various channel topologies can be optimized by aligning conducting channels with the magnetic field to minimize disruption forces under worst-case scenarios, this case study explores the scenario of helically twisted channels.

This work presents the LSVV design, describes the modeling tools used to analyze 3D structures with helical conducting channels, and examines implications for vacuum vessel design. It describes in detail COMSOL Multiphysics modeling strategies and verification processes employed to analyze this design. In particular, the working principle of passive disruption force mitigation will be demonstrated through numerical models, and an initial verification and scale-up in complexity for electromagnetic models of vacuum vessels is carried out. Then, disruption forces arising during a vertical displacement event are quantified, and the vertical plasma stability is assessed through eigenmode analysis.

CREATE PEGASO: Plasma Equilibrium Generation and Automatic Scenario Optimization tool

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

The design of electromagnetic plasma scenarios is a key step in the technological assessment of tokamak devices. It involves the generation and validation of a sequence of MHD equilibria representing the critical points of pulsed operation, typically described through the time evolution of plasma current profiles provided by heat and particle transport simulations.
While many equilibrium tools are designed to explore individual plasma configurations within a prescribed scenario, optimization approaches are often limited to single plasma snapshots. In these cases, the optimization is performed independently for each equilibrium configuration and does not account for the scenario as a dynamically evolving system.
The optimization of a complete electromagnetic plasma scenario is considerably more complex. In addition to the standard static constraints associated with each plasma configuration—such as plasma current, boundary flux, plasma shape, and electromagnetic loads on the conductors—dynamic constraints must also be considered. These include the voltage requirements associated with the temporal evolution of coil currents and plasma profiles across the scenario.
This paper presents an approach for the integrated optimization of tokamak electromagnetic scenarios, simultaneously accounting for both static constraints at the level of individual equilibria and dynamic, multi-step constraints across the entire scenario evolution. In particular, we introduce CREATE-PEGASO, a computational tool for the generation and optimization of plasma scenarios based on the coupling of the CREATE fixed-boundary (CREATE-FXB) and free-boundary (CREATE-NL+) equilibrium solvers with a linearized plasma model. Given plasma current profiles from transport simulations, the machine geometry and the electromagnetic constraints of the device, PEGASO computes a fully optimized electromagnetic scenario, defined as a sequence of consistent free-boundary MHD equilibria. Target plasma shapes can be either prescribed a priori or determined as part of the optimization procedure, together with the corresponding PF and CS coil current trajectories, ensuring the satisfaction of both single-step and multi-step operational constraints.

#13903

Demonstration of Real-Time Multi-Actuator Plasma Density Control in KSTAR via Adaptive Fueling

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Particle fueling is a key component of tokamak plasma operation because it directly affects several critical control objectives, including density regulation and plasma detachment. However, reliably injecting particles for precise control in large-scale devices is challenging due to their size and operational complexity. The injected particles do not affect the plasma immediately, and the resulting changes in plasma conditions are often difficult to accurately predict. These uncertainties can reduce control performance and ultimately compromise the plasma stability. Consequently, next-generation tokamaks require advanced control frameworks capable of overcoming these fueling uncertainties and validated on existing devices under reactor-relevant conditions. To support such efforts, KSTAR has recently been equipped with feedback capabilities for both gas puffing and pellet injection, along with emulation of ITER-like delayed gas actuation via the NERF mode. Leveraging these capabilities, a real-time indirect model-reference adaptive control algorithm has been developed and experimentally validated in KSTAR. The algorithm regulates the line-averaged electron density around desired targets using coordinated gas puffing and pellet injection under reactor-relevant conditions, despite unknown fueling efficiencies and input time delays. Experimental results demonstrate effective plasma density regulation, highlighting a control framework capable of addressing fueling challenges in large-scale tokamaks and applicable to other plasma control objectives.

Exploration of Operation Scenarios for Compact Pilot Device (CPD)

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Compact Pilot Device (CPD) is an intermediate-step tokamak bridging KSTAR [1] and K-DEMO [2]. In this bridging role, CPD aims to realize an innovative compact fusion pilot plant that advances fusion reactor development through public–private collaboration [3]. CPD features a compact tokamak design ($R_0 = 4$ m) with a high magnetic field ($B_0 > 8$ T) and high-energy positive-ion-based neutral beam injection (NBI) with $E_{NB} \approx 400$ keV. CPD targets two flattop operation regimes: (1) a steady-state burning plasma with fusion gain $Q > 5$ and a non-inductive current fraction $f_{NI} \approx 1$, and (2) a high-performance inductive operation designed to maximize the fusion gain Q for net electricity generation. In this work, these operation points with a fusion power of 600 MW were developed using FUSE (FUsion Synthesis Engine) [4], satisfying target performance requirements within key physical and engineering constraints. FUSE is an integrated systems code framework for fusion reactor design that replaces conventional first-principles models with analytical and machine-learning-based surrogates to enable efficient integrated simulations of fusion devices. These points were identified by optimizing plasma performance predicted by the FUSE transport solver (called FluxMatcher), which incorporates transport fluxes from GKNN [5] for turbulent transport and the analytic Hirshman–Sigmar model [6] for neoclassical transport. Optimization was performed using a multi-objective genetic algorithm by varying physical and engineering parameters, including NBI and electron cyclotron (EC) heating powers, beam geometry, and target density levels. Furthermore, these optimized points were extended into dynamic operation scenarios through time-dependent simulations.

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Design of boronization system in JT-60SA

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

JT-60SA is a superconducting tokamak for plasma fusion experiment developed in Japan. This is an international joint research and development project involving Japan and Europe. The first integrated commissioning of JT-60SA has succeeded in 2023. The maintenance and enhancement work is currently ongoing, and further modifications will be made to improve plasma performance.
We are planning to introduce a boronization system step by step according to the schedule of JT-60SA project. In the first phase of the JT-60SA research plan, the plasma-facing first wall consists of the graphite. It is planned to replace the first wall with tungsten in the future. Boronization is particularly effective for the tungsten walls because it lowers the atomic number of the first wall. Therefore, before introducing the tungsten wall, boronization system should be introduced and improved for stable operation.
In the boronization process, a boron film is generated on the surface inside the vacuum vessel by supplying a boron source gas during a glow discharge. As a result of boronization, oxygen in the vacuum vessel is fixed in the boron film, and it is expected that impurities will be reduced.
A boronization system was also introduced in the previous tokamak device JT-60U, and its effectiveness in reducing impurities was confirmed. Decaborane, which is solid at room temperature, was used as the boron source in JT-60U, but diborane gas is planned to be used in JT-60SA, as in other facilities such as ITER. We should design the boronization system with a safety mechanism based on laws and technical regulations because diborane gas is highly toxic and explosive.
The planned new system for boronization consists of a gas supply system, an exhaust system, and diborane gas removal equipment. The supply system introduces diborane gas into the vacuum vessel. The exhaust system is connected to the existing main exhaust system of the vacuum vessel and is a dedicated system for vacuum evacuation during the boronization process. The pressure inside the vacuum vessel is maintained for the glow discharge by balancing the supply and exhaust flow rates. By increasing the boron gas ratio in the supply gas, in-situ boronization is performed.
In this presentation, the design of the JT-60SA boronization system will be described.

#13973

A kinetic equilibrium reconstruction workflow for ITER using model-based estimation of internal plasma profiles

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

This work presents a Kinetic Equilibrium Reconstruction (KER) framework applied to synthetic measurement data from ITER scenarios, coupling model-based state estimation with magnetic equilibrium reconstruction, using an Extended Kalman Filter (EKF) to improve the inference of internal plasma profiles. The EKF allows inference of quantities such as safety factor and ion temperature in the absence of direct measurements. Standard magnetic equilibrium reconstructions (MER), relying exclusively on external magnetic measurements, exhibit weak sensitivity to internal profile variations and often fail to capture relevant physical features, like the pedestal current density peak in ITER H-mode scenarios. To address these limitations, we apply a KER framework, which extends MER by incorporating internal constraints derived from transport modelling and kinetic profile measurements. The RAPTOR EKF combines heat, particle and current diffusion equations with noisy Thomson-scattering measurements, providing dynamically consistent estimates of density, temperature, current density and related magnetic quantities [S. Van Mulders et al 2025 Nucl. Fusion 65 066006]. The reconstructed profiles are parametrized using spline basis functions and supplied as constraints to the LIUQE equilibrium reconstruction code, enabling a coupled RAPTOR–EKF/LIUQE workflow [S. Van Mulders et al 2026 Nucl. Fusion 66 026026]. Both codes operate as actors within the IMAS (Integrated Modelling Analysis Suite) framework, both for machine description and measurement data. Synthetic diagnostic measurements are generated from ITER free-boundary simulations with the DINA code. The proposed framework improves the inference of internal plasma profiles and the reconstruction of plasma current density, particularly for H-mode plasmas. A statistical analysis over multiple realizations of magnetic measurement noise demonstrates a reduction of up to approximately 50% in the mean square error of key equilibrium quantities, including poloidal beta, internal inductance, on-axis safety factor, peak toroidal current density and internal thermal energy. In addition, the localization of rational magnetic surfaces is improved, with direct implications for tearing-mode control.

Construction of JT-60SA massive gas injection system toward Operation phase 2

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

JT-60SA Massive gas injection system (MGI) is being constructed to research plasma disruption mitigation towards plasma operation in Operation phase 2 in 2026. The MGI is regulated following the rules of the Japanese High Pressure Gas Safety Act, due to its design pressure of 7.9 MPaG and operational pressure of more than 1MPaG. MGI valve installed to vacuum vessel and gas panel for producing mixed gas were designed and manufactured by the Max Planck institute for Plasma Physics (IPP). The interlock system, hydrogen dilution system, cable installation, and piping of MGI are designed and installed on High Pressure Gas Safety Act by QST. This system represents a novelty in the research field of plasma disruption mitigation in large tokamak devices. This presentation shows the design, installation, and operation procedure of the MGI system. The MGI valves are installed onto the stabilizing plate of JT-60SA at port section numbers P9 and P18. The MGI valves will hold a mix of hydrogen or deuterium and noble gases like Argon, Neon, or Helium. Each MGI valve holds a gas mixture of 0.98 liter per each port-section. The MGI valves are normally kept open, so they share the same vacuum as the vacuum vessel. They are closed during gas holding in preparation of their use. The MGI valves will experience baking operation at 200℃ together with the vacuum vessel and the other in-vessel components, to reduce the outgassing to the vacuum vessel. Thermo-couples are installed to each MGI valve to monitor the temperature, so to not exceed the design temperature of 220℃ in baking conditions and 100℃ in operation with interlock system. In the event of overheating during the baking cycle or MGI operation, the interlock system will stop the gas circulation system and will disable the MGI operation.
The injection gas will be prepared by means of pneumatic valves and electrical pressure control valves in a dedicated gas panel installed in the underground of the tokamak hall. To make effective use of hydrogen and deuterium gas cylinders, a pressure booster pump in the gas storage area utilizes compressed air at 0.5 MPaG to supply pressurized hydrogen and deuterium gas to the gas panel. A safety valve is installed to the gas cylinder inlet and the gas panel to prevent overpressure due to incorrect operation.

#13983

Model Predictive Path Integral Control of Tokamak Plasma Boundary Evolution Using a Physics-Informed Neural Equilibrium Surrogate

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Accurate control of plasma boundary shape is essential for reliable tokamak operation and advanced plasma scenario development. The plasma boundary is regulated through the currents in external poloidal-field (PF) coils, which must be adjusted to drive the plasma boundary toward a desired target shape. A plasma-shape controller is proposed in this work based on Model Predictive Path Integral (MPPI) control, a stochastic nonlinear predictive control method that optimizes sequences of coil-control actions over a finite prediction horizon. Because MPPI relies on Monte Carlo trajectory sampling rather than standard iterative optimization, it requires the rapid evaluation of the plasma equilibrium along a large number of candidate control trajectories at every control step, rendering conventional Grad-Shafranov solvers computationally prohibitive. To address this limitation, a physics-informed neural-network (PINN) equilibrium surrogate is introduced to approximate the mapping from PF coil currents to the poloidal flux distribution while enforcing consistency with the Grad-Shafranov equation. By evaluating equilibria two orders of magnitude faster than the original free-boundary solver, the PINN surrogate serves as a critical enabler for real-time trajectory optimization. To ensure the controller accounts for physical hardware constraints, PF coil actuation is modeled using simplified circuit dynamics. Each coil is represented by a first-order RL circuit model, enabling direct voltage-driven actuation. By explicitly incorporating these dynamics into the prediction horizon, the controller directly optimizes the applied voltages rather than relying on idealized lower-level current tracking. The resulting coil currents are used by the surrogate equilibrium model to predict the plasma boundary, which is parameterized by the major radius, minor radius, elongation, and triangularity. By combining stochastic predictive control, a physics-informed equilibrium surrogate, and actuator-consistent PF-coil circuit dynamics, the proposed framework enables computationally efficient and physically consistent predictive control of the tokamak plasma boundary evolution. Importantly, this approach functions as a real-time feedforward control optimization, which reduces the required feedback action and inherently supports graceful degradation of shape control when one or more coil currents or voltages approach their limits. Simulation results demonstrate that the MPPI controller successfully optimizes coil actuation in real time, driving the plasma boundary accurately to its desired target shapes.

Universal Scalable Software Framework for MTCA.4-based Image Acquisition Systems for Plasma Diagnostics

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Cameras are widely used in plasma diagnostic systems, where they provide essential measurements across the infrared and visible spectral ranges. Depending on the application and scale of the diagnostics (e.g. the number of lines of sight), a system may employ from a few to dozens of cameras, often using different technologies and data interfaces.

The rapid development of camera sensors has led to higher spatial resolutions and faster frame rates, significantly enhancing the diagnostic capabilities of imaging systems. However, these improvements also generate substantially larger data volumes. All camera data must therefore be reliably acquired, processed, and archived. Furthermore, in systems that supply real-time measurements for machine protection or plasma control, data acquisition must be performed with minimal latency to ensure timely system response.

These challenges are addressed by the high-performance, scalable image acquisition systems based on the MTCA.4 architecture developed at Lodz University of Technology. The systems support data acquisition modules (frame grabbers) with various camera interfaces—including Camera Link, Camera Link HS, 1 GbE and 10 GbE Vision, and FireWire—which can all be integrated into a single setup. Additionally, the system can be equipped with a time synchronisation module supporting the IEEE 1588 protocol. This enables precise synchronisation and triggering of all devices with nanosecond accuracy.

This contribution presents a software framework for MTCA.4 systems designed to acquire, process, and archive data, as well as configure and control hardware components such as frame grabbers and cameras. The framework offers a user-friendly, unified programming interface that is independent of the underlying hardware, especially the camera interface. Various camera interfaces are supported, and all are represented consistently on the client side. This allows the construction of scalable image acquisition systems that can be easily extended or upgraded—for example, when changing camera models or interfaces.

A key advantage of the framework is its high-performance data acquisition, implemented through efficient DMA transfers. It supports data throughput of up to 10 Gb/s while introducing very low software latency. The framework also provides a dedicated library for efficient management of the camera image stream and its delivery to multiple clients (e.g., for processing, archiving, or visualisation). The MTCA. 4-based DAQ systems and the presented software have been successfully deployed in the divertor protection system of the W7 X stellarator and applied to the ITER reference imaging system and prototype DAQ systems for ITER imaging diagnostics.

Towards real-time impurity transport modeling for tokamaks with RAPTOR

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Oral

Abstract

RAPTOR is a fast tokamak plasma transport code with applications ranging from pulse design and reactor scenario optimization to real-time plasma profiles estimation and simulation. This contribution presents the extension of the RAPTOR simulator, previously capable of solving the coupled dynamics of poloidal flux, electron and ion temperature and electron density [F. Felici et al 2018 Nucl. Fusion 58 096006], with main ion and impurity density transport equations up to three ion species. A tight coupling to the FACIT [P. Maget et al 2020 Plasma Phys. Control. Fusion 62 105001; D Fajardo et al 2022 Plasma Phys. Control. Fusion 64 055017] analytical model for collisional impurity transport is presented. In addition to the numerical implementation, results will be presented for WEST and ASDEX Upgrade discharges, modeling radiative collapse of the plasma core. The resulting impurity radiation and the non-linear feedback with the electron heat transport channel is accounted for with the collisional-radiative ADAS database coefficients previously coupled with RAPTOR [P Maget et al 2022 Plasma Phys. Control. Fusion 64 045016]. Finally we will present an outlook for future steps towards increased model fidelity, notably with respect to turbulent impurity transport, as well as towards real-time control applications.

1-D Kinetic Control Simulations of ITER with Maximization of Ion Cyclotron Power Absorption

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

Future reactor-grade tokamaks that produce fusion power, such as ITER, will require real-time control of the plasma temperature and density. To enable the development and assessment of kinetic control strategies for ITER, the control-friendly, MATLAB©-based COTSIM© (Control-Oriented Transport SIMulator) code has been updated with improved one-dimensional (1-D) transport models [1] that can simulate the entire plasma discharge, including the entry to and exit from burn. In this work, a zero-dimensional (0-D) burn control strategy specifically designed for ITER [2] is evaluated in the 1-D COTSIM© environment. This nonlinear burn control strategy is designed to maximize the plasma's absorption of Ion Cyclotron Resonance Heating (ICRH) power through the injection of helium-3 gas. A key component of this control strategy is the Diagnostic Residual Gas Analyzer (DRGA), which is a specific diagnostic tool planned for ITER operations. Unlike present-day experimental tokamaks, much of the first wall and divertor in future fusion reactors will be covered in blanket modules to maximize tritium breeding and energy conversion. Because the blanket modules will occupy most of this valuable space, diagnostic systems will have much less access to the interior of the vessel for sampling the plasma. This presents a significant challenge for feedback control algorithms that depend on real-time measurement data. The DRGA is designed to circumvent this limitation by accessing only the sub-divertor region. More specifically, the DRGA is capable of providing real-time measurements of particle concentrations, including hydrogen isotopes (the deuterium and tritium fuel) and helium isotopes (the helium-4 fusion product and the injected helium-3 particles), in the sub-divertor region. Tokamak experiments have shown that particle concentrations in the sub-divertor region correlate well with their concentrations in the core-plasma region after a time delay [3], meaning that the DRGA can monitor the composition of the main plasma. With time-delayed measurements of the particle concentrations, the feasibility of the proposed DRGA-enabled control strategy is evaluated in a 1-D simulation environment. By maximizing the plasma's ICRH power absorption through external helium-3 gas injection, the control strategy is shown to substantially increase the fusion power output.

[1] V. Graber and E. Schuster 2025 Fusion Engineering and Design 221 115362
[2] C. C. Klepper, et al 2025 Nuclear Fusion 65 086015
[3] C. C. Klepper, et al 2019 Nuclear Fusion 60 016021

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The EPFL–NVIDIA Fusion Digital Twin: A Modular Framework for Automating Fusion Component Design and Analysis

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Topics

D. Plasma Engineering, Plasma Control, and CODAC

Communication type

Poster

Abstract

The Fusion Digital Twin (FDT) project aims to unify physics simulators, engineering tools, database systems, and photorealistic renderers within a single computational framework to streamline and automate the design workflows of fusion devices. Conceived as an integrated digital environment, FDT enables fusion engineers to carry out coupled physics–engineering simulations, access to experimental and numerical databases, and generate photorealistic visualizations all from one single platform.
The EPFL–NVIDIA FDT project focuses on a digital 2D twin of the TCV divertor. Its architecture is based on a microservices paradigm where each simulator, tool, or renderer is integrated as an independent service. FDT services are orchestrated by NVIDIA Data Federation Mesh (DFM), which manages inter-service inter-resource communication, data transfer, and orchestration. DFM also enables the distributed execution of programmable pipelines directly on database-hosting resources, preventing the transfer of large or sensitive datasets.
A web-based user interface, implemented as an NVIDIA Kit Application, provides access to simulation pipelines and generates photorealistic visuals of fusion systems using CAD models, experimental, and simulation data (NVIDIA RTX). The interface also supports simultaneous simulation and visualization of plasma and system physics.
Six services have been integrated into the FDT framework. DEFUSE manages fusion plasma databases, retrieving experimental or simulated data and storing FDT outputs in an IMAS-compatible structure. MatLab enables the execution of MatLab-based scripts interacting with a headless shared Matlab session. SLURM integration enables job submission, monitoring, and result retrieval from remote clusters. Two Ansys tools—Design Modeler and Mechanical—handle meshing and thermal simulations of fusion components. Plasma heat and particle fluxes are estimated by SOLFS, a surrogate model replicating SOLPS-ITER edge plasma outputs through a principal component decomposition-based Gaussian Process Regressor. Despite being trained on only 112 simulations, SOLFS

accuracy is sufficient for current project objectives. The carbon sputtering service also uses a surrogate model, derived from Molecular Dynamics calculations (machine learning-based interatomic potentials), to estimate carbon fluxes and energy distributions caused by plasma–wall interactions.

Using these six services, FDT replicates a realistic fusion engineering workflow: loading experimental data, inferring plasma fluxes, computing the component thermal response, visualizing results, and storing outcomes. Demonstrations have been performed for three TCV divertor configurations: unbaffled, short inner/long outer baffled, and tightly baffled long legged divertors.

Future developments will extend FDT to full 3D thermo mechanical simulations and incorporate additional physics tools such as ASCOT, Geant4, and CHERAB, enabling comprehensive modeling of particle, mechanical, and radiative loads on a reactor first wall.