Docking and Charging Tower Concepts for Cooperative Trolley Fleets in Nuclear Fusion Facilities

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

During the lifetime of a nuclear fusion facility, ex-vessel transportation systems must ensure the safe, reliable, and continuous handling of heavy and potentially activated components. As human intervention is not possible in such environments, autonomous and remotely operated systems are required. Recently, a new transportation solution for heavy and activated loads is being explored as a possible alternative to the single pallet transporters. This new transportation solution uses a fleet of mobile cooperative trolleys. Although this transportation solution has relevant logistics potential, it also presents new challenges, such as the charging and parking of trolleys.

This work presents the development of a docking and charging tower concept for cooperative fleets of autonomous trolleys intended for operation in nuclear fusion facilities such as DEMO. The proposed towers serve as multi-functional interface systems, enabling parking, charging, alignment, and fleet availability management. The charging towers make use of an elevation platform so that trolleys can be stored vertically in individual drawers. The elevation platform can access any required drawer of the tower to store or retrieve trolleys. Each drawer is equipped with a charging and data transmission interface, as well as guidance system that enables the correct and safe interface between the tower and the trolleys without any damage. The docking tower is a modular system that allows scalable deployment by depending on the required number of trolleys and the hight limitations of the environment. Based on those two factors, the charging tower can be easily adapted to the overall requirements, both in number of towers and number of trolleys each tower can handle. A modular configuration is proposed and compared, addressing key design challenges such as guidance and alignment under limited sensing conditions, mechanical tolerance management, redundancy, and recovery from failed docking attempts. Design trade-offs are discussed in terms of footprint, operational complexity, reliability, and maintainability.

Engineering simulations, CAD-based modelling, and sequence-of-operations analyses are used to validate the concepts and identify potential risks and mitigation strategies. The results demonstrate that dedicated docking and charging towers are a critical enabler for long-term deployment of cooperative trolley fleets, contributing significantly to system availability, operational robustness, and plant maintainability. Although developed in a nuclear fusion context, the proposed concepts are also applicable to other highly constrained industrial environments requiring autonomous fleet management.

VIPER: A testbed for Fusion devices component validation under ALARA requirement

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Oral

Abstract

The increasing complexity of fusion devices and the constraints imposed by nuclear environments require advanced tools to support component integration, maintenance, and remote handling operations. The Virtual Integration Platform for Engineering and Remote Handling (VIPER) has been developed as a comprehensive testbed combining digital continuity, immersive simulation, nuclear constraints, and robotics within a unified framework. VIPER supports fusion projects by enabling early validation of design choices, operational feasibility, and human system interactions in realistic virtual environments.

The platform provides a wide range of simulation tools adapted to different fusion use cases. VIPER enables full-scale, one-to-one immersive virtual design reviews, allowing stakeholders to assess component integration, accessibility, and spatial constraints at early design stages. For installation and maintenance activities, the platform supports desktop-based assembly sequences and Augmented Reality applications, currently applied to the installation of new components on the WEST tokamak, such as the Travelling Wave Array (TWA) antenna.

A core capability of VIPER lies in the simulation of operator tasks, integrating feasibility, accessibility, and ergonomics analyses while accounting for human factors early in the design process. This approach allows the identification of critical issues related to posture, fatigue, and task duration before physical implementation. In parallel, VIPER investigates remote handling solutions, including telemanipulation studies with dual Wälischmiller manipulators and simulated teleoperation using force-feedback Haption devices coupled with virtual or industrial UR10 robots. The platform also explores autonomous robotic systems equipped with vision-based perception, aiming toward self-learning robotic solutions for nuclear environments.

A key demonstration of VIPER's integrative capabilities is the "SmartCrane" demonstrator, which combines several advanced technologies into a single experimental setup. It integrates a physical mock-up of a compact lifting device with force control and automatic load weighing, designed by CEA LIST to enable effortless manipulation by operators wearing ventilated suits during maintenance operations in confined ITER areas. This system is coupled with the T-VISION ventilated suit, developed in collaboration with MATISEC and adapted for immersive simulations. The demonstrator relies on a virtual environment incorporating a digital twin of the SmartCrane and an ergonomic human mannequin based on the XDE physics engine developed by CEA LIST. Compatibility with XR tools including virtual reality headsets, motion tracking systems and real-time rendering engines ensures flexibility and scalability. Ultimately, VIPER aims to integrate operator dose assessment to support compliance with the ALARA principle, reinforcing its role as a testbed for safe and efficient fusion device operations.

Manufacturing of High-performance Fasteners for ITER Vacuum Vessel Gravity Support

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

The ITER Vacuum Vessel Gravity Support (VVGS) is a critical component that must withstand substantial loads, including dead weight, thermal expansion, and electromagnetic forces. To ensure structural integrity, a pretension of approximately 1.6 MN is required, necessitating the use of large fasteners such as M72 fasteners. At this scale, conventional torque-based tightening is impractical, as the required torque increases cubically with the bolt diameter, making assembly extremely difficult. To address this limitation, mechanical tensioning nuts could be more suitable. However, these are proprietary products protected by intellectual property rights and can only be manufactured by a limited number of specialized companies. The ITER project must strictly adhere to the French nuclear regulations, RCC-MR, which require comprehensive quality control and documentation throughout the material fabrication process. Nevertheless, these manufacturers have limited experiences with these stringent nuclear-grade and ITER-specific requirements. Consequently, their direct production without specialized oversight may introduce risks such as non-conformity. This paper presents the strategies adopted to overcome these challenges, including collaboration between the prime contractor, material suppliers, and tensioning nut manufacturers. It further details how to meet the nuclear and the ITER-specific technical requirement.

A Data-Driven PHM Framework for Remote Handling Systems Using Acoustic Sensing on a Motor Test Rig

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

Remote handling systems are widely used in environments where direct human intervention is difficult or restricted, such as hazardous or highly constrained operational settings. Failures in such systems can lead to significant operational delays and costly recovery procedures. Therefore, robust condition monitoring and early fault detection capabilities are essential to ensure reliable operation, minimize unplanned downtime, and improve overall system availability and operational safety. This paper presents a data-driven prognostics and health management (PHM) framework for remote handling systems, developed and validated using a dedicated motor test rig. The test rig is designed to emulate key actuation characteristics representative of transport and maintenance subsystems commonly used in remote handling operations, providing a controlled yet realistic experimental environment for systematic data acquisition and repeatable testing. This approach enables the development and evaluation of monitoring strategies prior to their deployment in more complex robotic systems. The proposed framework employs acoustic sensing as a primary monitoring modality to capture system responses from monitored components on the motor test rig under both nominal and degraded operating conditions. Machine learning models are trained using the acquired acoustic measurements to identify anomalous behaviors associated with overload and degradation mechanisms in robotic actuation systems. By adopting a data-driven approach, the framework reduces reliance on manually engineered diagnostic features and aims to maintain robustness under varying load profiles and non-stationary operating conditions. Experimental results demonstrate that the proposed framework can reliably distinguish nominal operating states from fault conditions across multiple operating scenarios. The framework also exhibits low-latency inference performance, indicating its suitability for near real-time condition monitoring. By combining acoustic sensing with a motor test rig–based development strategy, this work provides a practical pathway toward improving the reliability and operational safety of remote handling systems used in complex robotic maintenance environments.

Inboard Shielding Module Concept Design for the STEP Fusion Powerplant

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

The STEP Fusion powerplant is the UK's prototype fusion powerplant that will demonstrate net energy from fusion. It will be a spherical tokamak aimed at demonstrating the key characteristics relevant to commercial power plants. The spherical shape demands a relatively small inboard build radius in which to fit the magnets of the central column and the shielding module for protecting those magnets from the plasma.

Lifetime requirements for the magnets are defined in line with the STEP Fusion powerplant's objective to demonstrate a route to commercial levels of plant availability. The neutron shielding performance requirements for the Inboard shielding module are directly derived from these magnet lifetime requirements.

The Inboard Shielding Module concept design is based on radial layers of different subsystems, complementarily configured to best achieve the required shielding performance within the tight spatial constraints.

However, 'sufficient magnet lifetimes' is not the only contributor to demonstrating a route to commercial levels of plant availability – the lifetime, reliability, maintainability and refurbishment of the Inboard Shielding Module itself are also very important. Moreover, the STEP Fusion powerplant has other measures of demonstrating the key characteristics relevant to commercial power plants, including net power output and tritium fuel self-sufficiency – these all contribute to achieving the desired level of steady-state operations for commercial relevance.

From each of these aspects, further requirements are derived for the Inboard Shielding Module itself, which presents notable challenges around efficient and effective coolant loops, magnet shielding from a cryoplant power perspective, the integration of pellet lines as part of the fuel recirculation system, lifetimes & reliability of key components, and maintenance & refurbishment of the module for continued operation.

Here, in answer to these challenging requirements, we will present the Inboard Shielding Module Concept Design for the STEP Fusion powerplant, the reasoning for key design features and decisions, the supporting analyses and assessments, the ongoing challenges, and the future development activities for the Inboard Shielding Module.

Electromagnetic analyses of DTT stabilizing plates during disruptions

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

The main target of DTT tokamak, currently under construction in the site of Frascati is, according to the European Fusion Roadmap, to investigate possible solutions for the heat-exhaust problem (Mission 2) such as alternative magnetic configurations or liquid metal divertors. The DTT device is in fact able to develop standard Single Null (SN) configuration and advanced plasma configurations including Negative Triangularity (NT), X-Divertor (XD), Double Null (DN) and long leg equilibria. Since DTT is capable to produce a large variety of plasma configurations, two passive stabilizing plates connected in anti-series have been designed in order to improve passive vertical stability performances. The anti-series connection has been chosen to have a zero net toroidal current avoiding problems during plasma breakdown. Furthermore, since stabilizing plates are located in proximity to the First Wall (FW), an evaluation of induced currents and forces during disruptive events is mandatory. A disruption is a sudden, uncontrolled loss of plasma confinement, terminating the fusion reaction in few millisecond and leading to rapid thermal energy loss and current decay, often generating runaway electrons and severe thermal and mechanical loads. This work aims to study the effect of disruptions on DTT stabilizing plates under the hypothesis of different plate configurations. The model adopted for disruption simulation is a multi-filament model which foresees the distribution of plasma current among three filaments during vertical displacement event (VDE), thermal quench (TQ) and current quench (CQ) by balancing the 0-th and 1-st orders moments. Furthermore, additional studies have been carried out to evaluate the impact of the skin effect and on passive vertical stability parameters such as plasma growth rate. The outputs of the proposed analyses are presented in terms of: induced currents and voltages (the computation of voltages is important in order to evaluate possible arcing phenomena due to high voltage) and total force on passive stabilizing plates during disruptions.

#12808

Remote Maintenance Constrained Multi-Objective Optimization of Parameterized Tokamak Blanket Geometry

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

Tokamak blanket modules provide tritium breeding and neutron shielding and must be periodically replaced for longterm safety. Because replacement occurs through a limited upperport opening, remote maintenance is the only viable method, and the tight space strongly couples blanket geometry, mass distribution, and extraction path. The blanket design must therefore balance neutronic and structural performance with extractability.

This study applies a parameterized modeling approach to inboard and outboard blanket modules, enabling systematic adjustment of geometry and mass distribution while keeping internal structures unchanged. By shifting the center of mass relative to the lifting point, tilting torque during extraction can be reduced. Neutronic performance (TBR and energy deposition) is evaluated using OpenMC, and structural stresses and deformations are assessed with ANSYS. Because geometric changes affect attitude and clearance, extraction paths are reevaluated using kinematic and collisionbased feasibility analysis.

To reduce computational cost, surrogate models for TBR, torque, and deformation are built from parameterized samples, and a randomsearch multiobjective optimization algorithm identifies designs that maximize TBR, minimize torque and deformation, and satisfy pathfeasibility constraints.

Optimized design, analysis and control of heavy-duty manipulator fusion nuclear application

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

Remote maintenance inside the vacuum vessel of future fusion reactors such as DEMO requires robotic systems capable of handling very high payloads within severe geometric constraints. This work presents the ongoing development of a Multi-Purpose Deployer (MPD) intended for the removal of multi-tonne in-vessel components as limiters and perform maintenance task through restricted equatorial ports. The MPD is designed to adopt a reduced-degree-of-freedom manipulator configuration to maintain structural stiffness while ensuring sufficient operational reach. The design process follows concept design of MPD, workspace mapping inside the DEMO vessel to verify reachability of MPD for designated application and structural assessment using finite element analysis to evaluate strength and deformation under high payload loads. The study aims to deliver a robust deployer concept capable of safely performing heavy maintenance tasks inside the reactor vessel , The kinematic, dynamic and deformation estimation for the manipulator has been carried out. finally, the optimized manipulator is presented in the paper. Future work will focus on detailed design refinement and the development of a control system to enable precise, stable, and safe manipulation of heavy components in the DEMO environment.

4D Viewer: An Interactive and Immersive Application for the Visualization of Technical Sequences in Fusion Engineering

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Oral

Abstract

Fusion engineering projects involve highly complex systems and critical technical procedures, including assembly, disassembly, and decommissioning operations. The spatio-temporal understanding of these procedures represents a major challenge for engineering design, scenario validation, and operational analysis. This paper presents the VIPER 4D Viewer, a real-time application developed with Unity within the VIPER platform (Virtual Integrated Platform for Engineering and Remote Handling) at the IRFM (Institute for Magnetic Fusion Research), designed to support interactive 4D visualization for fusion-related engineering activities. In this context, 4D refers to the integration of the temporal dimension in a 3D environment visualization, enabling the controlled playback and exploration of animated sequences over time.

4D Viewer provides tools aimed at improving the understanding and exploration of time-dependent technical procedures. A temporal navigation system allows users to browse animated sequences step by step, both forward and backward. A configurable multi-view system based on a split-screen interface enables simultaneous observation from multiple controllable viewpoints. Available viewpoints include a first-person view, an immersive virtual reality (VR) view, third-person views dedicated to monitoring a VR user or controlling an avatar in desktop mode, and a top-down map-like view. A runtime inspector complements these features by allowing users to explore the model hierarchy and access the data associated with individual components during execution.

Beyond visualization functionalities, this work addresses the workflow required to integrate sequences and engineering data into the 4D Viewer. This workflow encompasses sequence definition, 3D model conversion, and integration into the application. Partial automation has been implemented by leveraging data from commonly used engineering tools: Model-Based Systems Engineering (MBSE) approaches with Papyrus are used to define procedural sequences, while CAD geometry and metadata are extracted from Catia to support model preparation and semantic enrichment.

The approach has been applied in three different project contexts. For WEST, the 4D Viewer has been used to study assembly sequences, including the divertor assembly and the installation of the ICRH antenna. For ITER, it has supported dismantling studies conducted for AIF (ITER France Agency). For DEMO, it has been employed to explore the sequence related to the removal of the Lower Port Divertor. These applications illustrate how 4D visualization can support the exploration of complex technical procedures in fusion engineering and highlight the potential of interactive and immersive technologies for advanced engineering workflows.

Quantification of Variable Contributions using Dimensional Analysis and Pi Theorems for Early Conceptual Design of Complex Systems

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

Generalised Dimensional Analysis and Pi-Theorems offer a rapid qualitative physics-based reasoning framework in the early-stage risk analysis in the conceptual design of complex systems, such as remote handling equipment in an activated environment, such as Fusion Power Plants, governed by physical effect equations describing the interrelationships between input and output variables. However, existing qualitative reasoning approaches remain limited, since directional changes in output variables (e.g., risk levels) are primarily inferred from the derivatives of shared variables within dimensionless Pi groups, leaving the magnitude contribution of individual input variables insufficiently answered. This paper introduces a method to address these limitations through the use of derivative exponents, enabling a more explicit representation of each variable's change magnitude. Two mathematical design case studies: the climbing monorail deployment system on In-BioShield's Vacuum Vessel surface and the Pipe In-bore Inspection robot are presented to demonstrate the use of the method. The quantified contributions of input variables are compared against conventional sensitivity analysis techniques. Preliminary results demonstrate its feasibility in capturing change magnitude, supporting its potential use as an early-stage abstraction and risk analysis tool in the design of complex systems.

#13003

Design and prototyping of Cable Management System for the In-Vessel Viewing System scanning probe

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

The In-Vessel Viewing System (IVVS) is a diagnostic tool developed for the ITER project to inspect plasma-facing components inside the tokamak. A scanning probe is permanently stored in a Vacuum Vessel port extension and deployed over 8 meters to reach the first wall, where it gains direct line-of-sight access to plasma-facing surfaces. Deployment is ensured by the Deployment System (DS).

Following an initial prototyping phase of several DS subsystems in 2022, Bertin Technologies initiated the prototyping of the Cable Management System (CMS). The CMS supplies power and transmits signals during probe deployment. It operates through two pulleys that move synchronously with the probe to provide the necessary cable slack, while the probe itself pulls the cable.

CMS prototyping included the development of a customized 15-meter cable integrating 54 AWG20 power lines, 120 AWG26 signal lines, and up to 70 optical fibers within the core. The cable design had to comply with ITER environmental requirements, using compatible materials such as aluminized tape, stainless steel braid, and qualified Poliax® wires. Custom connectors were developed at both ends, incorporating ITER-compatible sub-D connectors for electrical interfaces while allowing uninterrupted optical fiber routing to avoid additional optical losses.

In 2025, the CMS mechanical deployment structure was prototyped and tested. The system features two mobile pulleys traveling 2 meters toward a fixed pulley at 2.5 mm/s, driven by a ball screw mechanism. A major challenge was operating the mechanism without wet lubrication. Initial tests with aluminum bronze bushings proved unsuitable due to mechanical complexity. Parallel tribological testing of coatings (MoS_2, DLC, TiN, WS_2, graphite) showed that none could withstand the required contact pressures.

As a result, all rotating components were fitted with hybrid ball bearings combining silicon nitride balls and steel races. The ball screw incorporated graphite-coated balls from the outset. Tests conducted in 2025 using an improved dry-lubrication strategy successfully demonstrated reliable cable pay-in and pay-out. The lessons learned during prototyping were integrated into the final CMS design.

Model-Based Design and Simulation of an Optimized Manipulator Arm for ICH Antenna RH in DTT

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

The Divertor Tokamak Test (DTT) facility was designed to investigate the heat exhaust problem under reactor-relevant high-power conditions. To achieve high plasma power, multiple heating and current drive systems were foreseen. Among these, the Ion Cyclotron Heating (ICH) antenna was intended to deliver electromagnetic waves into the plasma within the 60–90 MHz frequency range. To couple the maximum required power of 1.5 MW per antenna, it was designed wider in the toroidal direction than the equatorial port, making Remote Handling (RH) mandatory. The ICH antenna was therefore decomposed into two subassemblies: (i) the antenna box, consisting of the plasma-facing components that had to be remotely handled due to their toroidal extension; and (ii) the antenna plug-in, comprising the active subsystems that did not require remote handling.

This work presented the Model-Based Systems Engineering (MBSE) framework adopted to design the RH System for the DTT ICH Antenna, in which the frontal-grip end-effector and the dedicated manipulator were developed according to the V-model using a Requirements, Functional, Logical, Physical (RFLP) approach.

System-level requirements were elicited from stakeholders, particularly the ICH Antenna system and the Hyper-redundant Manipulator (HYRMAN), the largest DTT RH robot operating at equatorial level and composed of a planar arm and a dexterous arm. These requirements were refined and allocated to functions to define the functional architecture. Subsequently, functions were allocated to components in the logical architecture using MAGIC Systems of Systems Architect. Finally, each identified component was modeled in CATIA V6 to define the physical domain of the robot.

Finite element simulations of the robotic system, performed under representative operational loads, provided quantitative estimates of displacement and von Mises stress, supporting verification of structural integrity and interface robustness within the V-model framework. In parallel, kinematic simulations verified the RH procedure and the corresponding requirements.

The combined MBSE, structural, and kinematic results closed the first verification loop of the V-model, consolidated the minimum kinematic chain, interface design, and geometric constraints, and provided a traceable foundation for the next design iteration.

Reality Gap mitigation in Divertor Remote Handling

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

Remote maintenance of the ITER Divertor requires high accuracy from large, hydraulically actuated systems. Load induced deflections, modelling and cumulative sensing–actuation–control errors create a persistent reality gap between planned and executed motions, posing risks to cassette assembly and replacement.

Within the updated ITER project baseline, the assembly of Divertor Cassettes will be performed with a simplified version of Divertor Remote Handling System named Divertor Assembly Transporter (DAT). VTT Divertor Test Platform 2 (DTP2) will be utilized to perform DAT functions such as transportation and positioning of the Divertor Cassettes. Ongoing development activities focus on upgrading the control and hydraulic systems of the DTP2 Remote Maintenance Equipment to mitigate design uncertainties and technical risks associated with future DAT operations.

Reality gap mitigation at DTP2 is approached by model fidelity, calibration, control validation, and progressive physical testing.

A core mechanism of reality gap mitigation is structural deformation management. It is recognized that Remote Handling (RH) equipment experiences non-negligible deflections under load, which introduce discrepancies between nominal models and real behaviour. To mitigate this, calibration of the Cassette Multifunctional Mover (CMM) device and modelling of the DTP2 environment are required, ensuring that virtual and control models incorporate measured elastic and geometric deviations. This reduces the gap between planned trajectories and executed motions.

Second aspect is Virtual Reality (VR) - based validation coupled to the control system. The VR environment is not treated as a standalone simulator but as an operational tool integrated with the Remote Handling Control System to rehearse, execute, and analyse RH procedures. By validating task sequences and operator interactions against calibrated models, mismatches between simulated assumptions and physical constraints are identified early.

Third aspect is hydraulic and control system validation at component and system levels. Previous work mandates functional, integrated, and acceptance testing of hydraulic system components, and control software at DTP2. These tests ensure that control laws, stiffness characteristics, and dynamic responses observed in simulation remain valid in the real system, thereby closing the reality gap from an actuation and dynamics perspective.

Incremental demonstrations at DTP2 are used as a convergence mechanism. Repeated physical trials of cassette handling operations feed back into model updates, controller tuning, and procedural refinement. This iterative loop constitutes the non-vision-based reality gap mitigation strategy within the DTP2 framework.

Manufacturing Status and Technical Issues of the ITER Blanket Shield Block in Korea

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

The Blanket Shield Block (BSB) is a key in-vessel component of the ITER, providing neutron shielding and support for the vacuum vessel and ex-vessel components under severe nuclear, thermal, and electromagnetic conditions. Korea Domestic Agency (KODA) is responsible for the manufacturing of 220 BSBs and has established the necessary fabrication infrastructure, qualification procedures, and quality assurance system to meet ITER nuclear-grade requirements.

Several technical issues were identified during the manufacturing of the BSB, including welding, non-destructive examination (NDE), drilling, hot helium leak testing, and final cleaning. Welding for several tens of cover plates with complex configurations posed challenges in deformation control, residual stress management, and dimensional accuracy, affecting interface compatibility with adjacent in-vessel components. In parallel, extensive NDE such as ultrasonic and penetrant inspections was applied to ensure weld integrity, while deep drilling processes required enhanced control to maintain tight tolerances and surface quality. The hot helium leak test demanded strict leak-tightness control to satisfy ITER acceptance criteria, and final cleaning was treated as a critical process to remove contaminants and meet in-vessel cleanliness requirements. Through iterative process optimization and strengthened quality and dimensional control, these technical challenges were effectively addressed, supporting stable series production of the Blanket Shield Blocks.

The manufacturing experience and technical solutions developed by KODA demonstrate its capability to produce large, complex, and nuclear-grade fusion components. The lessons learned from the BSB manufacturing activities provide valuable input for future fusion reactor projects requiring reliable and reproducible fabrication of in-vessel components.

Assessment of inflatable seals concept for the ITER Remote Handling equipment

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Oral

Abstract

The maintenance of the ITER In-vessel components such as Divertors, Blankets and Port Plugs will be performed using Remote handling equipment. The extraction of these irradiated components requires a robust confinement strategy to prevent the spread of contamination during docking and transfer phases.
This work presents the results of a pre-qualification phase of a flexible sealing concept based on the actuation of extruded inflatable seals providing a confinement barrier. The same type of seal is implemented on the container to extract in-vessel components, called the Cask Envelope System (CES) and on the temporary door isolating the ITER vacuum vessel, referred to as the Cask Docking Flange (CDF)
Details of the design of the system mock-up, the inflatable seals and the Leak Survey and Pressure Control System (LSPCS) are provided in this paper. Furthermore, experimental results on the sealing performance of the inflatable seals evaluated by the pressure change method described in ISO 10648-2:1994 are reported. These results include the sealing performance of the mock-up tested under different operational conditions including various internal vacuum pressures, ambient temperatures and potential door misalignments.

Title Front face integration of the ion cyclotron travelling wave antenna in the main wall of WEST

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

The integration of large size components in the main wall of magnetic fusion machines is a significant conceptual challenge. Key design laws and numbers to this challenge are investigated here, for the WEST Travelling Wave Array (TWA) antenna.
The TWA concept aims at addressing key drawbacks of ion cyclotron resonance heating systems for future next step devices, by improving coupling conditions, reduce voltages and currents in the antenna, and improve reliability, availability, maintainability and inspectability.
The TWA concept raises however a wall integration challenge, because of the large required toroidal angle extension requirement (>2pi/N, N being the number of toroidal field coils N=18. For WEST, 2pi/N = 20°) compared to the typical equatorial port size (

Imitation of maintenance skills by autonomous agents

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Oral

Abstract

In the field of fusion technology, the allowable margin of error in positioning during maintenance and repair operations is extremely small. Recent machine-learning approaches that aim to teach robots to perform high fidelity tasks in complex environments—such as a DEMOnstration power plant—have faced challenges in balancing adaptability and accuracy. The proposed solution seeks to imitate the skills of human operators and further enhance them.

This study focuses on teaching the peg insertion skill to robot manipulators. A two stage approach is employed, combining kernelized movement primitives, collected using a haptic device, with adversarial inverse imitation learning. The objective is to replicate and refine the dynamics of the robot system in fusion environment. Training is conducted in a high fidelity simulation and supplemented with additional learning in the real environment to compensate for discrepancies between simulated and real world physics. Extensive ablation studies are performed on individual system components, and multiple variants of the proposed approach are evaluated and compared.

Recent Developments of the STEP Fusion Powerplant Divertor Cassette Design to Meet Integration with Other In-Vessel Systems

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

The concept design of the STEP Fusion powerplant Divertor has evolved significantly in recent months following a range of integration work with other in-vessel systems. The double null configuration seen in this reactor introduces a number of challenges which are not identical for the upper and lower Divertors. For example, the presence of a split in the upper Divertor results in an inboard and outboard cassette. This complements the STEP Fusion powerplant vertical maintenance strategy which considers simplifying the lifting sequence of different replaceable modules as well reducing the need to cut and reweld coolant pipes which serve the Divertor and Inboard build.

The high levels of neutron irradiation seen at the Divertor means that shielding performance is critical and therefore must be optimised where possible. This became a far more important issue with an evolving Divertor design, where shielding performance was worsening due to removal of structural material to accommodate service pipes and support structures. Modifications to the Divertor wall profile also contributed to this by making the Divertor cassettes thinner. Changes to the Divertor coolant (from gas-cooled to water-cooled) helped to address this due to the improved shielding capabilities of the water coolant. The material selection was also changed (from RAFM steel to austenitic steel) to satisfy the lower working temperatures of this coolant.

Neutronics analyses were performed on a range of Divertor configurations with respect to volume fractions of structure, shielding and coolant, and the most cost-effective option was adopted while maintaining sufficient shielding performance. Further assessments were made to the impact on waste, where it was found that switching coolant could reduce the overall shielding material requirement with minimal waste impact on the whole machine. The lower-grade heat from the water coolant reduces the amount of useful heat provided to the power generation cycle, but this is offset by the overall reduction in integration complexity of utilising the same coolant (at similar temperatures) as other in-vessel systems.

Details of how the Divertor cassette modifications are advantageous to the in-vessel architecture are presented and are backed up by results of neutronics analyses.

STEP Fusion Powerplant EM modelling strategy for off-normal events affecting steady state operations

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

Reliable prediction of electromagnetic (EM) loads during both steady-state and off-normal operation is essential for ensuring the structural integrity and maintainability of fusion reactor components. Transient EM events (such as Vertical Displacement events) and EM plasma instabilities can generate significant EM loads on in-vessel structures. Understanding these loads is therefore critical for supporting robust component designs and ensures they can withstand the EM loads applied to them.

To address this, STEP Fusion programme has developed a comprehensive EM modelling strategy using a tiered approach to modelling fidelity, ranging from fast-running analytical tools to full reactor-scale finite element (FE) methods. One tier in this strategy, is a finite-element “Helmholtz Coil” model implemented in Ansys Maxwell. Using plasma simulation derived magnetic fields, simulated Helmholtz coils are used to produce the required magnetic field in the model. The component of interest is added to the Helmholtz model, to determine the resulting forces and torques. The benefit of this modelling approach is a significant reduction in simulation time compared with a full tokamak-scale FE EM model.

Initial validation of the Helmholtz Coil model approach is carried out using a 3 axis Helmholtz Coil test rig, in which measured loads on components are compared against simulated predictions. These experiments help validate the EM modelling tools used to simulate EM loads on components.

Preliminary Design of In-Bore Cutting Tool for DN200 in-vessel pipes

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

The Breeding Blanket (BB) of tokamak machine with a vertical upper port configuration is responsible for heat extraction and tritium production. It is connected to the primary heat exchange system through a dense network of pipes, routed through the upper port and upper port annex, that must be periodically cut and welded in highly radioactive environments with extremely limited accessibility to allow for the blanket segments replacement. Because of the high number of these pipes, their maintenance is one of the most critical challenges in remote handling operations [1]. Moreover, due to the confined spaces between the pipes, the cutting and welding operations near the breeding blanket segments cannot be performed from the outside of the pipes. It is therefore necessary to develop fully automated tools able to carry out these operations from inside the pipes.

In this context, this article presents the development of the preliminary design of the IBCT (In-Bore Cutting Tool) for DN200 pipes, which must be able to cut the pipe from the permanent structures and also prepare the joint for subsequent welding. The engineering approach adopted for the development of the tool followed a multi-phase methodology.

The process began with an analysis to identify the functional and operational requirements that the device must address. Subsequently, a campaign of cutting tests was conducted. This phase was essential to define the cutting strategy, the type of insert to be used, and the design loads, in particular the cutting forces and torques applied to the tool components during the working phase. Starting with a pre-existent concept design [2] a new design was developed that meets all the requirements defined. In parallel, the control system architecture was defined by mapping all the input and output signals necessary for the complete operational management of the tool.

This study represents a fundamental first step toward the realization of the complete tool, addressing the complex challenges imposed by this harsh environment and setting a solid base for future prototype development, Furthermore, engineering solutions may be helpful to develop cutting tools for smaller pipes.

#13283

Thermo-Mechanical Analyses of W7-X Cryostat System in Preparation for Long Pulse Operation

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

The Wendelstein 7 X (W7-X) cryostat system includes key components such as the plasma vessel (PV), outer vessel (OV), connecting ports, thermal insulation, vessel supports, and the machine base (MB). Its primary function is to maintain an insulating vacuum around the cryogenic magnet system while enabling external access to the plasma vessel via ports for diagnostics, media supply and heating systems.
As planned, the W7-X experiment schedule is now entering phases with increasingly long pulses (up to 1800 s). Although the W7-X cryostat system was designed to cope with long pulse high power plasma operation, recent modelling and operational results triggered a number of new, specific tasks to identify and reduce operational risks for the long pulse. The main tools for these calculations are the Global Model of the Cryostat System (GMCS), created in 2005 and the Plasma Vessel Thermal Global Model (PVTGM) from 2020.
The GMCS originally considered two extreme thermal load cases to mimic plasma operation conditions. A uniform PV temperature of 20°C and of 80°C, the latter simulating long pulse high power plasma operation. The 80°C boundary condition represents the maximum allowable heat load from the PV wall into the cryostat when operating the super conducting magnets. The assumed uniform temperature distribution at 80°C has proven unrealistic by the more recent thermal analysis of the PV using the PVTGM. The thermal analysis shows hot spots exceeding 80°C near ports.
An update of the outdated GMCS with respect to finite element types was performed to ensure compatibility with the latest FEM software release. In addition the application of the PV temperature map for long plasma pulse operation as well as an update of mechanical port loads simulating the port installations were done.
Preliminary structural analysis of the updated GMCS, applying the more refined PV temperature distribution, revealed an asymmetric shift of the vessel of more than 3 mm compared to a uniform PV temperature. As this displacement affects first wall components attached to the PV, it may require mitigation through magnetic field adjustments and the plasma control system.
Additionally, the updated GMCS predictions for a pure dead weight load case will also be compared with room temperature measurements done in 2025 at the entrance of three PV big port openings. The objective is to assess PV temperature distribution during long-pulse high-power plasma operation and evaluate the risk of significant first wall deformations in the upcoming operational phases.

Experimental Validation of a Mechanical Multi-Pipe Connection for Blanket Maintenance

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

Maintenance of fusion reactor in-vessel components like e.g. breeding blankets require the reliable remote disconnection and reconnection of multiple coolant pipes inside the vacuum vessel under highly constrained environmental conditions. A Mechanical Multi-Pipe Connection (MPC) concept has been developed to enable simultaneous disconnection of several pipes using a compact flanged connection with bolted preload application.

Recent design iterations focused on optimizing flange geometry, bolt arrangement, and seal layout to ensure homogeneous sealing force distribution while maintaining compatibility with remote handling constraints and limited spatial envelopes. Finite Element Analyses (FEA) were conducted to evaluate stress distribution, flange deformation, and gasket compression behaviour under operational preload and thermal loading conditions.

This work presents the development of a dedicated experimental test bench for mechanical and leak-tightness assessment of the MPC concept. During testing, bolt preload is applied in a controlled manner while helium leak rates are continuously monitored to evaluate sealing performance. Following the tests, the components are examined for permanent deformation to assess structural limits and compare observed behavior with numerical predictions.

The study aims to evaluate the correlation between numerical results and experimentally observed sealing performance and post-test deformation patterns. Particular attention is given to potential deviations caused by friction effects, assembly procedures, and modelling assumptions. The findings provide input for further refinement of both the simulation model and the mechanical design, supporting the development of a robust remote handling compatible pipe connection.

Induction coil development for in-bore brazing of DEMO DN200 pipes

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

The Demonstration Power Plant (DEMO) first wall blanket components will be exposed to high neutron fluence and high levels of thermal energy, the latter being removed by a coolant fluid transmitted out of the reactor via service piping. Due to the thermal and nuclear loading on the first wall, the components will need to be periodically removed and replaced. Installation of new blankets requires a separatable joint, where the brazing can be a reliable alternative.

Brazing is one of the alternatives that are considered for joining the service pipes of the DEMO breeding blankets. An induction brazing test rig has been designed, manufactured and assembled to test the feasibility of this technology within the DEMO constraints. The operating range of the induction power supply used in this article is low in terms of inductance. In order to heat the Diameter Nominal (DN) 200 pipes used in the DEMO to the appropriate temperature for brazing, several types of induction coils were tested. During the heating tests, the coil was continuously optimized, and thus the set goal was achieved.

Last year, a 3D metal printed copper inductor made with new technology was tested to heat a DN 200 pipe. Numerous tests have been performed, from which it can be concluded that the inductor developed by the HUN-REN Centre for Energy Research could be suitable for heating the DN 200 pipe. The induction coil 3D geometry of the induction coil was designed to optimize the DN 200 geometry and to maximize the heating power of the power supply.

The article presents FEM (Maxwell and Thermal) simulations and other additional calculation methods what helped to develop an optimal induction coil geometry for the DN 200.

The article will also detail future geometric developments that could facilitate the brazing of these large diameter and thick wall pipes.

#13443

Remote Handling approach for Volumetric Neutron Source Divertor Replacement

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Oral

Abstract

In 2024, a panel of experts confirmed the feasibility the feasibility of the Volumetric Neutron Source (VNS). Since then, design studies of the machine have been undertaken in parallel with the development of a remote handling strategy for replacing the in-vessel component. Previous studies addressed a potential strategy for blanket replacement, focusing on contamination control and the availability of the system and decontamination, as well as standardization of components. In the current phase, the integration of the main system into the lower ports is being developed, and a strategy for replacing the divertor system is being studied based on the assumptions adopted for the blanket remote handling. The present work focuses on the development of the overall sequence for replacing the divertor system according to the current configuration of the lower ports, the definition of a preliminary list of tools required according with the current status of the integration, the conceptual design of the cask mover that will be adopted for the service of all required casks .This article describes the overall sequence for replacing the divertor, the configuration of the lower ports, the definition of a preliminary list of tools, the conceptual design of the lower port cask, including its interface to the Vacuum Vessel port and its mover that is planned to be adopted for the service of all lower port casks.

Fusion Orientated Robotic Technology

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Oral

Abstract

Fusion Orientated Robotic Technology (FORT) programme is a 4-year UK strategic robotics initiative, designed to advance robotic capability in developing radiation-tolerant technologies essential for future Fusion facilities. FORT focuses on delivering innovative, radiation-hard solutions across five critical robotic subsystems: Actuation, Communication, Automation, Sensing, and Processing. These technologies represent the components most vulnerable to failure in extreme radiation environments. Therefore, constitute the primary barriers to deploying reliable autonomous systems in Fusion reactor facilities and Fusion maintenance plants.

A key objective of the programme is to stimulate and elevate industry capabilities, particularly small and medium enterprises in the UK and Europe. This is done engaging them directly in the design, manufacture, and integration of radiation-hard technologies. This approach not only strengthens the UK and European supply chain but also ensures that the resulting innovations are industrially relevant, scalable, and aligned with long-term fusion sector needs.

FORT's technical themes each deliver targeted individual impacts. The actuation workstream will integrate a radiation-hard actuator with onboard motor control into a commercial robot platform, leveraging modular robot architecture topology. The communication workstream will develop a low latency wireless radiation-hard electronics prototype. This aims to reduce cabling complexity on remote machines through digitisation, supporting major cost savings in large fusion facilities. Automation efforts will expand AI and machine-learning capabilities for robotic radiation hard vision systems, enabling object recognition, object avoidance, and autonomous operation including grasping for robotic technologies. Complementing this, the sensing and processing workstream will integrate radiation-tolerant technologies to support real-time computation within the Radiological environment that will in turn collate environmental and operational data acquisition to support predictive maintenance.

Strategically, the programme recognises the need to balance long-term innovation with practical, scalable solutions. While custom semiconductor development remains an option, the programme increasingly favours a hybrid approach that combines commercial Radiation hard by design electronics, with radiation hard by nature components that have been determined acceptable via Radiological test campaigns. This pathway offers lower cost, greater design flexibility, and opportunities for new IP in materials science and shielding architectures, while avoiding the risks associated with bespoke chip development.

Ultimately, FORT aims to deliver a convincing robotic demonstrator while establishing a robust ecosystem for radiation-hard technologies. By generating a validated component handbook, strengthening industrial capability, and enabling faster, safer, and more automated robotic operations, the programme will play a pivotal role in shaping the technological foundations of future Fusion power plants.

Evaluating Deterministic Transport for LIBRTI Breeder Blanket Neutronics: A comparison with Monte Carlo Methods

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

Deterministic discrete ordinates (SN) methods offer potential computational advantages for fusion blanket neutronics, particularly where rapid evaluation of evolving breeder blanket concepts is required. Within the LIBRTI programme supporting fusion reactor programmes like STEP, accelerated yet reliable modelling tools are essential for exploring design space. However, deterministic transport methods remain less widely applied to the assessment of key blanket performance metrics, such as tritium breeding ratio (TBR), when compared to established Monte Carlo (MC) approaches.

This study presents an initial assessment of the deterministic solver OpenSN (v1.0.0, Texas A&M University) for breeder blanket applications. OpenMC is used as a high fidelity reference tool to quantify differences in predicted neutronic responses and to identify the numerical and modelling factors responsible for any observed deviations. A two stage evaluation strategy is adopted. First, solver performance is examined using selected fusion shielding benchmarks from the SINBAD database to establish confidence in the SN methods under fixed source, deep blanket geometries with fusion relevant conditions.

Subsequently, both methods are applied to a simplified breeder blanket configuration representative of published and well understood blanket concepts. Neutron flux spectrum and TBR predictions are compared to quantify deterministic biases and assess sensitivity to angular and spatial discretisation.

This work aims to determine whether deterministic transport method provide sufficiently accurate predictions for pre-concept design exploration space. The findings establish a foundation for future investigations into uncertainty quantification, sensitivity analysis, and topology optimisation aimed at enabling efficient neutronics informed blanket development.

Towards a unified ROS2-based architecture for heterogeneous robotic systems in fusion remote handling and inspection applications

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

Remote handling and in-vessel inspection systems in nuclear fusion facilities are characterised by long operational lifetimes, heterogeneous hardware, and significant hardware and software legacy. These constraints limit the reuse of control architectures across platforms and complicate the adoption of modern robotic software frameworks, such as widely used Robot Operating System 2 (ROS2), a robotic middleware for distributed robot control. This paper presents a ROS2–based control architecture designed to address these challenges and its application to two contrasting robotic systems used in fusion environments.

The architecture combines several components within ROS2. First, a hardware abstraction framework (ros2_control) that standardises actuator and sensor interfaces is integrated to handle control and monitoring of hardware equipments. Then, a software framework (MoveIt2) providing kinematic modelling, collision checking, trajectory planning is used for generating and executing collision-free motions. A behaviour-tree-based orchestration layer (a hierarchical task execution framework) was also used to separate high-level operational logic from low-level robot control. The approach was first implemented and validated on a 6-axis industrial serial arm manipulator performing pick-and-place operations of welding equipment and,serving as a reference deployment under controlled integration conditions.

The same architectural principles are currently being applied to the Articulated Inspection Arm (AIA), an in-vessel inspection robot operating in the WEST tokamak. The AIA is a unique and long-serving system whose evolution reflects extended operational use across multiple development phases. As part of this work, a harmonisation process is being undertaken to align existing control interfaces with more standardised interfaces. This paper presents the ongoing harmonisation effort, with particular emphasis on the mapping of legacy control interfaces to ROS2 concepts and on the reuse and extension of the control and planning layers developed for the industrial platform.

By contrasting a validated industrial deployment with an evolving legacy system integration, this work highlights both the strengths and the limitations of applying a unified ROS2 architecture to heterogeneous fusion remote handling robots. The paper concludes with lessons learned and recommendations for future fusion robotic developments.

Conceptual and Joint-Level Mechanical Design of a Long-Reach Tendon-Driven Manipulator for K-DEMO

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

An in-vessel maintenance environment of the Korea Fusion Demonstration Reactor (K-DEMO) requires access through narrow ports while covering a wide in-vessel working area. Under these requirements, the design of a manipulator must guarantee long-reach operation, rigidity, and operating stability. This paper presents a conceptual design of a long-reach tendon-driven manipulator targeting K-DEMO, along with several mechanical designs of joint mechanisms. We analyzed the environment and accessible port size of K-DEMO. We proposed a tendon-based conceptual design for a long-reach manipulator and developed three types of joint mechanisms. To implement the joint part, three mechanisms—multi-pulley-based, worm-gear-based, and bevel-gear-based—were designed, and their respective drive principles and structural characteristics were analyzed. The proposed framework systematically outlines design trade-offs for joint mechanism selection. A comparative mechanical evaluation reveals distinct structural and transmission characteristics among the three mechanisms, highlighting their respective advantages and limitations. The presented results provide a foundation for future control characteristic analysis and dynamic performance verification studies.

A preliminary Digital twin methodology for soft-real-time analysis of manipulator structural deformations during remote handling operations in compact tokamaks

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

Digital twin (DT) technology offers an innovative approach for testing remote handling tools and operations with graphics solutions that are extremely realistic. Above all, it provides the capability of simulating the characteristics of the real model (such as deformations, vibrations and energy consumption) more accurately and effectively.
The primary challenge confronting the proposed manipulators for the Divertor Test Tokamak (DTT) is the need for a software environment capable of simulating remote handling operations in real time and in a scalable, reliable and secure manner. In particular, scalability is imperative since the objective is to devise tools that can evolve over time by adding new features or performing increasingly detailed simulations. Reliability and safety are also essential aspects for the implementation of control systems that operate manipulators inside tokamaks.
An infrastructure based on an IaaS (Infrastructure as a Service) approach constitutes a perfect solution for the needs of the digital twin. This approach facilitates the reallocation of the module executing the tasks to a different node in the event of a failure. Furthermore, the software allows the distribution of calculations across multiple nodes if such a configuration has been properly envisaged.
The software development process is based on a modular approach. The collision module is one of the first modules to be developed. The scalable paradigm will enable the use of highly detailed meshes and the implementation of software real-time deformation functions. The meshes generated to quantify the deformations will be those displayed and used in inverse kinematics with obstacle avoidance. In conventional robotic control and simulation systems, this is approximated by external volumes containing the manipulated arm/object. Such an alternative is not viable when clearances are tiny as in the case of compact devices such as DTT.
The validation of the system will be conducted utilising two arms, Farm and Ivros, which have been successfully operated on the FTU tokamak. The data to be considered for the subsequent transfer to DTT remote handling will indeed be those acquired during the FTU maintenance campaigns. The already available data will be combined with new experimental campaigns that will use laser-tracking systems capable of detecting joint deformations with greater precision. The acquired data will be utilised to train a neural network, which will subsequently be deployed to calculate arm deformations in soft real time. The DT model may evolve over time thanks to the implementation of machine learning solutions for sensor data analysis.

WEST Operation Shutdowns Schedule Optimization: Focus on Divertor Technical Interventions

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

WEST is a superconducting tokamak capable of long-pulse operation and equipped with an ITER-grade actively cooled divertor tungsten. It provides a platform for a Plasma-Facing Components research program aimed at testing ITER divertor technology. To address risks associated with both industrial-scale manufacturing and the operation of such components, annual maintenance and divertor interventions are required. Rigorous control of all these activities is crucial to ensure the efficient operation of such a large-scale research facility and a high level of tokamak availability during experimental plasma campaigns. The experience gained over the past ten years through twelve experimental campaigns and their associated shutdown phases has led to the development of a robust shutdown scheduling strategy and strong technical control of divertor interventions.

In order to carry out the study of the tungsten divertor exposed to high heat loads, technical interventions on the divertor usually lie on the shutdown's critical path. An overview of these various interventions carried out on the divertor, divided into 12 sectors (30°), each comprising 38 Plasma Facing Units (PFUs), during an operation shutdown is presented. The divertor intervention sequences: removal of the divertor sectors from the vacuum vessel, their modification, their qualification and their reinstallation, over successive campaigns have been progressively optimized to reduce overall shutdown duration. Although most of this work must be carried out ex-situ from the vacuum vessel, an effort is now made to perform activities in-situ in order to save time and to be more ITER-relevant (laser cleaning or microscopic observations of material degradation). These interventions address several material-focused objectives, including the replacement of selected PFUs to evaluate new technologies, detailed characterization of plasma-facing components in support of the scientific program, and the development of methodologies for evaluating material degradation and component lifetime under high heat load conditions.

This study presents a comparative analysis of shutdown schedules highlighting the time savings from the opening of the plasma vacuum vessel to the commissioning phase before an experimental campaign. Improvements concern technical operations, coordination of concurrent activities and technical resource allocation. The scheduling approaches and resource management strategies adopted to improve execution efficiency, minimize interfaces between activities, and mitigate risks are also detailed.

Overall, this work highlights the diversity of interventions carried out during WEST operational shutdowns and demonstrates how structured shutdown scheduling and effective activity coordination contribute to risk mitigation and ensure the annual availability of the WEST facility for research on ITER divertor technology.

#13630

Development of the Vacuum Vessel for the COMPASS Upgrade Tokamak

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Oral

Abstract

COMPASS Upgrade is a medium-sized tokamak operating at high magnetic field (Based on operational requirements, the final design of the vacuum vessel was developed over the past six years. In total, 22 of the most critical scenarios were identified, considering combinations of vertical displacement events, current and thermal quenches, and halo currents [1]. The latest electromagnetic and thermo-structural models predict a vertical force up to 3 MN, a lateral force up to 1.5 MN, and a toroidal moment of ~50 kNm during the most severe disruptions. The highest stresses induced in the vessel structure may exceed 300 MPa under the highest operating parameters. In addition, thermal expansion of the vessel of up to 17 mm during high-temperature operation was taken into account. These results served as the main drivers for the final vacuum vessel design.
The nickel-based alloy Inconel 625 for the main body and heat-resistant stainless steel (AISI 309S/310S) for flanges were selected with wall thicknesses ranging from 23 to 35 mm. In total, 72 ports of various geometries provide access for diagnostics, heating, and maintenance. Flexible gravity supports accommodate thermal expansion during high-temperature operation. To ensure the thermal balance, in-vessel heating loops using gaseous helium at a pressure of 60 bar were integrated. The total weight of the vacuum vessel assembly is approximately 7.4 tons.
A thorough welding study was performed to select suitable welding techniques, evaluate induced deformations, and identify manufacturing risks. The key welding techniques selected include electron beam welding (EBW), tungsten inert gas (TIG), and metal active gas (MAG). Additional material studies were performed to characterize the base materials and tailor their mechanical properties. The first phase of production, material acceptance, is completed and welding of the vessel main body is currently ongoing. According to the procurement plan, the vacuum vessel is scheduled to be delivered in early 2027.
[1] J. Hromadka, et al., Fusion Engineering and Design 167 (2021)

Investigation into State of the Art of Remote Handling Compatible Bolt Threaded Connection Fastening Tooling Design

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

Maintenance of a fusion power plant in-vessel components includes the removal and replacement of plasma facing components, for example the breeding blankets (BB). For access, multiple coolant pipes need to be removed. A mechanical multi-pipe connection (MPC), which consists of multi-pipe flange connected with a threaded bolt connection allows removal of multiple pipes at the same time using remotely operated mechanical connections. To fulfil maintenance requirement of a fusion power plant such as DEMO, the MPC must be designed to be remote handling compatible, which includes the operation of the threaded bolt connection.

This paper delves into the current state of the art of remote handling compatible fastening solutions for application inside fusion reactor bioshield environment. Major challenges for the development of bolt fastener tool includes the highly constricted space inside the vacuum vessel, which limits the overall dimension and movement of the tooling, and the environmental condition inside vacuum vessel which includes elevated temperature and high radiation condition.

This study covers aspects such as fastening mechanism for single bolt or multi-jackbolt connections; bolt transport from maintenance hatch into MPC position on the BB, positioning and aligning methods, whether using electro-mechanical solutions or computer assisted solutions by means of visual or tactile feedback; remote handling compatible tooling design; and deployment feasibility study.

Keywords: Mechanical Pipe Connection, Maintenance Tooling Design, Remote Maintenance

#13649

Design of Remote Maintenance Robotic System for HL-3 Tokamak Divertor

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

Future fusion reactors face dual challenges of 14.1 MeV high-energy neutron irradiation and extreme heat flux of 10-20 MW/m^2, causing activation and vulnerability of in-vessel components. Radiation dose exceeding safety limits in maintenance zones renders traditional manual operations completely infeasible. This study presents a comprehensive remote handling test system specifically designed for HL-3 tokamak divertor, aimed at validating the engineering feasibility of fully remote maintenance technology.

The system adopts a hierarchical robotic architecture comprising five major components: simulation verification facility, remote maintenance robot body, end-effector tooling, control system, and digital twin system. The robot body integrates a maintenance port translation mechanism, a 4-DOF heavy-duty manipulator (payload ≥100kg), and dual 7-DOF radiation-hardened serial manipulators (reach 1.4m, payload 20kg each), enabling ±45° toroidal movement and flexible obstacle avoidance in confined spaces. Seven categories of modular tools are equipped, including pipe cutting/welding, bolt installation/removal, vacuum leak detection, dust extraction, hoisting, and 3D contour scanning.

A full-scale 90° mock-up vacuum vessel and three divertor modules (65kg each) have been constructed. Non-chip roller blade cutting technology and automated argon arc welding are implemented, integrated with high-precision 3D vision system to achieve millimeter-level remote operation accuracy.

Performance specifications: absolute positioning accuracy of end-effector ≤2mm, repeatability ≤1.0mm, divertor installation contour accuracy ≤1.5mm, force feedback accuracy ≤5%, vision system resolution ≥4K. Simulation validation demonstrates successful completion of 12 typical operation positions including vacuum vessel entry, pipe welding/cutting, bolt installation/removal, dust extraction, helium leak detection, and camera inspection.

System engineering approach for Tokamak component replacement under ALARA requirement: ITER TBM case

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

A major challenge for the Tokamak plant systems is to develop and implement, early in the design of the facility, maintenance requirements. ITER plant shall be designed so that occupational radiation exposure can be As Low As Reasonably Achievable (ALARA) over the life of the plant to comply with the policy for the ITER Plant Maintenance. Much of the maintenance of ITER will be performed in controlled area where personal would be exposed to nuclear hazards. In this paper, the TBM plant will be used as a study case.

The objective of the ITER Test Blanket Module (TBM) program is to provide experimental data on the performance of the breeding blankets in the integrated fusion nuclear environment. The ITER test blanket modules are installed and operated inside the Vacuum Vessel (VV) at the equatorial ports located within Port Plugs (PP), and each PP includes two TBMs. After each 18-month-long plasma operation campaign, the TBM research plan testing program requires the replacement of the TBMs with new ones during the ITER long-term shutdown, called Long-Term Maintenance (LTM).

The activity to perform TBM port cell equipment replacement shall comply with the need to minimise doses for occupational exposures using the recommendations from International Atomic Energy Agency using appropriate means to reduce personal exposure (Simplification of operations, collective or personal protections, remote means, …).

To implement ALARA process requirements, design activities shall consider careful integration investigations starting from the early phase to address all engineering aspects of the replacement sequence.

This paper proposes a System engineering approach to manage and control Tokamak plant design process, focusing maintenance operations under ALARA requirements. This paper describes also its implementation for the TBM port cell equipment replacement case where it shows benefits to manage design stakeholders, including transverse functions concerns such as safety or maintenance.

Keywords: Remote Handling, occupational exposures, ALARA, Fusion Devices, System Engineering, TBM, Verification and Validation.

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

DTT Port Plug Structure: definition and preliminary design

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

In the framework of the Divertor Tokamak Test (DTT) facility project, a structural component to be installed in several vacuum vessel ports is foreseen: the Port Plug (PP). The primary function of the PP is to host diagnostics, as well as selected auxiliary vacuum vessel components, enabling their insertion into the machine and their direct interaction with the plasma. The PP must be fully compatible with remote handling operations for maintenance purposes, allowing for modifications, replacements, and repairs of in-vessel machine components.

This work first aims to identify the main functions of the PP and to define all its interfaces. In addition to integrating diagnostic systems, the PP provides mechanical support for radiation shielding, acting as an extension of the first wall, and serves as a primary vacuum confinement barrier. It is directly connected to the cryostat as a cantilever structure bolted to the cryostat flange and hosts vacuum conditioning components, including heating cables. Given the wide range of requirements, mainly driven by diagnostic systems, the standardisation of the PP is a key issue. The interfaces of the flanges have to be defined, grouping the systems that interact exclusively with the flange rather than with the entire PP, including remote handling requirements and the gasket design.

Due to its several interfaces with multiple machine systems, the PP is a key component and its design crucial and non-trivial. Accordingly, a further objective of this work is the definition of a preliminary PP design based on initial analyses. Structural integrity assessments and clash analyses under deformation are performed by applying load combinations representative of machine operating states, including the most severe conditions, such as plasma disruptions.

Development of End Effectors and Tools for Initial Assembly of ITER Blanket Module

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Oral

Abstract

This paper reports the development of End Effectors (EEs) and Tools for Blanket Modules (BMs) assembly inside the ITER Vacuum Vessel (VV). Each BM consists of a Shield Block (SB) and First Wall (FW), and 440 BMs exist in total. The SBs are installed onto the VV. Then the temporary FWs (TFW) onto the SB prior to the Start of Research Operation (SRO). The TFWs have no cooling-water piping and to be replaced by the final FWs having water cooling, after the SRO for long-pulse operation in the DT plasma operation. Assembly of the BMs in the VV requires the EEs and Tools for four key operations: (1) transportation of each BM using dedicated grippers, (2) fixation of the BM by bolt-torquing, (3) welding of the cooling-water pipes inside the BM, and (4) cutting of pipes in the event of welding failure.

Development of these elements has been conducted for the BM initial assembly. The SBs, weighing up to 4.1 tons, and the 1.0 tons TFWs need transport with their dedicated grippers within the limited VV workspace. The grippers compatible with multiple variants and capable of temporarily fixing components at specified positions were designed. For the final fixation, torques up to 8.4 kNm are applied to the bolts to ensure structural integrity during plasma operation. Bolt-torquing tools were designed to achieve both high-torque capability and compactness for passing through the VV port with an identical tool accommodating multiple variants. After that, welding is performed between cooling-water circuit components located approximately 200 mm deep in a narrow space behind the SBs. Welding tools were designed to enable in-bore welding and allow observation of the bevel condition or welding results. In case of welding failure, cutting tools minimizing swarf drop into the VV were developed.

The development of the EEs and Tools for the BMs initial assembly has been steadily progressing. The designs of the grippers and bolt-torquing tools have been completed, and their prototypes are under fabrication. Feasibility evaluation tests of the welding and cutting tools were conducted using prototype tools manufactured according to the design results. Knowledge acquired through developing remote-handling EEs and Tools has been effectively leveraged for efficient development of the initial assembly EEs and Tools. The development of the initial assembly EEs and Tools marks an important step toward the successful construction of ITER and provides a foundation for future remote maintenance.

A Calibrated VR-Based methodology for accurate Trajectory Planning in ITER Remote Handling

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

Remote Maintenance is essential for ensuring the safe, reliable, and long-term operation of fusion reactors. Among its objectives, the ITER project must demonstrate that a Remote Handling System (RHS) can be successfully designed, integrated, and operated to execute both scheduled maintenance activities and unplanned recovery or rescue interventions in the event of failures or abnormal conditions. Because the tasks involved are highly complex, the ITER Remote Handling Control System (RHCS) adopts a man-in-the loop approach combining automated and teleoperated procedures. In addition to live camera feeds, operators are supported by a Virtual Reality (VR) System to make better-informed decisions during trajectory planning and while supervising or teleoperating Remote Handling equipment. To meet the reliability required in these tasks, a Structural Simulator has been developed to calibrate the VR models against real-world behavior. This is achieved by enriching the VR representation with Virtual Joints that reproduce the robot's dynamic pose under payload and self-weight effects, which cause deviations from the rigid nominal configuration typically assumed in VR environments.

This abstract proposes a methodology for Trajectory Planning in Remote Handling operations based on the calibrated VR models. The approach is demonstrated on the Cassette Multifunctional Mover (CMM), the 5-DoF robotic system used for ITER Divertor handling, which must safely manipulate the 10-ton cantilevered payload of the Divertor Cassette within highly confined spaces, where clearances may be limited to only 20 mm. The proposed method makes use of two distinct kinematic models: the nominal model, implemented in the GenRobot controller and operated through the Command & Control (C&C) GUI, and the real robot's VR model, enhanced with Virtual Joints automatically updated from nominal joint values to account for dynamic deflections. The latter is used both to monitor clearances along the motion and to define the poses of trajectory waypoints in the real robot space. Once defined, the waypoints are stored for the nominal robot space in a trajectory file, executed as a Linear Move command. Since this command enforces linear motion for the Tool Centre Point (TCP) of the nominal robot model, the TCP of the real robot deviates from the ideal straight path between consecutive waypoints. For this reason, a trajectory linearity analysis is performed, and any segment showing excessive non-linearity is subdivided into smaller subsegments, increasing the number of waypoints until a maximum deviation of 0.2 mm from the intended linear path is achieved.

Development of high-load Grippers for Initial Assembly of ITER Blanket Module

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

This paper reports on the development of grippers for installment of Blanket Modules (BMs) during the initial assembly inside the vacuum vessel (VV) of ITER. The BMs consist of First Walls (FW) and Shield Blocks (SB) with FW placed on top of the SB, and special-shaped Shield Blocks (15ND). The BMs are installed using dedicated grippers and temporarily fixed by bolts. The key design challenges are: (1) providing strength to withstand a weight of up to 4 tons and a moment load of up to 9.2 kNm, (2) achieving compatibility with 32 variants of FWs and SBs using a minimum number of grippers, and (3) providing tolerance to positional deviations between gripping holes and fingers, and between fixation bolt head and wrenches. Misalignment can lead to reduced contact area, resulting in stress concentration. The grippers shall meet the above requirements.

The development results of grippers to address these challenges are described with solutions for each issue. To withstand high loads, threaded pins of SB gripper engage with central bolt threads. The threaded pins protrude from the finger and engage with the threads, achieving both compactness and stable gripping. A cone-shaped gripping finger was developed to increase contact area between gripping finger and hole for the large moment loads. Performance evaluation tests were conducted on the cone-shaped gripping finger. The deformation amount of the cone-shaped finger was negligible compared to the geometric tolerance. Full-scale mock-up tests were performed using prototype FW gripper, confirming stable gripping. To accommodate the BM variants, The FW gripping fingers were equipped with a sliding axis to adjust to the different gripping-hole span. For the limited space of VV, dedicated FW gripper without a sliding axis was developed. The SB gripping finger has a vertical axis to adjust to the different gripping-hole depth. The configuration of the pad and temporary-bolting wrench was adjusted to SB variations in surface geometry and bolt-hole positions. To compensate for manufacturing and assembly tolerances, each gripper's temporary-bolting wrench and 15ND cone-shaped fingers were equipped with a misalignment compensation mechanism. The 15ND gripper incorporates a floating mechanism in one of its fingers, providing 1 mm of translational compliance and 0.6° of angular freedom. This design enables grippers to accommodate positional deviations during insertion.

The developed grippers will contribute to ITER initial assembly by achieving successful installation of BMs. The design knowledge obtained through this development will be applied to future remote-maintenance equipment.

#13795

Impact of Vertical Maintenance Strategy on Deliverability of the STEP Prototype Plant

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

Delivering a viable first-of-a-kind fusion prototype power plant requires balancing ambitious performance goals with realistic engineering constraints. This work examines how the choice of maintenance strategy directly affects the affordability, deliverability, and long-term commercial relevance of such a power plant. The STEP programme has chosen a Vertical Maintenance Strategy (removing large components vertically via clearance gained from re-mountable magnets). This decision is driven by the Concept of Operations' emphasis on demonstrating reduced maintenance durations, enabling rapid return to power generation for future commercial fusion power plants, whilst retaining sufficient flexibility compatible with a prototype plant.

The work highlights how maintenance requirements fundamentally shape overall plant architecture. The Vertical Maintenance Strategy introduces significant geometric, logistical, and structural constraints that influence building height, Tokamak component size and segmentation, service routing strategies, and the architecture of remote handling equipment. These constraints propagate into broader impacts on civil construction complexity, programme risk, capital cost, lifecycle cost, and decommissioning strategy.

Several Tokamak Building and Maintenance System layout options are assessed, each capable of delivering the core functional requirements for safe, reliable, and repeatable remote-handling operations. These options are evaluated for their potential to reduce capital cost, even out construction cost profiles, reduce programme risk, and expedite delivery timelines. The Vertical Maintenance Strategy is a key driver of plant availability, lifecycle cost, and the demonstrating of commercial power plant maintainability, and early integration of the strategy into the prototype power plant design is essential.

#13875

Lawson Machine 26 – Center Current-Conducting Shaft

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

LM26 is a large-scale magnetized target fusion (MTF) machine built to demonstrate that compressional heating of a spherical tokamak plasma can achieve fusion conditions. A coaxial helicity injector forms a magnetized plasma in a 5 to 7 cm thick cylindrical lithium liner inside a composite chamber. A surrounding 36-turn theta-pinch coil generates a strong magnetic field to symmetrically compress the liner and the deuterium plasma it contains in less than 4 milliseconds. This is done without damaging any major vessel structures, allowing the device to be rapidly restored for service.

To support repetitive plasma compression testing, each shot requires installation of a new lithium liner and center shaft assembly. The center shaft design and rebuilding process is critical for enabling LM26 program operations. The shaft interfaces with the injector, completing the machine's poloidal circuit and forming the plasma target volume into which the collapsing lithium liner compresses the plasma. Diagnostics are mounted to the in-vacuum components of the center shaft to provide critical measurements of the plasma and liner trajectory. During the compression, large dynamic loads generated by the impact of the compressing lithium liner must be contained within the center shaft assembly to prevent damage to other critical machine components.

The center shaft assembly process enables LM26 to operate as a high-cadence test platform by supporting efficient shaft assembly, diagnostic integration and calibration, and preconditioning of the shaft in a bake chamber prior to transfer into the machine using argon-filled bags. Multiple center shaft assemblies are staggered through this process cycle to ensure that immediately after a plasma compression test, a new shaft assembly is ready for installation, creating challenges between technical integration and feedback from test results.

This work describes the technical challenges of the LM26 center shaft design, including the high-current socket joint, structural load path, in-vacuum diagnostic mounting, while highlighting the multi-stage processing workflow developed to enable high-cadence plasma compression operations .

#13893

Design Update on the In-Bore Welding Tool for DN200 in-vessel pipes

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

The development of fusion power plants requires the definition of reliable remote maintenance strategies from the early design stages, together with their progressive integration into plant design digital twin. A critical issue is the remote maintenance of in-vessel service piping, which supplies coolant and other process media to reactor components. Although several industrial and nuclear-oriented pipe maintenance solutions are available, they are not directly compliant with the specific geometrical, operational, and environmental constraints of fusion applications.

Moreover, pipe cutting and welding strategies are still not fully consolidated, even in the most advanced fusion facilities. In this context, the availability of a dedicated facility capable of supporting the full maintenance workflow, from joint preparation and alignment to welding, inspection, and qualification, would be essential to address the problem in a systematic way, also considering that fusion plants typically include pipes with different diameters and wall thicknesses.

As a consequence, the development of individual maintenance tools is carried out together with the definition and validation of an end-to-end maintenance workflow. Within this framework, in-bore technologies represent a promising solution for pipe cutting and welding in space-constrained areas, such as upper-port service pipes. Building on previous work on the DN200 In-Bore Welding Tool (IBWT), this paper presents the latest design developments of the system, with a focus on engineering consolidation and integration aspects.

In particular, the paper discusses: (i) the redesign of the clamping system, supported by structural analyses aimed at improving stiffness, load transfer and stability inside the pipe; (ii) the revision of the overall architecture to improve maintainability and compatibility with remote-handling constraints; (iii) the development of an interface concept aimed at coupling the tool with the transporter system and ensuring the continuity of all the tool's utilities;

Special attention is also given to the identification of critical radiation-tolerant components and to design choices required for operation in fusion maintenance environments. The presented developments contribute to the maturation of the DN200 IBWT toward a more robust and qualification-oriented solution for future DEMO-relevant applications.

Decommissioning the DEMO Fusion Power Plant – Integrating Nuclear Dismantling Experience into DEMO Fusion Reactor Design

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Oral

Abstract

The successful construction and operation of the international fusion reactor ITER will represent a major step toward sustainable fusion energy. Experience gained from ITER will support the development of Europe's Demonstration Fusion Power Plant (DEMO), as the next reactor capable of producing electricity and bridging the transition to commercial fusion. DEMO construction is currently targeted for 2040–2050.

The decommissioning of nuclear power plants in Germany has provided extensive insights into regulatory frameworks, design requirements, and the technical challenges associated with dismantling. These findings are highly relevant as fusion energy advances, particularly for the design and future decommissioning of reactors such as DEMO. Germany's experience with nuclear power plant dismantling offers transferable knowledge for the emerging field of fusion decommissioning. This paper draws on this experience to inform design and operational strategies for future fusion systems, especially in the context of DEMO.

For fusion reactors, it is essential to integrate design features that enable efficient decommissioning, including suitable material selection, ease of disassembly, and effective waste management strategies. Incorporating such considerations during the design phase can significantly reduce future decommissioning challenges.

In the context of fusion, particular attention must be paid to the treatment and final disposal of radioactive waste. This paper provides an overview of Germany's existing and planned final repositories and their acceptance criteria. With this information, it is possible to assess whether waste generated from DEMO will meet disposal requirements for geological repositories for non-heat-generating nuclear waste—an essential factor in minimizing long-term environmental impact.

Framatome, a leading expert in nuclear power plant dismantling, has contributed substantial insights into the challenges and solutions associated with decommissioning. Their expertise has been instrumental in identifying key issues and developing strategies to address them. For DEMO, integrating these lessons early—covering material handling, radiation protection, and efficient disassembly—can streamline future decommissioning efforts.

Based on the specific DEMO design and Framatome's extensive experience in dismantling nuclear fission plants, a decommissioning plan has been developed outlining process chains, task sequences, waste treatment techniques and recommendations for the plants design process.

Progress in the design and the verification of the DTT vacuum vessel

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Oral

Abstract

DTT is one of the largest superconducting tokamaks with the mission to get scientific and technological proofs of power exhaust in prospect of the first nuclear fusion power plant.
The vacuum vessel containing the plasma is made of a double wall structure which acts as primary vacuum confinement barrier, radiation shielding, and provides support for in-vessel components such as the first wall, the divertor, in-vessel coils, and the in-vessel diagnostics. Eighty-two openings, or ports, in the vacuum vessel provide access for additional heating, fuelling, diagnostics, vacuum systems and services, installation and maintenance operations including remote handling.
The integration of in-vessel components involved design optimisation and virtual assembly simulation in order to sustain expected loads, merge supporting functions, and meet first installation and remote handling requirements in a compact, complex, and tight spatial environment. Reverse engineering is applied at interface supports of in-vessel components and of remote handling equipment in order to satisfy tight tolerances required by final position of divertor cassettes and first wall modules with reference to the tokamak global coordinate system.
Port plugs installed in the openings provide support for heating antennas and plasma diagnostics, and define interfaces with the cryostat, ex-vessel plants, and remote handling casks in the torus hall.
Load case combinations have been grouped in accordance with the RCC-MRx code distinguishing between a) normal condition (service level A) produced by operational loads such as mitigated disruptions and inter-shell pressure, and b) emergency/faulted condition (service level C/D) produced by unlikely/extremely unlikely loads such as severe disruptions and vertical displacement events. The vacuum vessel can safely withstand the expected loads (electromagnetic, gravity, pressure, earthquakes) and operate reliably throughout its designed life.
Pressure vessel design principles are applied to the inter-shell circulated by pressurised fluid, then essential safety requirements of the Pressure Equipment Directive are met through analyses and verifications against normal condition to eliminate/reduce hazards, and by implementing safety devices as protective measures. The process has been developed with the supervision of the Italian National Institute for Insurance against Accidents at Work considering the DTT vacuum vessel as a pioneering case study for the certification

of future vacuum vessels of nuclear fusion facilities.

Dimensional chain compensation, combined with the gradual replacement of virtual components with actual manufactured parts, welding procedure qualification, and in-process inspections will be monitored at the vacuum vessel factory in order to improve assembly accuracy, reduce rework, and optimise tolerances.

#13989

Fixation system mockup for VNS divertor: a systems engineering design framework

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

The Volumetric Neutron Source (VNS) is an experimental nuclear fusion reactor whose main purpose is to test and validate solutions for breeding blanket technology. The architecture of the machine, which has a plasma major radius of 2.67 m, integrates a single null divertor at the bottom of the vacuum vessel. The divertor spans radially from 1850 to 3330 mm and is composed of 36 divertor modules. Each divertor module system is attached to the vacuum vessel, using a semi-flexible preloaded mechanism that ensures physical contact on both radial sides i.e. on the inboard and on the outboard side. The fixation system ensures support against all loads that occur during normal and off-normal events including plasma disruptions and is compatible with the differential thermal expansion of the cassettes relative to the vacuum vessel. The concept of the fixation system of the VNS divertor is described here. For the verification and improvement, a testing and evaluation campaign is foreseen, which is outlined, too. It has been developed using a systems engineering approach, from requirements definition to modeling and analysis, highlighting the challenges of the mockup and possible improvement of the concept.

#13998

GRIP : A Single-bolt hydraulic connection for Plasma-facing components

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

Hydraulic connections for plasma-facing components (PFCs) in fusion reactors are traditionally made by welding or with multi-bolt flanges. Welded connections require cutting and re-welding for component replacement, while multi-bolt flanges occupy significant space, require individual access to each bolt, and result in lengthy assembly procedures. Both approaches need substantial access space around the connection. Each bolt access point represents an area exposed to plasma, creating additional vulnerabilities. The GRIP (Generic Rapid Interface for Plasma cooling) project addresses these limitations through a patented single-bolt connection concept designed for rapid component replacement and minimal plasma exposure.

The prototype is manufactured from 316L stainless steel using additive manufacturing, enabling enhanced design freedom particularly for cooling channel optimization. The hydraulic sealing system employs two coaxial Helicoflex® metallic seals that require a compression force of 84 kN. These seals were custom-designed by Technetics in collaboration with CEA's ISEC laboratory. A single M24 stainless steel silver coated bolt provides the clamping force. To minimize plasma exposure, the bolt is captive within the PFC and bears against an M36 nut. Both threaded elements are secured with Helicoil Screwlock inserts to prevent loosening under operational conditions. This design reduces the plasma-exposed surface to the minimum required for tightening tool insertion. The assembly and disassembly operations are performed perpendicular to the PFC surface, which is advantageous for walls composed of contiguous PFC modules.

A dedicated test bench was developed to qualify the bolting system and validate the concept. The first phase characterized the relationship between applied torque and compression force to ensure the 84 kN is achieved consistently. Following mechanical qualification, the prototype PFC was assembled with the Helicoflex® seals and tested for leaks using helium detection. Tests were performed at both operational pressure (30 bar) and hydraulic test pressure (45 bar) of the cooling circuit, at ambient temperature and at the tokamak baking temperature (200°C) to validate seal performance across the expected thermal range.

The validation campaign demonstrates that the GRIP concept achieves the required leak-tight performance while delivering the operational advantages of single-bolt assembly: reduced plasma exposure, simplified maintenance access, and accelerated component exchange. These results confirm the viability of this connection design for various PFC applications in future fusion devices.

#14082

Definition and Implementation of Dimensional Risk Management Strategy for the DTT Remote Handling Facility

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Topics

H. Remote Handling ; Vessel/in-vessel Engineering

Communication type

Poster

Abstract

In a magnetic fusion power plant, as well as in current experimental machines, the assembly phase is crucial for bringing together all the systems coherently aligned, resolving issues by targeted R&D activities. The component variability after manufacturing can cause deviations of the components from their nominal design with consequent installation misalignments. The proper positioning and alignment of as-built components to the main reference systems are primarily important for allowing the assemblability and the fulfilment of the whole set of functional assembly requirements, introducing eventual mechanical compensations due to manufacturing unavoidable errors.

The realization of a DTT Remote Handling Facility (REMHAT) at University of Naples Federico II (Naples, Italy) is necessary: 1) for the development and optimization of the Remote Handling (RH) system, to demonstrate the fulfilment of all the design requirements; 2) for testing and validation of the RH maintenance procedures; 3) for the qualification and acceptance testing of RH equipment and tooling. Considering the maturity level of RH facility components, the paper will deal with better definition of the assembly sequences, and all custom mechanical compensations. The work will allow not only to improve the design phase, but also to support the later on-site assembly phase and the dimensional inspections for implementing as built data in the assembly model. Taking advantage of the previous experience for the ITER tokamak gained in F4E, the study, simulation and testing activities will be applied to the mock-ups of Vessel and In Vessel components (Inner First Wall and Divertor) and their interfaces, considering their actual design, size constraints and RH compatibility requirements. The digital model will be managed setting up the available data on dimensional variation having previously defined nomenclature and priority rules for the relevant interfaces. Virtual simulation and alignment studies has been carried out using the available knowledge for satisfying and verifying a set of defined requirements.