Strategy for Considering Uncertainty in Design Process of Future Fusion Power Plant

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

The increasing recognition of the critical impact of uncertainty on designs across various fields highlights the need for a systematic approach to managing this complex aspect in design. Inadequate management of uncertainty can lead to significant design bias and even catastrophic consequences, particularly in complex, innovative, and multidisciplinary.
The successful development of future commercial fusion power plants based on magnetic confinement relies on the effective operation and control of D-T fusion reactions that effectively integrates technical and engineering constraints. The current lack of knowledge in designing and operating a fusion power plant is a significant challenge that is addressed through the use of extrapolations and assumptions in the fields of physics, engineering, and technology. The power plant is made of interconnected systems designed to operate together, each with their own levels of uncertainty. The two main types of uncertainty are: aleatory uncertainty arising from inherent variability in system behaviour, and epistemic uncertainty stemming from incomplete knowledge or modelling limitations. Crucially, the interactions and interdependencies among these systems enable the propagation and even amplification of uncertainty within the plant, posing significant challenges to reliability and safety.
This paper presents the general approach that aims to provide a set of standardized rules and guidelines as an intrinsic feature of the design process to manage uncertainty. In the frame of EU-DEMO development activities, this approach was applied to manage aleatory uncertainty of the magnet system [Zumbolo et al., this conference] and epistemic uncertainty of the plasma system [Bourdelle et al, this conference]. The rules are to be applied consistently throughout the project lifecycle to identify, declare, quantify, process and manage uncertainty throughout the project lifecycle with a particular emphasis on the early design phases where decisions have the greatest long-term impact. The specified systems parameters are always associated with a range which is defined as the sum of the tolerance plus the design margin plus the nuclear safety margin. Integrating margins is crucial to enable the plant to absorb frequent and significant changes during the design phases and ensure robustness against evolving requirements and unforeseen changes. These margins are not arbitrary but are systematically justified based on risk assessment, uncertainty quantification, and probabilistic safety analysis. By integrating uncertainty management as a core design practice, this approach aims to enable clear documentation, enhance decision-making with traceability, reduce risk and improve confidence levels, and accelerate the safe and reliable deployment of future fusion power plant.

DYT: Design Your Tokamak

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

Design Your Tokamak (DYT) is an integrated software framework for physics-driven, system-level tokamak design, enabling feasibility assessment and optimization from a minimal set of defining parameters. DYT couples transport modeling, plasma shaping, magnet system design, and engineering constraints within a unified forward-modeling workflow, supported by a modular architecture and a web-based interface for remote access and collaboration. Each stage of the design process is visualized through interactive diagnostics, including time traces and radial builds, providing transparency and physical intuition.

At the core are the physics modeling capabilities of the ASTRA transport code. An initial ASTRA simulation with simplified geometry and heating deposition models confined plasma profiles and their temporal evolution, including current ramp-up, providing estimates of loop voltage and inductive requirements. The Central Solenoid (CS) and Toroidal Field (TF) coils module performs electromagnetic and structural analyses for multiple material options, automatically defining the radial build and supporting wedged and bucked TF-inboard-leg concepts to optimize flux consumption and discharge duration while preserving mechanical integrity.

The Poloidal Field (PF) coils module supports magnet-system design by rescaling coil sets from existing devices or automatically generating optimized coil distributions based on the TF contour. Plasma shaping is handled by a dedicated module employing a moments-based formalism for time-dependent boundary evolution, allowing controlled variation of elongation and triangularity during ramp-up and flat-top phases while ensuring compatibility with divertor and first-wall constraints.

These elements couple to a heating and current-drive module configuring Neutral Beam Injection (NBI), Electron Cyclotron Resonance Heating (ECRH), and Ion Cyclotron Resonance Heating (ICRH). Beam geometries and power specifications are passed self-consistently to ASTRA to compute heating deposition profiles and current-drive efficiencies. A second ASTRA run evaluates PF coil currents and electromagnetic forces, closing the integrated design loop.

We demonstrate DYT through the conceptual design of ASPECT, a proposed medium-size tokamak for testing reactor-relevant core–edge integration. Design studies for ASPECT, including D-T plasma simulations, indicate the device can reach breakeven under plausible operational assumptions, illustrating how the framework harmonizes engineering constraints with performance objectives. Finally, DYT is released as an open-source educational tool, providing an intuitive, interactive simulator that enables students and newcomers to fusion research to explore design trade-offs and develop a coherent understanding of tokamak physics and engineering.

Challenges in the design of the Atmosphere Detritiation System for UKAEA's Tritium Research Facility

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

The tritium fuel cycle of a fusion facility requires robust systems to purify, store, and safely manage tritium inventories while minimising environmental releases. The UKAEA–Eni H3AT facility will consist of a closed loop tritium fuel cycle simulation plant and experimental laboratory. Within this context, the Atmosphere Detritiation System (ADS) plays a critical safety role by continuously removing and recovering tritium from gaseous effluents prior to discharge to the atmosphere. The ADS for the UKAEA–Eni H3AT Tritium Loop has been designed to process gaseous effluents containing tritium in all foreseen chemical forms arising from multiples subsystems and glovebox environments. The ADS design is further constrained by safety requirements, sub-atmospheric containment, high detritiation performance, system availability, and the need to protect both workers and the public while maintaining releases As Low As Reasonably Practical.

The ADS process design integrates several key components, including recombiners, chiller, blowers, saturator, and scrubber columns, to enable effective tritium recovery. Throughout system development, multiple design challenges were identified, driven primarily by stringent safety requirements, system availability needs, and demanding performance targets. One major challenge involved selecting the appropriate detritiation technology. Scrubber columns were chosen over molecular sieve driers. Further design refinement was required within the scrubber system itself, particularly regarding the choice between parallel or series configuration, in order to identify which arrangement would achieve higher detritiation factors while maintaining operational robustness.

Optimisation of the blower configuration also proved critical. Variable-speed drive blowers were identified as the preferred option, and several configurations were considered to determine which could best maintain sufficient suction across all operation scenarios while effectively accommodating fluctuating flow conditions.

Together, these design choices illustrate the complexity of developing safety-critical tritium systems for future fusion facilities and highlights the importance of integrating safety, performance, and operational flexibility from the early stages of design. This presentation will highlight the principal technical challenges faced during the ADS design phase and the engineering approaches adopted to address them, providing a clear understanding of how the current design was established.

Forced Convection Heat Transfer Experiment with Molten Salt

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Poster

Abstract

The decoupling of the Primary Heat Transfer System (PHTS) from the Power Conversion System (PCS) introduces an Intermediate Heat Transfer System (IHTS) with an Energy Storage System (ESS). Molten salt systems are being considered and investigated for this purpose in the European DEMOnstration Fusion Reactor (DEMO) Helium Cooled Pebble Bed (HCPB) Balance Of Plant (BOP) Indirect Coupling Design (ICD). Within helium-cooled Breeding Blanket (BB) concepts, the different thermophysical parameters and material properties of heat transfer media in both the PHTS and IHTS challenge the design of intermediate heat exchangers (IHX). The validation, adaption and development of correlation equations for the determination of heat transfer coefficient of internal flow with molten salt over a wide range of relevant operating conditions support the design of advanced helium/molten salt heat exchanger by reducing design uncertainties and can contribute substantially to the IHX development of fusion power plants. The present study reports on the forced convection heat transfer experiment with molten salt, composed of alkali nitrites and nitrates (KNO_3 (53%), $NaNO_2$ (40%) and $NaNO_3$ (7%)), in circular tubes salt under fusion power plant BOP thermal-hydraulic operating conditions in the HELOKA-US test facility. The construction, instrumentation and mechanical design of the experimental test section are described in detail. Particular emphasis is placed on the instrumentation strategy adopted to achieve high measurement accuracy. A comprehensive layout of temperature measurement concept for both the wall and fluid temperature measurements is presented, along with calibration procedures and uncertainty analysis.

PPPL Alternate Structural Studies of the Tokamak Energy ST-E1 Reactor

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

Tokamak Energy's ST-E1 pre-concept Fusion Power Plant design is being developed under the DOE' Milestone Fusion Pilot Plant Program. It addresses the practical needs of a commercial power plant. A proposed layout of the machine has been chosen. The design is based on experience with Spherical Tokamak (ST) physics. Other major influences on the configuration include realistic shielding, servicing based on radial extraction of core components, and highly reliable "permanent" REBCO HTS coils designed to avoid quench and requiring no joints. This paper addresses modest alternatives to the point design. This work was done at PPPL during the pre-conceptual design phase of the TE ST-E1 project. It includes benchmarks of TE and PPPL modeling.

The TE ST-E1 reactor shares some structural characteristics with the Fusion Nuclear Science Facility (FNSF) , primarily due to the choice of radial servicing. ST-E1 is initially pulsed, targeting steady state operations. It has a solenoid which produces global twist and local out-of-plane (OOP) loading.

The alternates studied include, 18 and 12 coil configurations, and the addition of a "plug" in the solenoid bore to help support the proposed bucked and wedged solution. Vertical and toroidal interactions between the TF and OH were studied. A tierod was added to offset some of the tensile stresses. The "square" TF profile vs. a "D' profile was found acceptable. Corner radii were studied. There were minimal disadvantages of the 2 m radius corner for in -plane loads. The larger 4 meter radius showed advantages. Added structures are needed to support the out-of-plane loads. Space is limited for inter-coil structures due to the necessity for wide openings for radial servicing. "Covers" top and bottom were added, inter-coil structures were added at the inner TF corner radii, and removeable outer bracing was suggested.

Inner leg torsional shear will need features to improve the shear capacity of the wedged faces. High friction material treatments were suggested. Shear keys in the corner and possibly "ITER like" corner tensioned rings. The need for tensioned rings to avoid de-wedging in the TF inner corner The need for rings was compared with other reactors with different ratios of outer leg radius to inner leg radius.

Means of increasing the available volt seconds were investigated. Increasing the solenoid current and increasing the solenoid radius, giving up some TF territory , were found possible.

Toward an Integrated Magnetic Cage for the Volumetric Neutron Source (VNS): Physics-Driven Design and Enabling Technologies

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

The Volumetric Neutron Source (VNS) is being developed as a steady-state, beam-driven tokamak to provide DEMO-relevant neutron irradiation conditions for the integrated qualification of fusion reactor components. Within this framework, the magnetic system plays a central role in determining plasma controllability, machine compactness, neutron shielding effectiveness, and overall facility feasibility. As VNS progresses toward its conceptual design phase, the selection of a robust and high-performance magnetic cage architecture has become a key design driver.

This work reviews the evolution of the VNS magnet system concept, focusing on the comparative assessment of the most promising poloidal and toroidal field coil configurations. Two main magnetic cage options are discussed: a conventional PF-ex-TF configuration and an innovative PF-in-TF solution, in which the poloidal field coils are placed around the Vacuum Vessel, closer to the plasma. Systematic optimization studies of the PF system demonstrate that the PF-in-TF architecture enables a substantial reduction in PF currents and stored magnetic energy, significantly improving plasma equilibrium and controllability in a compact, heavily shielded device. The presentation further highlights the enabling technologies required to realize this option, including the use of high-temperature superconductors (HTS), flexible dry conductors, and in-situ winding machine for the TF or the PF coil systems. The assembly strategy represents here a major departure in approach from conventional tokamaks. Structural, electromagnetic, thermal, and neutronic considerations are addressed to illustrate how the proposed solutions balance performance gains against manufacturing and assembly complexity.

The paper concludes by outlining the rationale for converging toward a reference concept for VNS and identifies the remaining design challenges and qualification needs on the path to a credible and scalable conceptual design, positioning VNS as a candidate concept for a first Fusion Pilot Plant (FPP). Besides, the design options explored here, most notably the in-situ TF coil winding approach, are conceived with scalability to DEMO-size devices.

#12909

Design and R&D achievements for the Balance of Plant and Tokamak Coolants System of the European DEMOnstration Fusion Power Plant

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

The European DEMOnstration Fusion Power Plant (DEMO) is a tokamak currently under conceptual design in Europe as a possible First-Of-A-Kind fusion power plant. It has the main goal of demonstrating the feasibility of producing electrical power from fusion by the end of the century. In the context of this project, its Balance of Plant (BoP) comprises the Intermediate Heat Transport System (IHTS) – when applicable – equipped by an Energy Storage System (ESS), and the Power Conversion System (PCS), which receives heat from the Tokamak Coolants System (TCS).

This article deals with the concept design of the Balance of Plant systems carried out under the Fusion Programme 9 (FP9) until 2025 for both breeding blanket (BB) concepts under assessment, namely the Water Cooled Lithium Lead (WCLL) BB and the Helium Cooled Pebble Bed (HCPB) BB. It provides readers with a brief description of the conceptual designs of the different BOP variants investigated, focusing the attention on the PCS and reporting on the feasibility and maturity level achieved. Furthermore, this paper reports the main progresses in the R&D activities performed on the most critical TCS and BOP technologies to be operated under “pulsed” regime in the fusion power plant: the DEMO steam generator; the steam turbine of the water cooled BoP and the heat exchangers of the helium cooled BoP.

The achievements described in this paper are the results of several design and R&D tasks, successfully carried out during 2021-2026 period, according to the WPBOP work plans and in close synergy with DEMO WPDES.

#12945

Multiphysics systems simulation for rapid breeder blanket concept analysis

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

A performant and reliable breeder blanket system is fundamental to the commercialisation of fusion energy. These complex multifunctional systems remain at low technology readiness level. As breeder blankets systems are integrated into power plant designs and subsequent rig testing programmes, it is essential that design progression is guided by quantitative understanding of the inevitable trade-offs. In this work, an integrated and multiphysics approach for breeder blanket concept analysis has been developed. The foundation of this is a systems modelling library, ARTEMIS, which provides base modelling elements (conductors, pipes, beams), implementing 0D and quasi-1D models which cover thermal, mechanical, hydraulic and fuel cycle domains. A workflow engine can then be used to integrate with existing codes, such as OpenMC for neutronics analysis and plant level codes such as PROCESS and BLUEMIRA. This system presents a flexible, extensible and impactful approach to assess breeder blanket concepts.

The systems modelling library, ARTEMIS, has been rigorously benchmarked against higher order analysis for breeder blanket applications. This will be presented alongside initial studies which demonstrate the impact of completing assessments such as LOCA scenarios and tritium permeation/retention modelling at a concept phase. The route to assess breeder blankets alongside plant concepts will also be shown, demonstrating the two-way influence of design decisions. The current capability is focused on optimisation of conceptual blanket designs; this methodology will continue to be developed to include progressively higher fidelity simulations, with the next stage incorporating parametric 3D CAD, and reliability analysis through probabilistic simulation. This framework will provide a path to continual performance verification of the blanket system whilst increasing the accuracy and confidence in assessments.

Experimental investigation of DEMO relevant HCPB ICD operating scenarios using the new HELOKA-US molten salt facility at KIT

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

The HELOKA-US experimental facility has been successfully constructed and commissioned at KIT in 2025. It is designed to investigate the functionality and operability of an Intermediate Heat Transfer System (IHTS) using molten salt as heat transfer fluid for an architecture typical of the European DEMOnstration Fusion Reactor (DEMO) Helium-Cooled Pebble Bed (HCBP) Balance of Plant (BOP) Indirect Coupling Design (ICD) concept. The test facility includes a 260 kW electrical heater for representing the heat load coming from the Breeding Blanket (BB) Primary Heat Transfer System (PHTS), a two-tank Energy Storage System (ESS) and a molten salt/water heat exchanger as a heat sink simulating the steam generator of the Power Conversion System (PCS). As in the DEMO HCPB BOP ICD concept, the molten salt used in HELOKA-US is a mixture of alkali nitrite and alkali nitrates with a composition of KNO3 (53 %), NaNO2 (40 %) and NaNO3 (7 %), commonly known as HITEC salt. This salt can be used in a temperature range between ~142 °C to 550 °C. HELOKA-US operates at the DEMO IHTS relevant operational conditions.

The first experiments performed in HELOKA-US focused on the qualification of the effectiveness of the thermal storage concept with a pulsating power source with typical DEMO time profiles and on data collection for system code validation. The experimental campaign comprises steady-state operation and smooth transitions between steady-state operation. Similar sequences to those in the DEMO HCPB BOP concept were realised, with multiple cycles of ramp-up to nominal power, a 2-hour pulse mode, ramp-down to dwell mode and a 10-minute dwell mode.

This paper reports on the definition and description of operating scenarios in HELOKA-US representing DEMO operational states followed by a detailed analysis of the experimental data obtained during this test campaign, and the verification of reproducible results. The paper concludes with statements on the overall performance of the loop and its interactions and interfaces with connected systems, and provides an outlook on the capabilities of alternative testing and qualification campaigns beyond the original scope.

Negative triangularity and low aspect ratio for compact high-field fusion pilot plants

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

The development of compact fusion power plants is essential to minimize capital costs and accelerate the commercialization of fusion energy. Achieving high power density at a reduced reactor scale requires a robust technological and physical framework. Within this context, High Temperature Superconductors (HTS) emerge as a critical enabler, allowing for significantly higher on-axis magnetic fields and current densities. This technological advancement enables a high-field central column even under the strict spatial constraints of compact reactor geometries, providing the necessary confinement to reach fusion gain in a smaller device.

However, the high-power density enabled by high-field operation magnifies the challenges of power exhaust, plasma stability, and radiation protection. This study proposes that the integration of Negative Triangularity (NT) and Low Aspect Ratio (LAR) configurations offers an effective approach to these bottlenecks [1-2]. While the LAR design provides an efficient path to high-pressure operation, NT provides the physics basis necessary to manage the resulting heat loads and neutronics requirements. The combination of NT-LAR benefits from:

1. ELM-free operation: achieving H-mode-level core confinement within a robust L mode-like edge, thereby eliminating the risk of transient heat pulses to the plasma facing components.
2. Increased divertor wetted area: the geometric nature of NT places the X-point at a larger radius and allows for more efficient, outboard-located divertor structures, which are easier to implement in a compact device.
3. Enhanced stability: combined enhancements in MHD stability limits that permit reliable operation at the high plasma pressures required for a pilot plant.
4. Neutronic volume optimization: by shifting the plasma volume toward the outboard side, the NT configuration maximizes the available space for neutronics in the inboard, where space is highly constrained.

In this work, we present the results of integrated modelling for a pilot plant ($Q > 5$) and a fusion power plant ($Q > 25$) based on this concept. The operational points for these machines, identified through an optimization of physical and technological constraints, are introduced and analyzed, demonstrating the viability of the high-field NT-LAR architecture as a feasible pathway for commercial fusion energy.

[1] D. J. Cruz-Zabala et al., NF 2024
[2] F. Puentes-del-Pozo et al, in preparation.

The STAR fusion power plant design completes a pre-conceptual design phase millstone

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

The STAR fusion power plant design completes a
pre-conceptual design phase millstone*

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An architectural layout defining the overall machine framework of the Spherical Tokamak Advanced Reactor (STAR) fusion power plant provides an arrangement of critical components, interfaces and planned maintenance strategies to complete a required pre-conceptual design phase millstone. The evolution of the STAR design has been documented in past conferences [1 thru 5] where the device general arrangement, assembly approach and definition of critical configuration components and services have been established within underlining physics parameters with a design goal striving to come close to performance parameters of a Gen III AP1000 fission power plant.

Current fusion pilot plant and DEMO design topologies are heavily focused on innovative maintenance strategies to manage the replacement of neutron degraded breeding blankets, with three approaches emerging as the primary design strategies:
1. Incorporating horizontal maintenance
2. Incorporating vertical maintenance
3. Incorporating demountable HTS Coils

Following a horizontal maintenance scheme requires locating TF coils outer legs at a larger radius to gain space to install larger blanket sectors, an added corridor around the device to move blankets to and from the maintenance building with horizontal movement impacting placement of diagnostics and services traditionally located at the device midplane. Increasing TF coil size also can lead to on-site TF coil fabrication, negating a lower cost offsite fabrication approach. Following a vertical maintenance option through small vertical ports with piping services introduces its own issues associated with piping removal to gain blanket access; high-risk remote handling making removal, insertion, and alignment challenging. Incorporating demountable magnets can gain full blanket access, but this approach involves engineering, operating, and safety risks related to thermal cycling, joint mechanical stresses and ohmic heating of joint interfaces.
To be clear, defining an economically viable tokamak fusion power plant design is exceptionally difficult with outstanding issues associated with all emerging next step design strategies.
Progress reported in earlier STAR papers did not fully cover the concept details defining how blanket sectors located under TF coils are installed/removed, the crane-accessed remote maintenance strategy and planned basement arrangement of piping services that feed the majority of the device core components. Progress made in these areas will be highlighted in this paper along with concluding remarks suggesting other design and technology advancements that could enhance performance and lower costs.

#13018

A validated fast predictive model for disruption heat loads and vapour shielding

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

Tokamak plasmas are prone to discharge-terminating relaxation phenomena, known as disruptions. During such short transients, the peak thermal and magnetic energy stored in the plasma are released to plasma facing components (PFC) in tens to hundreds of milliseconds, potentially leading to damage. Scenario mitigation strategies must rely on predictive advanced nonlinear models, which are however typically complex to use and numerically expensive.

Fast and reliable models play a key role in the risk assessment of viable tokamak scenarios, where a high compute speed justifies lower accuracy. A validated and fast semi-analytical module is presented in this study, implementing simple physical assumptions, namely multi-device scaling laws and zero-dimensional analytical models. Only relying on well defined equilibrium physical quantities, our module predicts time scales and heat fluxes during eight classes of disruptions, both mitigated and unmitigated, along with providing an estimate for relative tolerances. Wall melting risk is assessed, also considering the shielding action where extreme heat fluxes may locally vaporise the solid tungsten surfaces: a specific module evaluates refracted loads to the walls and reflected radiation to other PFCs. The vapour shielding model is based on state-of-art advanced models and includes several physics-based correction parameters (e.g., vapour cloud opacity, prompt redeposition of vapourised material, diffusion rate of the ionised cloud along magnetic field lines…).

A validation against published experimental JET disruption data is made, finding promising agreement on a range of key parameters.

Optimised for use on next-generation spherical tokamaks, our model has already found application for STEP (Spherical Tokamak for Energy Production): presently envisaged scenario points have been mostly confirmed to preserve first wall integrity, with some concern for sacrificial limiters. Vapour shielding action is found to be crucial in limiting the plasma-wall interaction, and our results shed light on the real benefit of active mitigation for some classes of disruptions.

Our model is applicable to all tokamaks, not just the spherical tokamak, making it a generally useful tool.

This work has been funded by STEP Fusion, a major technology and infrastructure programme led by UK Industrial Fusion Solutions Ltd (UKIFS), which aims to deliver the UK's prototype fusion powerplant and a path to the commercial viability of fusion. To obtain further information on the data and models underlying this paper please contact PublicationsManager@ukaea.uk.

STEP Fusion powerplant vacuum vessel and in-vessel systems design approach for managing radiation loads during steady state operations

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

The STEP Fusion programme aims to deliver a UK prototype fusion energy plant, targeting 2040 and a path to commercial viability. The proposed STEP Fusion Powerplant concept is a steady-state D-T operating spherical tokamak; this imposes specific radiation considerations which need to be included in the design approach for the Vacuum vessel and In-Vessel Systems (VIVS).

VIVS are required to handle the heat & particle loads from the plasma, and create a suitable vacuum environment as with current short-pulsed experimental fusion devices. However, the combination of high neutron and gamma radiation fluxes from a D-T plasma multiplied by long commercially-relevant operating durations and component lifetimes results in requiring VIVS to withstand high radiation fluences as well. Sensitive systems within the tokamak, such superconducting magnets and plasma diagnostics, have low tolerances to radiation damage. VIVS need to provide sufficient radiation shielding to protect these systems. This shielding creates additional space requirements and design integration complexity. The radiation shielding is a major driver for the overall radial build and tokamak size, particularly with a spherical tokamak geometry.

The components with VIVS (plasma facing components, radiation shields, breeder blankets & vacuum vessel) will witness high radiation doses and will undergo various radiation-induced degradation effects such as embrittlement, loss of thermal conductivity, swelling, corrosion and transmutation. The high radiation fluxes also generate nuclear heat loads within the VIVS. The radiation fluxes and nuclear heating exponentially decrease with distance from the plasma. The design approach needs to include these nuclear heat loads and consider their variation through the VIVS. The high radiation fluxes will cause nuclear activation which means the VIVS will have still have residual nuclear heating and emit radiation during shutdown and maintenance phases. Additionally, coolants will become activated and transport radiation throughout the supporting pipework and plant equipment outside the tokamak.

The STEP Fusion powerplant tritium breeder blankets are situated on only the outboard of the tokamak. However, the tritium breeding performance is not just determined by the outboard, but by the overall VIVS nuclear design. Neutron absorption and scattering in the inboard and exhaust regions has a significant effect on the number of neutrons reaching the outboard and the resulting fuel self-sufficiency, especially with a spherical tokamak geometry.

Here, we will present the STEP Fusion powerplant VIVS concept design and how the design approach taken incorporates radiation shielding, material degradation, nuclear heating, activation and holistic nuclear design during steady-state operation.

#13046

Systems analysis of reactor-scale stellarators with optimised plasma–coil spacing

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

In our investigations of stellarator power plant concepts, we found that the plasma-coil distance is one of the key elements in achieving a more compact design, which was previously overlooked in system level studies. For the first time, we coupled coil optimisation with systems analysis to achieve a best-fit coil set that overall reduces machine size.

This study evaluates the feasibility of a stellarator power plant using a new Stable Quasi-Isodynamic Designs (SQuID) configuration within the PROCESS systems code. We explore the system-level challenges associated with building a compact power plant based on a stellarator. By incorporating updated stellarator modules into the PROCESS code, we demonstrate that selecting the optimal minimal plasma-coil distance during coil set optimization is key to achieving the best machine design.

For a target electric power of 1 GW, we analyze the impact of the confinement renormalization factor, neutron wall load limits, and forces acting on the coils. Our findings indicate that, under conservative assumptions, the neutron wall load limit becomes the dominant constraint. To achieve a machine with a major radius smaller than 17 meters, neutron limits would need to be significantly relaxed compared to those of DEMO.

The research also includes an economic analysis, comparing the performance of low-temperature superconducting (LTS) and high-temperature superconducting (HTS) magnetic systems, and discusses the competitiveness of both technologies from the perspective of a commercial power plant.

Electrical Infrastructure Design Challenges for the Steady-State Operation of STEP Prototype Plant

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

As efforts toward fusion commercialisation continue, the supporting electrical infrastructure must evolve from powering pulsed experimental machines to enabling reliable and continuous operation. STEP (Spherical Tokamak for Energy Production), the UK's first prototype fusion powerplant (SPP), aims to demonstrate net electricity delivery to the grid and therefore demands an electrical infrastructure fundamentally different from that used in pulsed experimental machines. This paper presents the developing design framework for the SPP's electrical network and outlines the strategies required to support reliable steady-state operation.

A key challenge is maintaining stable and compliant interaction with the National Grid. Historically, pulsed experimental machines have imposed only brief disturbances lasting a few seconds. In contrast, a powerplant device such as STEP will introduce power variations and transients that impact the grid for longer periods of time. The SPP's electrical infrastructure design therefore incorporates approaches to meet power variation limits, stay within allowable power factor range, and provide voltage support across all operating scenarios. These measures are essential for ensuring full Grid Code compliance during operation.

High availability and reliability are equally critical for a commercial scale fusion plant. SPP's electrical systems must support high operational availability through robust redundancy strategies, identifying critical systems that require enhanced resilience to internal or external electrical faults. Efficiency is also important in a commercial or prototype powerplant as it directly influences achievable net electrical output and overall plant performance, driving the need for improved power supply designs for major systems such as plasma heating and current drive, alongside optimised power management across all other plant subsystems.

These challenges must be addressed while avoiding unnecessary complexity, as commercial viability and scalability are fundamental goals of the STEP programme.

This paper sets out the design framework for the SPP's electrical infrastructure, focusing on grid compliance and integrating availability, reliability and redundancy requirements to enable safe and stable steady-state operation. The outcomes of this work will inform the electrical architectures of future fusion facilities and contribute to the design of subsequent generations of fusion powerplants.

This work has been funded by STEP, a major technology and infrastructure programme led by UK Industrial Fusion Solutions Ltd (UKIFS), which aims to deliver the UK's prototype fusion powerplant and a path to commercial viability of fusion.

Status of the EU-DEMO WCLL BOP variants and provisional selection

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

The European DEMOnstration Fusion Power Plant (EU-DEMO) represents the European tokamak currently under design as a possible First-Of-A-Kind (FOAK) fusion power plant. It has the main goal of demonstrating the feasibility of producing electrical power, and in the framework of this project, the Balance of Plant (BoP) is one of the most relevant topics being directly involved in the conversion of pulsed thermal energy in a continuous power output.
Three main BoP lines are being investigated under the EUROfusion Framework Programme 9 (FP9), each for a specific Breeding Blanket (BB) technology: the Water-Cooled Lithium Lead (WCLL), the Water-cooled Lead-Ceramic Breeder (WLCB), and the Helium Cooled Pebble Bed (HCPB).
This article provides an overview of the three WCLL BoP variants investigated during FP9: (i) the Direct Coupling Design (DCD) with Small Energy Storage System (ESS); (ii) the DCD without ESS (usually referred as “No Storage”); and (iii) the Indirect Coupling Design (ICD) with Small ESS. The simultaneous development of these three configurations was motivated by the specific advantages of each variant. The design activities carried out in the 2021-2025 timeframe reached a milestone at the end of 2025, when a provisional selection was performed to focus the future work only on the most promising option.
This work summarizes the rationale guiding the activities carried out within the frame of the EUROfusion BoP Working Package (WP), and the status of each configuration as of December 25. The criteria and metrics adopted for the provisional selection are also

presented, including compliance with EU-DEMO high-level requirements, plant efficiency, components size and footprint, cost, safety aspects, FMEA and RBD analyses, operational modes and plant control, constructability, commissioning and inspection, assembly and dismantling and technology maturation level.
The DCD with Small ESS layout was identified as the preferred solution, as it provides the best overall balance across the criteria previously mentioned. The other two layouts failed in matching the high-level requirements (No Storage) or showed excessive spatial requirements and associated costs (ICD with Small ESS).

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BRIDGE - a Multiscale and Multiphysics Approach to the Development of a High-Fidelity Physics Plasma Simulator for Burning Plasma

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

The BRIDGE project addresses current gaps in modelling a magnetically confined plasma in experimental conditions, bringing together models for actuators, observables and plasma response, in a multiphysics and multiscale environment. BRIDGE is designed around IMAS and applies principles of modern code architecture. Development is guided by assessing software components, using a modified version of NASA's TRLs, combined with System Readiness Assessment for integration of software components. In practice, the simulator as a complex system is broken down into sub-systems that need to be coupled self-consistently with each other to provide a realistic representation of the system dynamics. Here realistic means as close as possible to the real plasma-plant system, within the uncertainties of the models used. BRIDGE can therefore be seen as a Digital Twin for a tokamak that allows a hierarchy of fidelity in the models. The plasma response to actuators is calculated in an environment where models for controllers use simulated signals from diagnostics. While still a physics-based model, the simulations can be used for assessing the time scales of the plasma response that might affect the design of control algorithms. Similarly, the model for observables is used in combination with the plasma response to assess uncertainties in the processed data that are input to the controllers and affect the controllability of a plasma discharge. Two applications are discussed as demonstration of coupled simulations. The first is the modelling of an ITER plasma for the Start of Research Operation, with focus on the assessment of the minimum Electron Cyclotron (EC) power to sustain an L-mode plasma. This application combines a physics-based predictive model for the core transport, a reduced model for the Scrape-Off-Layer, an equilibrium solver, and feedback control on the EC power and deposition. The second application focusses on an assessment of uncertainties in the data analysis from the interferometer/polarimeter system. The line integrated density and Faraday rotation calculated directly from profiles and equilibrium of a simulated scenario are compared with calculations where these quantities are replaced with synthetic diagnostic signals. Since the interferometer analysis during ITER operation will use channels from the polarimeter to correct fringe jumps, and will infer electron density from magnetic flux measurements (or equilibrium reconstruction) and from electron temperature measurements with the ECE, verifying and validating these calculations enables step-by-step assessment of uncertainties in diagnostics data processing, by combining forward and backward models, to support the development of the data analysis toolchain.

#13149

From Perfection to Pragmatism: A FOAK-Oriented Strategy for Materials and Component Qualification in Fusion Power Plants»

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

While major progress has been achieved in plasma physics and reactor concepts, the timeline toward fusion power plants is increasingly dominated by conservative, pre-deployment qualification strategies for materials and components. In particular, the expectation that near-final performance must be demonstrated prior to deployment has become a pacing item for fusion technology. This contribution argues for a paradigm shift toward structured pragmatism, in which first-of-a-kind (FOAK) fusion plants are explicitly used as platforms for generating in service qualification evidence for materials and components

From a materials and components perspective, several key subsystems have already reached Technology Readiness Level (TRL) 4-6 in relevant environments. These include high-heat-flux tungsten divertor targets, tungsten first-wall armour, structural steels such as Grade 91 and reduced activation ferritic-martensitic (RAFM) variants, biphasic lithium ceramic breeder materials, tritium extraction systems, and liquid-metal neutron multipliers. For each of these elements, licensable baseline solutions exist, with clearly defined operating windows, known degradation mechanisms, and established manufacturing routes. Recent global TRL assessments confirm that the transition to TRL 7–8 cannot be achieved in laboratories or test facilities alone but requires integrated operation under reactor-relevant conditions. As well as TRL, it is also important to consider the likely availability of industrial-scale supply chains for any material or manufacturing process on the proposed timescale of a given project.

We propose a FOAK-oriented, owner-led qualification framework combining (i) bounded baseline materials and component selections, (ii) embedded in-service surveillance including witness samples and diagnostics, and (iii) staged power ascents governed by component-level key performance indicators such as heat-flux resilience, structural lifetime, tritium breeding margin, and availability. Importantly, this framework explicitly addresses component interfaces, manufacturing tolerances, joining technologies, and maintainability, which are often underrepresented in pre-deployment qualification but dominate real plant performance and availability. This approach aligns with historical practice in early fission, aerospace, and other large-scale energy technologies, where operational feedback accelerated standardization, codification and industrialization.

Adopting this structured pragmatism in materials and components development reduces schedule risk, better uncertainty quantification on performance, accelerates learning, and enables earlier engagement of industrial supply chains for steels, tungsten components, breeder materials, and lithium enrichment. By shifting the focus attempts to achieve full qualification before operation to managed in-service qualification, fusion technology can transition more rapidly from experimental devices to deployable power-plant components within societally relevant timeframes.

#13204

Using PROCESS and BLUEMIRA as training tools for Fusion Power Plant Design

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

There is demand for engaging, accessible materials for upskilling staff new to fusion power plant (FPP) design or coming to fusion engineering from different research backgrounds. Currently, large workforces are required for FPP design, but our open-source systems codes PROCESS and BLUEMIRA make it possible for small teams and individuals to find starting points for concept designs in minutes that would typically take months to achieve. This makes them the ideal
apparatus for fast, easy teaching of our systems codes for achieving their own designs and the key ideas and technical knowledge needed for understanding fusion.

We lay out the tools to give the user an educational workflow starting from the initial optimization of a fusion reactor design in PROCESS, to using that output in BLUEMIRA for full FPP realisation; including CAD models of the reactor, and accounting for balance of plant and fuel cycle models. We aim to teach users not just how to execute the code, but core concepts needed to quickly grasp the importance of key inputs and the impact they have on the success, cost and efficiency of FPPs. Our training materials, applied in Jupyter notebooks, encourage active engagement through interaction with deliberate erring, applied user choices and diagnostic evaluation - demonstrating how PROCESS and BLUEMIRA can be used for problem-solving in reactor design. Users will be able
to interact with the course through their browser, removing the requirement for installation of PROCESS and BLUEMIRA to reduce barriers to entry.

We discuss the challenges in crafting this course such as; ensuring easy access while balancing the technical aspects of protecting the codebase, and being engaging while not requiring high-levels of prerequisite knowledge. This resource can be suggested in isolation to new starters within teams, or as an activity as part of a longer program. We will be showcasing a mini-demo for conference participants.

As well as upskilling individuals, the business benefits of having these quick and easy courses are many. Education in these areas can improve staff retention, upskilling the fusion industrial supply chain, accelerate learning for graduates and PhD students, and encourage “career migrants” from other fields to bring new ideas, perspectives and skills to the fusion workforce. This will also be essential for the advent of public-private partnerships globally, particularly in the US and EU.

#13312

Disentangling the aspect ratio dependence of energy confinement time with multi-machine integrated modeling

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

Energy confinement time predictions used for designing future fusion power plants rely on empirical scaling laws derived from multi-machine databases. These scalings suffer from strong correlations among engineering variables and uncertainties on fitted exponents, which limits the robustness of extrapolations beyond today's operating domain [1,2]. In particular, geometric effects may be misattributed to correlated parameters, leading to an under- or over-estimated role of plasma shaping and aspect ratio. In parallel, recent progress in high-fidelity integrated modeling has shown that predictive tokamak simulations can reproduce major confinement trends with plasma current, toroidal field and heating power [3,4], opening a pathway to use validated simulations to disentangle correlated dependencies and better constrain confinement predictions.

This study addresses the impact of aspect ratio on global energy confinement for
predominantly electron-heated, low-β, L-mode plasmas. The IMAS-coupled integrated modeling suite High Fidelity Pulse Simulator (HFPS) is applied in a predictive manner to reproduce steady or quasi-stationary phases of well-diagnosed discharges from complementary devices spanning a
wide range of aspect ratios, from conventional tokamaks (WEST, ASDEX Upgrade) to a spherical tokamak (MAST Upgrade). Electron and ion temperature and density profiles are systematically validated against available diagnostics to establish confidence bounds on the resulting energy confinement time predictions.

Building on this validated dataset, controlled numerical experiments are then designed to isolate dependencies. Following previous experimental approaches [5], scans are performed not only at fixed engineering variables but also under constraints on key dimensionless parameters governing transport. The resulting simulation database will be used to identify the mechanisms linking aspect ratio to transport and to provide refined confinement scalings with explicit geometric dependencies.

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Advanced Neutronics Analysis Framework for the Helical Fusion Reactor Helix KANATA: Efficient Tetrahedral Mesh Modeling, Variance Reduction, and Integrated 3D Visualization Using PHITS

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

This paper presents a comprehensive neutronics analysis framework developed for Helix KANATA, a helical fusion reactor being designed by Helical Fusion Co., Ltd. in Japan. Building upon our previous work presented at SOFT 2024, which introduced tetrahedral mesh modeling techniques for complex helical geometries using the PHITS Monte Carlo particle transport code, we report significant advancements in preprocessing efficiency, computational accuracy through variance reduction, activation analysis, and three-dimensional visualization capabilities.

A critical challenge in Monte Carlo neutron transport calculations for helical fusion reactors lies in faithfully reproducing complex helical coil and blanket structures within the computational geometry. We have established an advanced workflow employing Altair SimLab for surface-to-volume meshing of 3D-CAD design data to generate high-quality Nastran bulk data files (.bdf) compatible with PHITS. Through refined meshing techniques, including uniform mesh aspect ratio control during the surface meshing stage and supplementary mesh repair using Polygonalmeister (UEL Co.), we achieved a remarkable 75\% reduction in the CAD-to-computation-ready model preparation time compared to the previous FPP design, reducing it from approximately six months to only 1.5 months. The resulting homogeneous tetrahedral mesh quality also eliminated the computational instabilities previously encountered during long-duration parallel calculations.

To address the statistical challenges inherent in neutron transport through large-scale helical structures, we implemented the Weight Window Generator (WWG), a variance reduction technique native to PHITS. Our results demonstrate that calculations with WWG using 100 million histories yield superior statistical accuracy compared to conventional calculations without WWG using one billion histories, achieving a dramatic improvement in computational efficiency while maintaining high precision across the entire reactor geometry, including deep-penetration regions such as the biological shield.

Furthermore, we have established a coupled calculation workflow integrating PHITS with dchain-sp for comprehensive activation analysis of blanket materials, enabling evaluation of radioactive nuclide production and subsequent gamma-ray dose rates following reactor shutdown. For three-dimensional visualization of neutron flux distributions, we developed an integrated pipeline combining phig3d (recently updated to support tetrahedral mesh rendering), ParaView for detailed quantitative analysis, and Blender with the Cycles rendering engine for photorealistic output. This multi-tool visualization approach enables both rapid verification and publication-quality representations of neutronics results mapped onto the actual reactor geometry.

These developments collectively establish a robust and efficient neutronics evaluation capability essential for the ongoing design

optimization of the Helix KANATA helical fusion reactor, positioning Helical Fusion competitively alongside international stellarator-type reactor development programs.

Enabling knowledge extraction and remote data analysis in large research facilities: a use case

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

Nowadays, scientific facilities produce a vast amount of data during each experiment. To maximise the knowledge extraction from these data, it is essential to access them in an efficient and user-friendly manner so that they can be properly processed and analysed. The scientific community can then leverage these data to advance research in areas such as fusion technologies or particle physics. A notable example, and the specific use case addressed in this work, is IFMIF-DONES (International Fusion Materials Irradiation Facility -DEMO Oriented Neutron Source), in which a variety of experiments will be conducted to identify materials capable of withstanding the extreme temperature and radiation conditions present inside fusion reactors. However, accessing these data can be challenging or even unfeasible for many scientists or institutions due to the complexity of data governance. To maximise their value, a solution that enables authorised researchers from different institutions to obtain secure and reliable access is clearly needed. Here we present a framework that facilitates and streamlines data access for the scientific community through user-friendly and efficient web applications capable of rapidly handling large-scale datasets. We developed a web platform that eases controlled and structured access to data for both internal personnel and external collaborators. The platform enables seamless data retrieval from designated facility systems within specific experiments. It also supports the assignment of different user roles to ensure that the infrastructure's security is not compromised. Furthermore, we implemented a database-replication system that offers faster data access, ensures data recovery in the event of a geographical disaster, facilitates data transfer to external specialised processing frameworks or databases, and allows the creation of local subsets of the global database to accelerate query performance while addressing data governance issues. This framework has been successfully implemented in the IFMIF-DONES prototype, LIPAc (Linear IFMIF Prototype Accelerator). The development of this methodology paves the way for enhanced accessibility to data from nuclear fusion research infrastructures. We demonstrate that it serves as a foundation for scalable data management practices that can be adopted by future large-scale scientific facilities. Such practices would enable predictive maintenance and establish a clear abstraction layer between the systems that store the information and the state-of-the-art and upcoming processing tools. Ultimately, broadening access to these data for researchers could accelerate scientific discovery, foster institutional collaboration, support the adoption of AI techniques, and contribute to more transparent, interoperable, and collaborative research workflows.

#13362

SYCOMORE system code modernization and preliminary application to high field compact tokamaks

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

SYCOMORE is a modular integrated systems code for tokamak reactor design developed in IRFM/CEA in the recent past, the modules communicated to each other using the European Integrated Tokamak Modelling framework (EU-ITM) and were linked in the graphical interface software Kepler to create the workflows [1]. The ITM is migrated to IMAS [2] and SYCOMORE was not been usable in IRFM servers. To address this SYCOMORE has been revived and comprehensively modernized. This includes code migration, integration with IMAS 4.1 data standards and MUSCLE3 [3] multi-language coupling framework (C++/Fortran/Python) which allows the different actors in the code to communicate by sending IDSs to each other. The versions of Python and C++ have been updated in the new version. SYCOMORE is also coupled with a recent version of URANIE [4] platform for statistical analysis and optimization. In this contribution we present the details of the modernization and benchmarking of SYCOMORE.

We describe the evolution of the models in order to allow analyzing high field HTS-based tokamaks, e.g. the implementation of a bucking model. For this, we use the configuration generated by D0FUS systems code [5].

We then present the first application of SYCOMORE to a high field HTS-based power plant design, namely the Pfus = 2 GW plant studied in the frame of the SF-Plant project. We investigate a specific aspects of the design: the amount of impurities necessary to achieve plasma detachment. Finally, we also show the design optimization studies using SYCOMORE within the URANIE optimization framework.

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#13365

Realizing SPARC: Engineering Integration and the Operational Roadmap to Net Energy

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

Commonwealth Fusion Systems (CFS) and its partners are transitioning the SPARC tokamak (B_T = 12.2 T, R_0 = 1.85 m, a = 0.57 m, designed to achieve Q ~ 11) from a collection of high-performance designs into an integrated, operational fusion device.

SPARC is currently under construction and assembly in Devens, Massachusetts. This presentation provides a status update on the delivery and installation of major fusion core components, including high-temperature superconducting (HTS) toroidal field coils, poloidal field coils, copper magnets, vacuum vessel, in-vessel components and diagnostic systems.

As the project shifts from component delivery to large-scale assembly, the team has encountered and resolved the real-world engineering complexities inherent in the first-of-a-kind integration of a compact, high-field device. This presentation discusses the technical solutions developed to overcome some of these challenges, ensuring structural, electrical and thermal requirements are met across the entire integrated machine and plant architecture.

Beyond physical assembly, CFS is executing a strategic, phased commissioning roadmap to bridge the gap toward first plasma. This begins with a partially integrated “dry” commissioning phase, or dry dress rehearsal (DDR), on track to complete in 2026. Unlike purely virtual testing, the DDR involves the active utilization of the actual plant and control systems, augmented by Hardware-in-the-Loop (HITL) simulations, to orchestrate physical plant subsystems (including cryogenics, power supplies, and vacuum systems) in a non-nuclear environment. This strategy is designed to retire infrastructure risks early, ensuring that the critical path to $Q > 1$ is dictated by the performance of the tokamak itself rather than its supporting systems. These operations leverage the in-house Neutrino framework, transitioning it from a developmental environment into its role as the machine’s unified control and data architecture.

By iteratively building upon the operational tools and frameworks established during the dry commissioning, CFS is establishing a repeatable pathway for fully integrated commissioning. This rigorous approach ensures the machine is prepared for its initial operational campaign, focusing on the milestone achievement of First Plasma and the ultimate scientific demonstration of $Q > 1$ net energy. Progress on SPARC remains one of the primary vehicles for informing the design and accelerated deployment of ARC, the first fusion power plant, in parallel to other R&D efforts.

#13404

Review of corrosion behavior in fusion reactor

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

The corrosion behaviour in cooling loops of fusion reactor is an important aspect to be assessed for reactor design and operation. It can lead to concerns due to the generation of Activated Corrosion Products (ACP) radiation source term, as well as the deterioration of structural integrity of cooling components. This paper presents a review of the physical and chemical behaviour of corrosion in fusion reactor environment for various materials and coolants, and its application under ITER operating conditions. The key parameters to impact the corrosion behaviour of materials in nuclear fusion environment are: the response of new materials such as CuCrZr alloy, the reduced activation ferritic-martensitic steel under the combined effects of high energy neutron irradiation, the induced water radiolysis effect and water chemistry management, presence of intense magnetic fields and impact of working fluids temperature and velocity. Experimental tests were performed in 2025 to evaluate CuCrZr alloy water corrosion behaviour considering temperature and water chemistry conditions relevant for ITER. Manufacturing-induced material properties impact is also discussed. Design and operation solutions are proposed to limit the impact of corrosion in the cooling loops of a fusion reactor. The ITER knowledge acquisition program on ACP, which aims at taking benefit of early operations to support later nuclear phases, is also presented and discussed.

#13411

Design approach for STEP Fusion Powerplant outboard component integration

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

The integration of in-vessel systems in fusion power plants presents a complex, multi-disciplinary engineering challenge that is central for a reliable design. The STEP Fusion Powerplant introduces further complexity due to its compact major radius, high plasma elongation, and maintenance concepts that differ significantly from conventional tokamaks. These impose stringent geometric and engineering constraints on the arrangement, support, and serviceability of in-vessel systems.

In the STEP Fusion Powerplant outboard region, a diverse suite of components—including first walls, limiters, blanket modules, passive stabilisation coils (PSC), resistive wall mode (RWM) coils, runaway electron mitigation (REM) coils, diagnostic systems, and heating-and-current-drive (HCD) interfaces—must be accommodated within a highly restricted vacuum vessel. Interfaces between these components further complicate the design integration due to differential thermal expansion, complex load paths, electromagnetic loads, and remote maintainability.

This paper presents the major integration challenges encountered in STEP Fusion Powerplant and outlines the engineering strategies focusing on the spatial position and mechanical support. A major barrier to integrated design is the early-stage design of individual systems often developed in silos. This leads to spatial clashes and conflicting design drivers across systems.

Strategies to mitigate these challenges are presented through a structured, integration-led design. Key stages include functional allocation of ports for service lines and diagnostics, definition of HCD and PSC coil locations to establish spatial constraints. These decisions form the geometric foundation that drives subsequent limiter spatial design and blanket modularisation. Mechanical support concepts for PSCs and Blanket have been developed through an integration design process. Coolant manifold routing between the vacuum vessel and blanket modules has been developed within restricted space. A maintenance strategy has been developed to support component installation and removal. The work demonstrates how structured and cross-disciplinary design coordination is essential for realising an integrated in-vessel architecture in next-generation fusion power plants.

A Surrogate Boundary Condition for Acceleration of Fusion Neutronics

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

Important design considerations for fusion devices, such as shielding and activation, require Monte Carlo neutron transport simulations. Neutronics analysis of fusion devices is computationally challenging because of the complexity of proposed geometries. Iterating through the neutronics design of individual components is often impractical. Because transport inside a component depends on that outside it, expensive full-model simulations are often required even when the region of interest is local. Current approaches impose vacuum or reflective boundary conditions on the component. These techniques avoid modeling the surrounding geometry, reducing cost but omitting potentially important physics. We propose a surrogate, physics-informed boundary condition that captures the transport in the surrounding geometry while allowing the component to be designed in isolation. We implement the boundary condition as a neural network that predicts how particles leaving the component return after interacting with the surroundings. The network is trained on a single full-model simulation and, once trained, enables rapid iteration of a component design. We demonstrate the method on a test geometry and compare results to the full-model transport solution.

Thermal-Hydraulic Modeling and Preliminary Transient Performance Evaluation of the ITER IBED Cooling Loop

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

ITER Cooling Water System (CWS) is designed to reject into the environment all the heat generated within the ITER tokamak machine. Four main sub-systems compose the CWS: the Tokamak Cooling Water System (TCWS), the Component Cooling Water System (CCWS), the Chilled Water System (CHWS) and the Heat Rejection System (HRS). The ongoing research activity is focused on TWCS Primary Heat Transfer System (PHTS), specifically on the IBED PHTS (Integrated loop of the Blanket, Edge Localized Mode (ELM)/Vertical Stabilization (VS) coils, Divertor (DIV) PHTS) responsible for the cooling of First Wall Blanket (FW/BLK) modules, DIV cassettes, ELM/VS coils, upper and equatorial port plugs. The IBED PHTS is a pressurized, closed-loop cooling water system that supplies cooling water to five cooling loops and consists of eight parallel primary pump/heat exchangers trains, one pressurizer, supply and return headers, ring manifolds and connections to all the in-vessel components. The present research work aims to investigate the dynamic behavior of IBED PHTS with emphasis on thermal hydraulic and control point of view. The IBED PHTS system-level model is updated after an initial phase of detailed review of existing ITER documentation related to system's architecture and operational functions. A system code is used to further upgrade existing thermal-hydraulic models of the cooling loop and to conduct studies of interest. Transient analyses evaluate performances and provide sizing verification of critical components, such as the pressurizer (PRZ) and the pressure relief tank (PRT), including a comprehensive development and validation of pressure control strategies, along with representative scenarios covering pump trip events, loss-of-power accidents and valves closure. These simulations allow the assessment of characteristic response times of the system along with key thermal-hydraulic parameters with focus on pressure evolution, mass flow distribution, temperature transients. Attention is also devoted to analyzing different IBED configurations in staged approaches associated with the new ITER Baseline. Preliminary results obtained within this study will be further developed and published in subsequent work.

Topology-Aware Modular Surrogate Modeling for Separation Process Networks

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

This study explores a modular surrogate modeling framework for the Isotope Separation System (ISS) to reduce input dimensionality and enhance sample efficiency. Surrogate models were employed to interconnect unit-level models within the fusion fuel cycle, improving computational efficiency while effectively addressing process nonlinearity. The ISS, consisting of four interconnected columns, exhibits an inherently large design space, thereby requiring a substantial number of training samples. In practice, each numerical simulation requires approximately one minute per case, and nearly 20,000 samples are needed even when the input space is limited to key operating variables such as feed conditions and heat duty. To overcome this, we decomposed the ISS into column-level surrogate modules, reducing input dimensionality and consequently improving sample efficiency. Although the modular approach can reduce simulation time, error propagation inevitably occurs between interconnected modules. Because the ISS includes recycle loops, a graph neural network (GNN) architecture was adopted to mitigate error accumulation. In this study, a topology-aware modular strategy was developed to explicitly represent inter-unit connectivity within modular decomposition. Rather than relying on local simulation-based corrections, the framework integrates process topology into the modeling structure to control error propagation across units. This structural formulation ensures scalability as the number of interconnected units increases, providing a basis for extending the approach to process-level modeling and optimization.

High-temperature superconductors for fusion power plants: preliminary design and multi-code assessment of a 2 GW tokamak

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

The emergence of private fusion ventures has renewed interest in compact tokamak design strategies, with proposals for smaller and more economical power plants enabled by high-temperature superconductors (HTS). Understanding the conditions under which HTS technology delivers enhanced performance requires systematic assessment of both technological choices and plasma physics modeling assumptions.
In the framework of the French PEPR SupraFusion program, the system code D0FUS has been developed to explore these questions [1]. Parameter scans reveal that high magnetic fields enabled by HTS significantly expand the accessible design space from a plasma stability perspective [2]. However, optimal configurations depend critically on structural strategies: conventional wedged designs favor large aspect ratio machines, while compact geometries require alternative approaches (bucking or stronger steel). Additionally, D0FUS
has been used to evaluate plasma physics model uncertainties. Building on these insights, a preliminary Pfus = 2 GW fusion power plant design has been
studied. The optimized configuration, trade-off between plasma performance, mechanical constraints and physics margins, is achieved for ????0 ≈ 6???? and ????0 ≈ 10 ????, both evaluated on the magnetic axis. This design has been assessed using multiple codes and approaches: system-level optimization was performed with D0FUS following previous conclusions and benchmarked against SYCOMORE [3] PROCESS [4,5] and FUSE [6]. A higher fidelity assessment has then been performed employing METIS [7] for plasma simulations, NICE [8] for equilibrium calculations, and MADMACS [9] and CIRCE [10] for magnet analysis. Cross-comparisons between these codes provides confidence in key
predictions while highlighting uncertainties inherent to different modeling approaches.
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An Improved Integrated Physics and Technology Tool for Fusion Plant Design

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

To meet the need for tools that balance predictive fidelity with reduced runtime in system-code studies for fusion plant design, we developed an advanced computational framework that integrates plasma physics and tokamak technology. It couples the METIS physics engine -a fast plasma simulation code based on 0D scaling-law, normalised heat and particle transport with 1D current diffusion modelling [1]- with the state-of-the-art magnet sizing capabilities provided by the MADE (Magnet Design Explorer) algorithm [2].
The adoption of METIS allows for current profile prediction and a self-consistent power balance (heat exchange and radiation), representing a step forward in fidelity for the plasma description compared to many existing optimisation-based system codes. Additionally, the MADE module jointly optimises Toroidal Field coils and Central Solenoid (CS), aiming to maximise the magnetic flux available at the CS. It uses a micro-to-macro approach and, starting from the electrical, magnetic, thermal, and structural assessment of the conductors, produces consistent magnet winding packs without setting a priori limits on the development of their geometry, reducing the conservativeness compared to other system codes and providing designs that are less affected by sources of uncertainty.
The framework also embeds a simple optimisation logic that drives selected plasma and plant parameters toward user targets, and it currently includes magnet shielding and plant-cost modules. The code is written in MATLAB and supports parallel execution, allowing for wide parametric scans within practical computational times, even if runtimes exceed those of simpler codes.
This tool acts as a pre-screening step before higher-fidelity physics and engineering analyses and has already been employed to verify the designs of both VNS and DEMO-LAR through system-code scans. In this work, the current capabilities of the code are presented, together with a description of the optimisation algorithm employed, an application example on VNS, and the planned developments.
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Plasma system in fusion plant design: challenges, uncertainties and perspectives

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

System codes are used to integrate plasma physics with engineering constrains and design fusion power plants. The plasma parameters impact the fusion power, Pfus (core ion temperature and fuel density) and the pulse length, tpulse (electron temperature, impurity content). In system codes, such as PROCESS, up to 14 ad-hoc parameters and 4 scaling laws are needed to compute Pfus and tpulse. Moreover, scaling laws are used far from their training domain. For example, EU- DEMO energy is expected to be around 1000 MJ while the highest energy achieved in JET is less than 10 MJ, the H mode access threshold is expected 100 MW while in the used scaling law is at most tested around 10 MW, etc. To address fusion power plant design robustness, it is mandatory to assess the impact of epistemic plasma physics uncertainties. The EU-DEMO new baseline is used for illustration [Coleman NF 2025] in which a design margin has been taken on q95, hence plasma current.

Increasing the fidelity level of the plasma model is an effective way of reducing the number of ad-hoc parameters. In this improved framework (METIS [Artaud NF 2018]), the impact of uncertain energy content extrapolation is explored while keeping fixed Pfus or tpulse. The later can vary from 2 h down to 1h.

We have further reduced ad-hoc choices by self-consistently computing the plasma critical density and temperature gradients [Bourdelle PPCF2025]. In absence of experimental burning plasma data for validation, we have verified turbulence models against frontier GPU accelerated simulations by GENE-Tango [Di Siena sub. NF 2026]. The burning plasma modelled is the inner ITER 15 MA plasma (

#13597

Uncertainty Quantification of Failure Severity for DEMO Using Interval Analysis

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

Fusion power plant systems codes such as PROCESS can be used to explore the design parameter space to find feasible and optimal power plant designs. However, under epistemic uncertainty those designs may become infeasible by failing to meet design requirements (e.g. ability to access H-mode, minimum burn time), an effect which can be quantified by the severity of failure. Therefore when assessing the suitability of a given design, it is important to consider the severity of possible failures under uncertainty. In this work, a technique is described to determine the failure severity response interval for a design under uncertainty. Firstly, PROCESS was used to optimise a generic DEMO (similar to EU-DEMO) fusion power plant design problem, for maximum net electric power and minimum major radius to create two design points for comparison. The epistemic uncertainties (e.g. tungsten impurity concentration) were treated as intervals and an interval-based sensitivity analysis method was used to analyse the two designs under uncertainty in order to quantify the relative influence of the uncertain parameters on failure severity. Due to the need to reduce the high dimensionality of the problem, the least-influential parameters were "screened out" at this stage. A Gaussian Process model of the failure severity for each design was then trained with the most influential uncertainties as input parameters. Subsequently Bayesian Optimisation was used to find the bounds of the failure severity response interval for each design. For the first time, this technique quantifies the failure severity uncertainty for existing designs and allows for design selection based on minimisation of the upper bound of failure severity.

Assessment of the design space for a high field negative triangularity burning plasma tokamak

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

In the frame of the SupraFusion research program funded by the French National Research Agency, which aims at developing High Temperature Superconductor (HTS) technology and evaluating the impact of HTS magnets on fusion devices, we assess here the design space for a high field negative triangularity burning plasma tokamak.
Negative triangularity offers the perspective of good plasma confinement with a much facilitated heat exhaust thanks to the absence of H-mode and edge localized modes, the placement of the divertor on the outboard side and the absence of an upper limit for outer core radiation (since there is no need to maintain an H-mode).
Several groups have proposed preliminary designs for such machines, in particular CENTAUR (Columbia University) and LUCIOLE (Firefly Fusion). The goals and main parameters of both machines are similar: a net fusion gain (Q>1) for CENTAUR or a burning plasma (Q>5) for LUCIOLE at moderate cost, relying on a magnetic field on axis of about 10 T to reach a relatively small major radius of about 2 m for CENTAUR and 2.5 m for LUCIOLE (both designs having an aspect ratio close to 3).
We use the system code D0FUS, plasma scenario modelling codes METIS and HFPS (the latter embedding the quasi-linear transport model TGLF), edge plasma and divertor modelling code Soledge3X, free-boundary equilibrium code NICE, and magnet modelling codes CIRCE and MADMACS, to analyse:
1) The impact of assumptions and uncertainties on the global design and machine cost;
2) The design of the central solenoid, toroidal and poloidal field coils (paying special attention to mechanical stress in the magnets as well as plasma vertical controllability);
3) The design of the divertor;
4) The possibility to reach extended objectives, such as a burning plasma with Ohmic power as the only auxiliary heat source, or a

sufficient neutron fluence to test breeding blanket modules, at the cost of a moderate increase in machine size.

#13675

Inboard Shield Blanket Options and Neutronic Analysis for Compact Fusion Pilot Plants

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

Compact fusion pilot plants are envisaged to demonstrate tritium breeding and electrical power generation and provide a solid technical basis for extrapolation to net electricity generation in DEMO/power plants. Due to the tight inboard space, only shield blanket is considered on the inboard side which pose a unique optimization problem for the inboard shielding. It has to have optimum thickness for the radiation protection of the magnets as
well as have simultaneous heat extraction at a low temperature. Therefore, the thickness, choices of materials and coolant, flow rate, volume etc. all become interlinked.
In this work we discuss the parametric analysis of the overall performance of the shield blanket using a combined thermal and nuclear analysis. This has been carried out using a fast-converging 1D-effective(heat) loss channel model (ELCM) which is benchmarked with ANSYS simulations implemented in SARAS systems code. We take a potential configuration of 3.6 m major radius with a fusion power of 300 MW and an average neutron wall load of 0.75 MW/m2 for detailing the optimization challenges in the shield blanket. As a
starting point 1D neutronic analysis of the entire plant is carried out using MCNP6.2. The shield is modelled as a combination of plates consisting of WC and B4C materials of 10 mm thickness immersed in water, enclosed in SS-316 LN (IG). Using this as the base model, an extensive multi-parameter-space scan of the shield blanket performance is conducted as a function of the composition of the shielding blanket (WC and B4C), water content, flow rate,
neutron load, fast neutron flux consistent with the thermal performance. We will also report on the comparison with TRIPOLI code using FENDL-3.1d and FENDL-3.2b libraries that will be cross-compared for neutron transport, and EPDL-97 library for photon transport.
The parametric analysis will contribute to train a metamodel based on linear regression models or artificial neural network models capable of instantly reproducing heating in TFC or Inboard Shield. This will be applied to larger power plant designs in order to check its consistency for other pilot plant configurations as well as compact HTS power plants.

PROGRESS ON EUROFUSION R&D PROGRAM ON NEUTRONICS SUPPORTING ITER NUCLEAR OPERATIONS

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

In ITER and future fusion devices, neutronics plays a central role in the characterization of the radiation sources, activation inventories, shielding performance, tritium breeding capability, as well as in plasma diagnostics and nuclear safety assessments. Until 2025, within the EUROfusion Work Package (WP) Preparation of ITER Operations (PrIO), the sub-project "Neutronics, Nuclear Waste and Safety" achieved major progress in neutronics modelling, experimental validation, and multi-physics integration in support of ITER nuclear operations. These advances were built on the experience gained during JET DT campaigns, complemented by parallel experiments at European facilities and dedicated methodological studies. The activities are continuing under the WP Design and Operation (DES).

The 2026–2027 programme aims to consolidate and expand EUROfusion activities in neutronics in support of ITER nuclear operations and the Knowledge-Acquisition Program for licensing. The objectives are to perform advanced neutronic studies and code developments aligned with ITER priorities; to validate nuclear codes and methodologies, covering neutronics, activation, shutdown dose rate and waste characterisation, during the JET decommissioning phase; to integrate plasma physics and neutronics simulations in support of the updated ITER baseline and neutron synthetic diagnostics; and to carry out dedicated experiments on fluid activation, Activated Corrosion Products (ACP), and Single Event Effects (SEE) on electronics at major European facilities, including the Frascati Neutron Generator (FNG), the JSI TRIGA research reactor (KATANA facility), and the WEST tokamak.

The activities include the preparation of neutronics experiments to support ITER DT2 licensing, based on JET DT experience. Integrated plasma physics–neutronics modelling will enable improved radiation source definition across ITER operational phases and enhance predictive capabilities for neutron diagnostics. Validation campaigns during JET decommissioning will reduce uncertainties in shutdown dose rate and activation assessments and refine techniques for irradiated material analysis and waste characterisation.

Dedicated experimental programmes benchmark multi-physics tools for water activation at the KATANA water loop. ACP experiments on copper under ITER-relevant conditions will support the validation and further development of predictive tools.

In parallel, open-source radiation transport codes, CAD-to-Monte-Carlo geometry conversion tools, and coupled multi-physics methodologies on shutdown dose rate, water activation and ACP are further developed. Single Event Effect (SEE) modelling and irradiation tests using the GENeuSIS (General Experimental Neutron System Irradiation Station) at FNG will assess electronics reliability under ITER-relevant neutron energy spectra.

Overall, the efforts strengthen the experimental basis and predictive capability of fusion neutronics, reducing uncertainties and supporting safe ITER nuclear operation and future devices design.

#13863

Use of Divertor and other Heat Sources for Preheating and Reheating in A Rankine Cycle System

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

Unlike a conventional or a fission power plant, a fusion power device (such as a stellarator or a tokamak) features multiple heat sources that can feed into a power conversion system (such as a Rankine cycle) to produce steam. These heat sources include the First Wall (FW), Breeder Blanket (BB), Vacuum Vessel (VV), High Temperature Shields (HTS), and divertors. Of the total fusion power produced by the plasma, most of the heat is deposited in the BB and extracted through its cooling system. However, a significant amount of heat is also deposited in the other systems. Understanding the impact of all these heat sources on the power conversion loop is crucial for maximizing the plant's efficiency. This study evaluates this impact by performing a parametric analysis in which the amount of heat deposited in and extracted from each system is varied, and by analyzing the effect on the plant's net efficiency. The current work also explores eliminating the intermediate loop used in other designs. The BB was used as the primary heat source, with the other heat sources (FW, divertors, and VV) serving as reheating and preheating stages in a Rankine cycle. The study estimates the potential benefits of removing the intermediate loop and transferring heat directly into the power conversion loop.

Lower Port Neutronics Analyses for EU-DEMO and Volumetric Neutron Source – common challenges and shielding options

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

Development activities performed at EUROfusion for a demonstration fusion power plant DEMO and a Volumetric Neutron Source (VNS) tokamak confirm the significant impact of neutronics simulation results on design integration issues considering safety and licensing aspects.

One common area of these machines where system integration is particularly challenging are the lower ports as sufficient shielding of critical components competes with the open space necessary for operation (vacuum pumping), machine assembly and in-vessel component replacements or maintenance by remote handling systems. Although the sizes of DEMO and VNS differ significantly, with major radii of approximately 9 m and 2.5 m, respectively, similar approaches for shielding solutions can be applied. Some potential options require design changes to components in the lower port region, such as the divertor, blanket structures, and lower port walls, where the inclusion of shield components and mitigation of direct streaming paths are beneficial. Other potential options include local shielding, such as shielding sleeves around divertor cooling pipes designed to locally reduce helium production at the location where pipes will be cut and rewelded. At the conceptual design phase, it is important to understand their feasibility, effectiveness and, whenever possible, identify several alternative shielding solutions.

In this paper, we present design options for shielding in the area of the lower port of DEMO and VNS based on insights from neutronics analyses.

A novel Multi-Timescale strategy for Fusion Systems Codes and its impact to parametric analyses of Fusion Power Plants

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

Fusion Systems Codes (SCs) are fundamental tools that allow for parameter space exploration and evaluation of technology integration through parametric analyses that may identify relevant systemic dependencies in Fusion Power Plants (FPPs). However, current state-of-the-art SCs model each plant system operating within its own inherent timescale, and neglect the dynamic interdependence between them at a power plant level. However, current state-of-the-art SCs model each plant system operating within its own inherent timescale, and neglect the dynamic interdependence between them at a power plant level. A multi-year project is underway to illustrate this gap with novel parametric dependencies, e.g. between SOL plasma filaments parameters and the net power production of the plant, or between the First Wall design temperature and aspects of tritium self-sufficiency.

For that, a novel strategy to produce Multi-Timescale (MT) SCs was tested with a EU-DEMO model built in MIRA, a multi-fidelity SC developed at the Karlsruhe Institute of Technology (KIT). This strategy identified three timescales in which most FPP systems are categorized: (a) the Operation of a FPP (a collection of pulses), (b) a single reactor Pulse and (c) the characteristic time in which Plasma dynamics evolve.

New models have been developed to depict essential phenomena of systems in each timescale. For (a), a multi-species Fuel Cycle module computes fuel accumulation rates using a residence-times (τ) model, with τ-parameters derived from the characterization of selected technologies. For (b), a Power Cycle module computes the net power production by the Balance-of-Plant systems using thermodynamic models. For (c), SOL plasma dynamics are computed with scaling laws and an optimized surrogate of the code TOKES, to estimate heat and particle distributions along the reactor chamber wall. Consistent coupling is ensured by using results from (c) to compute outgassing fluxes with the double-diffusion code TESSIM-X. This impacts the dwell-time, and thus the dynamics of both (a) and (b), which leads to stronger dependencies than those obtained from non-MT codes.

Preparation of the tokamak assembly in the DTT facility

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Poster

Abstract

The Divertor Tokamak Test facility (DTT) [1] is a superconducting tokamak currently under construction at ENEA C.R. Frascati. The mission of DTT is to provide an integrated environment where to test power exhaust solutions in support and in alternative to ITER baseline. At this aim, the facility has been designed to be flexible in terms of plasma scenarios, heating system and divertor system. The construction of the critical components has started and the site is in preparation for starting the assembly activities.
This paper focuses on the preparation activities for the assembly of the components and auxiliaries located within the perimeter of the Torus Hall building. Indeed, to validate and set-up the assembly procedures a wide use of mock-ups is foreseen. In this phase, a building will be used to house the mock-ups to be manufactured to validate the special processes and to train the workers in order to ensure that the final assembly activities are carried out in compliance with the quality and time requirements imposed by the project. In this regard, the use of models of various components, such as the vacuum vessel, cryostat, first wall and divertor, made during the manufacturing of the related series components, will have to be integrated to produce a nearly complete full-scale model of the machine in which the welding procedures, non-destructive testing and metrology will have to be tested and qualified.

[1] Francesco Romanelli et al 2024 Nucl. Fusion 64 112015

Keywords: DTT, tokamak, assembly

ITER technical Baseline 2024: full implementation under configuration control

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Topics

A. Integrated Plant Systems and Pathways Toward Fusion Power Plants

Communication type

Oral

Abstract

The ITER technical Baseline 2024 has been comprehensively implemented in the project engineering files through rigorous configuration control. All aspects of the machine and plant setup have been optimally configured to meet the objectives of the new ITER Research Plan with its new corresponding operation stages (SRO, DT-1 and DT2), together with the incorporation of a number of concurrent notable changes like the adoption of the tungsten first wall, the use of boronization, the increased EC H&CD power or the new tritium processing strategy. Strict management and engineering integration of the new technical and programmatic proposals have been essential, especially considering the advanced design and construction stages of many of the involved systems.

To facilitate effective change management, multi-system design integration reviews were conducted. These reviews provided a structured strategy for implementing changes by grouping them according to geographical areas. This organization helped managing interfacing requirements between systems in an integrated manner, minimizing risks of inconsistencies or gaps. For a period of one year and half, the Configuration Management team in ITER Organization coordinated the synchronous update of more than 1000 engineering files (including requirement documents, interface data, CAD models and diagrams) that were considered essential to keep the configuration under control. The Configuration Management team did also preserve the consistency of the overall project architecture in relation to the fulfillment of the project objectives: from updated requirements to design solutions, from load specifications to process diagrams.

This paper describes both the key elements of the new technical baseline and the process to update the project configuration with no disruption to the ongoing construction activities.