PROGRESS OF ITER AND ITS IMPORTANCE FOR FUSION DEVELOPMENT

P. BARABASCHI[1] and THE ITER TEAM
[1]ITER Organization
St. Paul-lez-Durance, France
Email: Pietro.Barabaschi@iter.org

Abstract

This paper summarizes the progress of the ITER project and highlights its significance for fusion development, especially in light of recent global growth in public and private fusion initiatives.

Following a significant series of project reforms that included reorganization, adjustments to major contracts, renewal of trust with the regulator, and repair of key components, the ITER Project has performed with unprecedented schedule and cost efficiency over the past two years, progressing strongly under its new Baseline 2024. This baseline is a comprehensive and feasible plan for assembly, integrated commissioning and operation under a stepwise safety demonstration, developed to deliver the key objectives of ITER, indispensable for fusion and achievable only by ITER, as early as possible. Significant progress on the first-of-a-kind manufacture of fusion components and formation of their associated global supply chains has been led by the ITER Organization, the Domestic Agencies of the seven Members, and their national industries. These achievements have included repairs of some vacuum vessel bevel joints and thermal shield cooling pipes, completion of all superconducting magnets, completion of six of the nine vacuum vessel sectors, start of series production of divertor components and gyrotrons for the electron cyclotron heating, and more. Assembly and installation of vacuum vessel sector modules are progressing with a Schedule Performance Index (SPI) above 1, with two installed and two more expected in the coming months. Commissioning of plant systems, including the cooling water system, the pulsed power electrical switchyard, and the world's largest cryoplant, is also going well. A new magnet cold test facility is nearing completion and will start operation in winter 2025. In the Neutral Beam Test Facility, the SPIDER experiment and MITICA final installation are progressing as planned.

In Baseline 2024, the first operations phase is a scientifically meaningful research phase including deuterium-deuterium H-mode operation and achievement of full magnetic energy. Towards these operations, extensive integrated and advanced modelling and plasma predictions have been carried out. Using these studies and collaborating with other tokamak research facilities globally, the tungsten first wall concept has been assessed and adopted.

Reflecting on recent developments in fusion, this paper also considers ITER's mission and objectives in the framework of the remaining challenges still to be addressed for fusion to become a practical source of electrical power generation.

1. INTRODUCTION

The ITER Project is progressing strongly, reflecting the success of a series of significant project reforms followed by robust execution under a new cost and schedule baseline (Baseline 2024). The reforms included substantial reorganization, review and adjustment of major contracts, and the successful reestablishment of trust with the French nuclear safety regulator (ASNR). Repairs to key components—vacuum vessel sector bevel joints and thermal shield cooling water piping—have progressed according to plan.

Baseline 2024, submitted to the ITER Council in June 2024, is a comprehensive and feasible plan for tokamak assembly, integrated commissioning, and operation under a stepwise safety demonstration. The goal of the new baseline is to deliver the key objectives of ITER, indispensable for fusion and uniquely achievable by ITER, as early as possible; the baseline is structured to achieve ITER's missions of demonstrating extended burn with a fusion gain of $Q \geq 10$ and validating the availability and integration of technologies essential for fusion reactors. At its essence, the role of the ITER project is to move fusion development forward from science to industry. ITER is the first-of-a-kind (FOAK) industrial-scale deuterium-tritium (DT) fusion device, far exceeding the dimensions and parameters achieved in current devices [1]. Improvement measures based on lessons learnt, risk & opportunity assessments and stepwise progression must be efficiently adopted in manufacture, assembly, commissioning, operation, and licensing for ITER, with implications well beyond ITER. The ITER Council has signalled its overall approval of the Baseline 2024 schedule, as well as approval of the project cost profile—subject to approval from capitals—through the end of 2028.

Significantly, the ITER Project has been performing with unprecedented schedule and cost efficiency, executing to this new baseline. Throughout 2024 and 2025, the Project has maintained a schedule performance index of 1.0 with a cost performance index of 1.1, meaning that 100 per cent of what was planned is being achieved, remaining well within the budget. In 2024, the Project implemented 30 per cent more work than in any previous year, and 2025 is progressing with the same execution rate. This translates to the strongest performance ITER has ever delivered; it validates that the design of Baseline 2024 is grounded in realistic assumptions and reliable technical data, incorporating lessons learned; and it serves ITER's stakeholders well, both Member governments and the global fusion community.

2. ITER PROGRESS AND VALIDATION OF KEY FUSION TECHNOLOGY

Significant progress on FOAK manufacture of fusion components, with the corresponding formation of their global supply chains, have been led by the ITER Organization, the Domestic Agencies (DAs) of the seven Members, and their national industries.

2.1. Production of all superconducting magnets completed

The production of the ITER superconducting magnets (FIGs 1,2,3) has been successfully finalized in 2025 and lasted about 17 years (including conductor production). Most of the ITER Members (6 of 7) were involved in this gigantic effort: it represents, both from the technological point of view and in terms of procurement and industrial complexity, one of the biggest challenges that ITER has successfully overcome [1,3,4]. A global supply chain for mass production of ITER-grade superconducting magnets has thus been established.

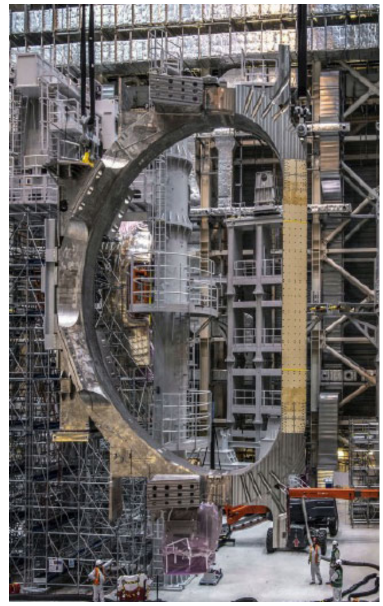

FIG. 1. One of the TF coils built in the EU and JA before being assembled on VV

ITER's superconducting magnet system consists of a central solenoid (CS), made of 6 modules (+1 spare); 18 toroidal field (TF) coils (+1 spare), 6 poloidal field (PF) coils, and 18 error field correction coils (CC). The D-shaped TF coils are about ~16.5m × 9m in size and weigh about 300 tonnes, operated with a constant 68 kA and 11.8 T peak field. The PF coils have diameters between 8m and 24m and weigh up to 400 tonnes; they are pulsed up to 55 kA with 6.4 T. The vertical stack of the CS assembly will be about 13m high and weigh about 1000 tonnes, supported from the bottom of the TF coils through its pre-load structure; the nominal CS current is 45 kA, achieving close to 13 T.

Two types of superconductors are used for ITER magnets: NbTi for the CC s and PF Coils and Nb_3Sn for the CS and TF Coils. The ramp-up of the production of the conductor necessary for the magnetic system was one of the first big ITER challenges. For the NbTi based magnets, about 300 tonnes of superconducting strands were produced between 2011 and 2014. The conductors were produced by the DAs of three Members (China, Europe, and Russia), involving two suppliers for the superconducting strands and 4 suppliers for cabling and jacketing.

FIG. 2. The fourth CS module being stacked on CS Assembly. All 6 modules are now delivered from the US

The situation for the Nb_3Sn conductors was more challenging, as the pre-ITER global production of Nb_3Sn strand was about 15t/year and ITER required several hundred tonnes. The strand production was distributed across nine suppliers involving six DAs (China, Europe, Japan, Korea, Russia, United States), delivering almost 700t of Nb_3Sn strand in about eight years with a peak production rate of 100t/year. Cabling and jacketing of the CS and TF conductors involved 13 suppliers and more than 30 sub-suppliers (including the procurement of steel and copper). In addition, a large number of institutes, laboratories and specialised companies were involved in the quality control and performance verification. In total 299 conductor lengths, equivalent to a total of 2015 km, were manufactured. The production of the magnets was also distributed among DAs: China produced 1 PF coil (procured by Europe in China) plus 18 CCs; Europe produced 10 TF coils and 4 additional PF coils; Japan produced 9 TF coils, Russia produced 1 PF coil; and the United States produced all of the 6 CS modules plus 1 spare.

FIG. 3.The Largest PF Coil, PF3, Delivered in May 2024

The logistics of delivering the conductor lengths produced in six Members to the different magnet suppliers distributed across three continents also was a big challenge. The FOAK and complex nature of the coils posed high technical risks, and large financial investments were required to develop the facilities and support production. An interesting aspect of this distributed and diversified supply chain is that different DAs had to use different procurement strategies, each tailored to local industrial practices, industrial market availability, and the risk appetite of the different stakeholders. In Europe, the 10 TF coils were procured using three suppliers, each composed of several companies, and each working in a different phase of the production chain. In Japan, the 9 TF coils were produced by two suppliers, each managing the whole production chain. For the PF coils, Europe, China and Russia used different approaches. In Europe, because some of the PF coils were too large (up to 24m in diameter) to be transported on public roads, a dedicated manufacturing facility was erected at the ITER site. For this production a cluster of six suppliers were used to develop the production tooling and cold testing equipment and to carry out engineering integration and manufacture of the different coils. In Russia and China, the PF coils were produced using a combination of local industries and academic institutes. Lastly, the CS coil was produced in the US using a single supplier managing the whole production cycle.

2.2 Series production of tungsten divertor underway

In the ITER tokamak, 54 divertor cassette assemblies (CAs) are to be installed to complete the toroidal ring of the full tungsten (W) divertor before the Start of Research Operations (SRO) phase [1]. The CA consists of outer vertical targets (OVTs), inner vertical targets (IVTs), domes, and cassette bodies (CBs). Three DAs and their suppliers are involved in the procurement of the CAs (Japan supplies the OVTs, Europe the IVTs and CBs, and Russia the domes). The main technical challenges for the ITER divertor involve developing a robust cooling structure to handle an intense heat flux as high as 10-20 MW/m^2, a level unprecedented in conventional engineering components, in a quasi-steady state. This also requires achieving high reliability in joining hundreds of thousands of tungsten tiles to the cooling structure and meeting stringent geometric tolerances, such as a surface contour tolerance of ±250 µm over a 1.5-meter component, to prevent the leading edges from melting. Through carefully coordinated R&D including high heat flux tests [5], prototypes of IVT, OVT, and dome have satisfied these requirements, and all divertor components are now in the series production phase (FIG.4).

Contracts for series production of OVTs, domes and IVTs have been placed by Japan, Russia, and Europe, respectively. The first OVT will be delivered in early 2026, the first dome in 2026, and the first IVT in 2027. Europe has signed the contract for the full scope of CB series production (54 and 4 spares), and the first CB is expected by the end of 2026. The manufactured components are delivered to the integration site in China where the CAs will be placed in their final configuration. This activity also includes the assembly of divertor-mounted diagnostics where applicable. The integration contract was awarded to a SWIP-led consortium. Receipt of components will be followed by assembly, factory acceptance tests, and delivery to the IO site. At ITER, the delivered CAs will be installed through three lower ports into the vacuum vessel (VV) by divertor cassette movers. One of the key operations is the custom machining of the divertor toroidal rail elements to ensure alignment requirements in relation to the tokamak assembly datum, accounting for as-built VV dimensions. The detailed processes, necessary resources and activities were examined in the recent divertor installation review.

FIG.4. Divertor cassette series production and prototypes of IVT, dome and OVT

2.3 New tungsten First Wall (FW) design and manufacture started

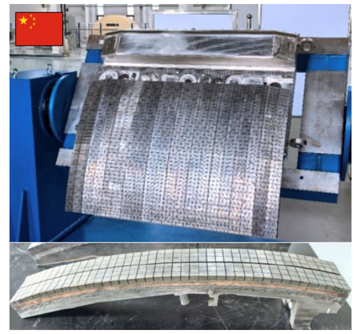

FIG.5. Semi-prototype of a First Wall panel as a part of their Tungsten armour qualification program.

In the Baseline 2024. the armour material of the FW has been changed from beryllium to tungsten and the new "Start of Research Operations (SRO)" phase starts with an inertially cooled FW [1]. This "temporary first wall" (TFW) [6] will be more forgiving to disruptions, resulting in a reduced risk of an in-vessel water leak that would hinder subsequent DT operations. The TFW will mimic the actively cooled FW in material, geometry and interfaces, with a shape and structure that is intentionally similar to its successor in later DT-1 and DT-2 operations. Since these TFW panels will experience different loading depending on their position in the machine, the TFW design includes a mixture of potential plasma-facing design types, including bulk tungsten blocks, tungsten heavy alloy, and tungsten-coated steel. Since tungsten coating on steel is a relatively novel technology for fusion, a robust coating qualification plan is underway: first by cyclic transient loading and high heat flux testing (via electron beam), and second by ITER-relevant loading of large areas in existing tokamak(s). A considerable design challenge for the TFW will be its tolerance to the anticipated high transient loads of the SRO phase, especially in the current quench phase during disruptions and runaway electron events. Modelling of these loads and their impact is underway to assist in the design development of the TFW that can enable successful completion of the ITER research plan for SRO [7, 8] while providing investment protection.

The change of FW material is accompanied by a desire to mitigate a long-standing concern of runaway electron damage causing water leaks in the final water-cooled tungsten wall for DT, for which a thicker armour layer is proposed in the impacted areas [6, 7]. While beneficial, this material change, together with the increased (local) thickness, required an adaptation in the design of each panel type to re-align all panels to a new position. Since more than half of all shield blocks, on which the FW panels are mounted, are fully manufactured, the re-alignment of panels is done by the addition or removal of thickness in the FW panels. The change to tungsten also has benefits to the design, with a considerably higher melting temperature relative to beryllium, and a more robust bond between the armour and the copper alloy heat sink. These changes may permit design simplifications and demand a re-evaluation of the tolerance requirements of all panels, which were all based on a beryllium armour. While several full-scale prototypes of the beryllium FW panels have been completed, work is now underway for re-production of similar tungsten-armoured representative mock-ups to demonstrate the performance of newly developed technology. This qualification work is proceeding in parallel with ongoing manufacturing of stainless steel and copper alloy parts of the FW, making the finalization of the tungsten FW a delicate balance to design, qualify, and manufacture simultaneously.

2.4 Electron cyclotron system supply chains established, and installation started

In the Baseline 2024, the electron cyclotron (EC) System has been upgraded to a total of 40 MW delivered to the plasma in SRO (48 gyrotrons) and 60 MW in DT-1 (72 gyrotrons). While the EC launchers and ex-vessel waveguides are still in the design phase, the supply chain is fully established for the rest of the EC subsystems.

Manufacture of the high-power gyrotron system in the frequency range of 170 GHz has been established [9, 10]. Following successful factory acceptance tests, the first gyrotrons from Japan and Russia have already been delivered and are being installed in their final locations. Commissioning of the first gyrotron from Japan is starting. The original scope of 20 MW will be completed by six gyrotrons from Europe (Final Design Review (FDR) held in July 2025) and two gyrotrons from India (FDR in 2026). For the additional scope of 20MW in SRO and 40 MW in DT-1, two task agreements with Japan and Russia are being used to procure an additional 40 (+8) gyrotrons. These task agreements provide the option of delivering higher power gyrotrons, up to 1.5 MW each. The increased power is a powerful tool allowing increased system availability; power loss due to faulty components can be compensated by increasing the power in working gyrotrons. It is of note that transmission lines (TL) and launcher mirrors are also compatible with increased power, albeit for shorter pulses. All TL components have been prototyped and tested by the US, and the supplier has been selected. The high voltage power supplies (HVPS) from Europe and Japan have been procured and installed in their final location. Both power supplies are being commissioned and will be used for the gyrotron commissioning starting in 2025. The HVPS from India will be installed in 2028. In response to the need for 24 additional HVPS sets in the Baseline 2024, the first HVPS for the new scope will be installed in 2028.

2.5 Disruption Mitigation System design matured

ITER's Disruption Mitigation System (DMS) is based on Shattered Pellet Injection (SPI) technology and consists of 27 injectors distributed over 6 ports, capable of injecting large pellets (28.5 mm in diameter) of H, Ne and mixes as cryogenic fragments. After passing FDR in 2024, this system is now being brought to the maturity required for manufacturing [11]. As part of this effort, the formation of ITER-sized pellets has been demonstrated in the test bench at the Low Temperature Systems Department (DBST) at CEA with supercritical helium coolant at ITER-like flow rates. For DMS, effective retention of propellant gas is essential to avoid plasma edge cooling which could trigger thermal quench prior the fragment arrival. To address this issue, a solution based on a muzzle brake to ensure a straight flight path, validated at the DMS Support Laboratory in Budapest, is being developed. The geometry of the shattering chamber and the strains on its materials by pellet impact have been assessed and optimised to obtain the desired lifetime and fragment size distribution (FSD) to maximise neon and hydrogen assimilation for achieving the desired mitigation.

A model based on the discrete element method has been developed to predict the FSD and validated with experiments showing that previous pellet fragmentation assumptions are inaccurate; this is important to assess the effects on SPI in the plasma. The viability and compatibility of DMS with all the required load mitigation objectives (thermal loads, electromagnetic loads and runaway electron (RE) avoidance and mitigation) have been assessed in Member tokamaks such as ASDEX-U, DIII-D, EAST, HL-3, JET, J-Text, KSTAR and TCV. These experiments have provided key inputs for the DMS design (e.g., jitter requirements) but also identified possible limitations. These include upper pressure limits for benign RE termination and physics processes leading to impaired penetration of the fragments. These limits can be alleviated by using neon-doped hydrogen pellets, and extrapolations of this effect are being carried out with the code PELOTON to provide accurate input to disruption mitigation codes such as JOREK, INDEX and DREAM. The resilience of the tungsten wall to disruption loads is being used to advantage to make all DMS objectives realizable by decreasing mitigation targets for thermal loads, which facilitates RE avoidance, since the latter is negatively impacted by the amount of neon used for thermal load mitigation.

3. TOKAMAK ASSEMBLY

ITER's tokamak assembly is a FOAK demonstration of the technology for assembling a full-scale fusion reactor.

3.1 Vacuum vessel and thermal shield successful recovery from non-conformances

Repair of the vacuum vessel (VV) sectors, which had geometric non-conformities in the field bevel joints, is well underway, based on successful R&D of an optimized combination of build-up and machining [12,13]. VV sector welding bevels were reconstructed using manual tungsten inert gas (TIG) addition of metal and partially mechanized TIG; then machining was performed with newly developed portable milling machines to achieve the required tolerances for sector welding and NDE. As of mid-2025, four VV sectors have been fully repaired. Bevel repair of Sectors 1 and 4 is in process and scheduled for completion in Q4 of 2025.

During the receiving inspection of the thermal shields, three leaks were detected on the vacuum vessel thermal shield (VVTS) equatorial port shroud. Root cause analysis confirmed that the failure was caused by stress corrosion cracking (SCC) in the cooling pipes. Removal of the propagated SCC was not practically feasible, so full pipe replacement has been implemented [12]. For this purpose, 316L grade pipe and ER317L Mn Mod filler material was employed to improve corrosion resistance. In the vicinity of the welded regions of the pipes, a 2mm thick panel layer was removed to ensure the elimination of residual chloride prior to new pipe rewelding. In other regions, the coating layer was removed and polished to $Ra < 0.1$ µm to ensure adequate emissivity. Following the removal of cooling pipes and coating layers, the panels were examined for SCC by means of fluorescent penetrant testing. The success of the repair was confirmed by leak testing under equivalent pressure boundary conditions of operation in a vacuum chamber, and geometrical conformity was confirmed by pre-assembly test. As of mid-2025, five VVTSs have been successfully repaired. Completion of remaining sectors is scheduled for Q1 2026.

3.2 Vacuum vessel sector module (VVSM) assembly and positioning in pit progressing ahead of schedule

After repair of these major components, the assembly of the core machine has restarted in September 2024 with sector module sub-assembly (SMSA) activities [14]. A sector module (SM) consists of the VVTS and two TF coils surrounding a VV sector. The repair time was used to reverify assembly documentation and incorporate lessons learnt from the previous assembly attempts.

SMSA activities start with installations of diagnostic systems on the outer surface of each VV sector before it comes into the assembly hall. The assembly hall is equipped with two purpose-built sector sub-assembly tools (SSATs), which enable parallel production lines for module preparation. Nine of these completed sector modules will form the complete tokamak torus. VV Sectors 6 and 7 have been assembled in the SSATs and transferred to the tokamak pit in April and June 2025, respectively, each ahead of schedule (FIG. 6). Sectors 5 and 8 are currently installed in the SSATs. The third VVSM is expected to be installed in the pit by the end of 2025, and the fourth soon after.

In the tokamak pit the module positions will be fine-tuned and the overall VV prepared for final welding once all VV sectors are settled together. In addition, the inter-coil structure between adjacent modules is a major part of pit activities. Precise measurements are necessary to ensure correct customization of required pieces to perform the connection. A new tokamak in-pit assembly approach has also been adopted, based on the simultaneous welding of all nine VV sectors after positioning in the pit.

FIG.6. Sector modules 6 & 7 positioned in the pit

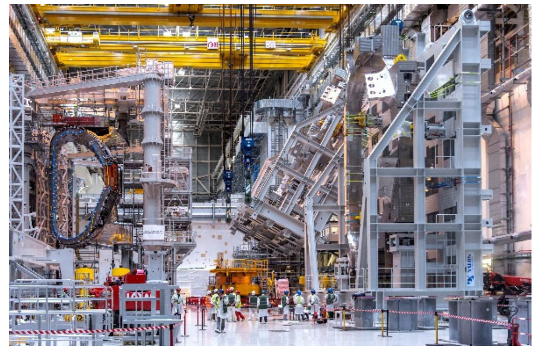

FIG.7. Two upending tools enabling parallel works

It is noteworthy that, with great effort from IO and associated contractors, the overall time required for assembling and installing a VVSM has been reduced significantly (from 18 months in 2021 to 7.5 months for SM 7 and 6.5 months for SM 6), optimizing processes to accelerate sector assembly and save time for late installations. As an example, in 2024 a second "upending" tool was ordered for the assembly hall, to enable parallel preparation of components. Two TF coils can now be pre-positioned for lifting to the SSAT with the same rigging configuration (FIG. 7). After SMSA works, many more components and systems will need to be positioned near the major components, such as other coil systems (e.g., PF coils, CS), cryostat thermal shield, cooling water and cryogenic pipes, and instrumentation. [1]

3.3 In-vessel assembly; a reliable and efficient plan established

The in-vessel assembly project (IVAP) is one of the most critical undertakings within ITER's machine assembly program, as it focuses on integrating the highly complex and technically demanding systems that form the core of the tokamak. Its scope is broad, starting with many small and fragile components encompassing the installation of diagnostic and fuelling systems (18 km of mineral insulated cables, 100,000 clips, 21,000 bosses, 9,000 clamps, 1000 sensors), and moving to the heavier heating and diagnostic port plugs (36 plugs in total), in-vessel coils (27 edge localized mode (ELM) control coils, 2 sets of vertical stability (VS) control coils), blanket manifolds, 440 shield blocks, FW panels, and the 54 divertor cassettes. These activities are indispensable to ITER's transition from construction to successful commissioning and operation.

Executing such an ambitious scope comes with significant challenges, including the tight tolerances to be satisfied under the final as-built dimensions of the vacuum vessel. IVAP develops and qualifies unique assembly and welding processes including deployment of custom-built tools [15]. In order to address these challenges, IVAP has adopted a phased and structured strategy. This includes robust contracting methodologies, strong collaboration across ITER programs and industrial partners, and a progressive build-up of work to de-risk the most critical activities. Early phases focus on engineering work packages, mock-ups, and qualification of processes, providing valuable input for optimization of sequencing and minimizing technical and commercial risks. As the project ramps up to full-scale installation, the lessons learned from qualification and mock-up activities will be applied to streamline assembly on site.

4. PLANT SYSTEMS: PREPARATION AND OPERATION

4.1 Cryoplant commissioning successfully underway

The ITER cryogenic system is a critical infrastructure enabling the operation of superconducting magnets, vacuum systems, and diagnostics by maintaining ultra-low temperatures. Utilizing supercritical helium at ~4 K and helium-cooled thermal shielding at 80 K, the cryoplant, one of the largest of its kind, includes multiple helium and nitrogen refrigerators, extensive cryoline networks, and a 25-tonne helium inventory. It delivers cooling at three temperature levels —4 K, 50 K, and 80 K— through liquid helium baths and forced-flow systems. Designed to handle dynamic heat loads during pulsed fusion operation, the system supports a wide range of plasma scenarios and is engineered for gradual cool-down and magnet preparation. With a cooling capacity of 75 kW at 4.5 K and 1300 kW at 80 K, the ITER cryogenic system sets a benchmark for large-scale fusion cryogenics.

Over the past two years, the ITER cryogenic system has reached major milestones in installation, commissioning, and operational readiness. All core components—including the three liquid helium plants, the two nitrogen refrigeration plants, gas and liquid storage tanks, helium recovery and distribution systems, and cryogenic lines—have been successfully installed and tested. The cryogenic plant is now fully prepared to support superconducting magnet cold tests, a critical step toward validation of the key components. Significant progress has also been made in the tokamak construction area, with full installation of cryolines across the cryo-bridge, delivery of auxiliary cold boxes (ACBs), and steps towards the interface connections with clients. Electrical and control systems have passed factory testing and are ready for integration.

A notable milestone has been achieved by providing cryogens for the first time to a client, specifically the test of the first cryopump connected to the cryoplant infrastructure. The test has shown performances above expectations for pumping hydrogen-like and helium isotopes for this first of a kind component. With inlet pressure of 10^{-2} Pa, the test showed a helium pumping speed of 200 m^3/s. The pumping capacity attained, as expected, 24 moles of helium over 30 minutes.

The cryogenic system will still face a few challenges, including managing dynamic heat loads and maintaining precise temperature control of the superconducting magnets in a wide range of plasma scenarios, handling thermal cycling during cryopump regeneration. Ensuring safety, reliability, and flexibility in a nuclear environment also remains a top priority.

The ITER cryogenic system sets a global benchmark for industrial-scale cryogenics in fusion energy. Its successful implementation provides valuable engineering insights and operational experience that will benefit future fusion reactors—both public and private—by demonstrating scalable cryogenic solutions, fostering international collaboration, and developing a skilled workforce for the fusion industry.

5.2 Magnet Cold Test Facility starting operation in 2025 winter

In June 2023, the ITER project baselined the construction of an onsite Magnet Cold Test Facility (MCTF) to test several (~5) selected TF coils and the PF1 coil. These tests will be carried out within a range that does not affect the tokamak assembly schedule. Beyond the testing itself, the test bench integration and operation present a valuable opportunity to gain experience ahead of ITER's commissioning and operation phases. To optimize the relevance of this experience, the facility has been designed and assembled from ITER components wherever feasible. Similarly, the operating procedures are intentionally as representative as possible of the ones expected for tokamak operations.

The assembly of the facility is near completion after 2.5 years, including design and manufacturing, with all dedicated equipment delivered onsite:

- Several components were specifically procured for this facility: a large-scale cryostat (21.5m x 10.5m x 6m, 300 tonnes); vacuum system; customized lifting tools for TF and PF1 coils; interconnection valve box and approximatively 150m of cryolines to connect to the ITER cryoplant; high current power convertor rated at 70 kA; warm busbars; fast discharge unit (FDU) rated at 70 kA / 4 kV; dump resistor with an energy absorption of 810 MJ; and 4 K busbars up to the coil.
- Several systems are ITER tokamak components temporarily used for the facility: one coil terminal box (CTB) with the 68kA current distribution and the cryogenic control valves, I&C cubicles (including control, instrumentation, quench protection, interlock, data acquisition and archiving).

All equipment except the cryostat and its vacuum system and the power systems have been installed. The first phase of facility commissioning consists of cooling down a superconducting jumper connected to the CTB cold terminal end. In this configuration the CTB vacuum chamber is closed with an end cap. The next steps will be the cryostat installation and the connection of the vacuum system and the power systems. During this second commissioning phase, the cryostat will be evacuated and the superconducting jumper energized and quenched. Once all commissioning phases are successfully completed, by mid-December 2025, the facility will be ready for operation. It is then foreseen to test selected TF coils in 2026 before they are installed in the tokamak, and then to test the spare coil (TF19) and PF1 in 2027/2028.

The test sequence for the tokamak TF coils proceeds as follows, starting from the cryostat closure: evacuation and leak test; cool down to 4 K (2 weeks); current tests up to 34 kA (three weeks); hydraulic performance verification; quench detection calibration; progressive current ramps followed by fast discharge at +/-90 V; steady operation at 34 kA; and then warming up (2 weeks). It is important to note that the fast discharge voltage has been intentionally limited below the minimum Paschen voltage. This test parameter choice simplifies the insulation process during the coil assembly in the facility, preventing the risk of electrical breakdown. The spare TF coil, TF19, will be energized up to the nominal TF current of 68 kA. This coil will be re-enforced with a mechanical structure to prevent the deformation of the straight leg of the D shape coil during energization. This deformation induces a higher compressive strain on the conductors than when tested in final configuration in the tokamak, due to the standalone TF coil in the facility. The superconducting performance is planned to be checked with a temperature margin test. It will be conducted by heating the central double pancake up to the quench. PF1 will follow a similar test plan up to the nominal current of 48 kA.

4.3 Neutral beam test facility: progress on SPIDER experiment and MITICA final installation

The ITER neutral beam (NB) system provides 33 MW of heating power with high-energy beams of deuterium or hydrogen in DT-1. The neutral beam test facility (NBTF) was established in Padova with the aim to demonstrate the engineering design and the performance of the core components of the system [16]. The facility includes two main test beds: SPIDER, a full-size prototype of the ITER NB ion source, and MITICA, a full-scale prototype of the whole injector, including a 1 MV power supply system. During the last year SPIDER completed a significant experimental campaign; for the first time, it operated with 1/4 of the available beamlets simultaneously, achieving a beam power of around 700 kW. These results confirmed that the source is following the expected scaling of current density versus input power delivered to the source, previously achieved in smaller scale devices at IPP-Garching. This result suggests that the target beam current for hydrogen should be achievable when the full input power of 800 kW will be delivered to the source. The campaign also showed a tenfold increase in beam duration compared to the previous campaign. Issues identified, such as the need for enhanced vacuum pumping and more efficient plasma generation, are being addressed with the integration of new non-evaporable getter (NEG) pumps and solid-state amplifiers, which are being installed during the ongoing shutdown.

Concerning MITICA, a critical high voltage insulation test in vacuum was completed in early 2025, giving high credibility to the capability of the system to withstand insulation voltage >910 kV (above the target value for hydrogen operation of 870 kV). The insulation at the maximum voltage was held for over an hour, and the conditioning time was only a few days. Meanwhile, new protection systems have been designed, manufactured and installed to prevent overvoltage, to avoid the breakdowns that occurred in 2021. In the last year, a new NBTF research plan was developed that is fully integrated with the new ITER Research Plan [16]. This includes milestones aimed at demonstrating the soundness of the configuration for NB heating operation by 2032. SPIDER and MITICA will alternate between operational and maintenance phases to incorporate lessons learned.

5. IMPORTANCE OF THE NEW BASELINE FOR FUSION AT ITER

5.1 Baseline 2024: a realistic and feasible plan

In 2024, the ITER Council endorsed the overall approach of the new ITER Baseline 2024 (FIG. 8) with the main target dates of cryostat closure in 2033, start of DD H-mode operation in 2035, full plasma current I_p=15 MA and toroidal field of 5.3 T operation in 2036, start of the DT operation phase in 2039 and Q ≥ 10 in 2044. This baseline is a comprehensive and feasible plan for assembly, integrated commissioning and operation, developed to keep the already agreed final project goals and to deliver the key objectives of ITER as early as possible, with a stepwise approach to safety demonstration and licensing. The first operation phase, named Start of Research Operation (SRO), is a scientifically meaningful research phase with a sufficiently high heating power of 40MW ECH and 10MW ICH, a tungsten water cooled divertor, and in-vessel coils (ELM and VS control) to start DD H-mode operation and to demonstrate integrated fusion system performance at the nominal magnetic energy with I_p = 15MA/B_t = 5.3 T. The SRO phase will largely demonstrate ITER's mission of an integrated tokamak system operating at industrial scale. The Baseline 2024 was developed based on the most up-to-date knowledge with wide input from the Members' scientific communities. As noted earlier, one consequence is the change of the FW armour material from beryllium to tungsten [6, 7]; beryllium armour bridging phenomena during unmitigated disruptions and the resulting high electromagnetic forces leading to single event failure are now better understood. Other reasons include the toxicity and the magnitude of erosion of beryllium and the associated tritium retention, and the fact that future fusion reactors (DEMOs) already use tungsten as the standard for their FW. During intensive evaluations on the adverse effects of the high-Z tungsten first wall on plasma performance, conducted with the involvement of Member experts globally as well as the ITPA, no show-stoppers for ITER have been found to date on the basis of experimental results and validated model predictions. Of course, mitigation for the high tungsten contamination risk is necessary and included in the new baseline by introduction

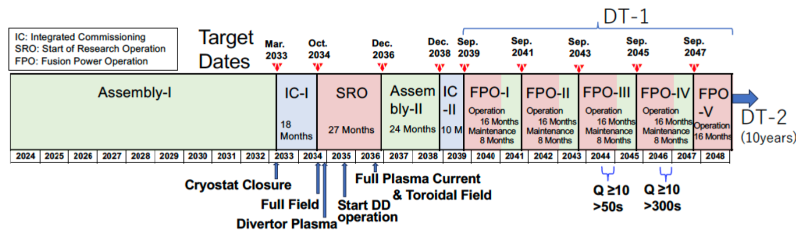

FIG.8. The overall approach of the ITER New Baseline 2024 with the main target dates

of new systems (boronization) and modification to existing systems (increase in heating power and ECH vs ICH power sharing).

5.2 Licensing activities progressing

Since the previous FEC in 2023, extensive interactions have taken place between the ITER Organization (IO) and the French regulator, ASNR, to re-establish constructive relations and adapt the regulatory framework to address the unique characteristics of the ITER project. Given the Project's complexity and inherent uncertainties, such as dust and activated corrosion products generation, tritium retention, and equipment reliability and maintenance, a stepwise demonstration of nuclear safety for ITER has been identified as the most feasible approach. In developing the Baseline 2024, IO proposed a phased authorization process to ASNR, aligned with this progressive safety demonstration. This approach envisions that the ITER's licensing pathway will be structured with defined hold points and corresponding safety report updates. At each stage, the detail and scope of the nuclear safety case will match the project's maturity and incorporate knowledge gained from prior experimental phases.

Discussions with ASNR are also ongoing to address ITER's characteristics as a tokamak, noting that significant differences, such as lower decay heat and dominant electromagnetic loads, distinguish it from conventional fission light-water reactors.

The key licensing documents submitted by IO to the regulator since 2023 include:

- A dossier covering radiation protection and tokamak support structure aspects. Another dossier focused on the vacuum vessel itself is scheduled for submission in 2026. These 2 dossiers are necessary for authorizing the start of tokamak assembly operations (VV sector welding);
- A request to update technical prescriptions in alignment with the new Baseline 2024;
- A proposal to exempt the ITER VV from the European Pressure Equipment Directive, recommending instead regulation under a more appropriate framework that recognizes electromagnetic loads as the primary design criterion;
- Documentation outlining the evolving strategy for nuclear maintenance and radioactive waste management, reflecting optimization and compliance with the Baseline 2024.

In 2025, IO made presentations at several IAEA meetings, highlighting the necessity of a graded regulatory approach for fusion, and discussing the practical application of regulations, codes, and engineering standards. Through these efforts, ITER supports the International Tokamak Engineering Criteria Activity in developing suitable regulatory frameworks and engineering standards for future fusion facilities, leveraging lessons learned from ITER's licensing experience.

5.3 Change of FW material to tungsten assessed

The new ITER baseline includes the change of FW material from beryllium to tungsten, following evaluations of its implications, with emphasis on robustness for successful implementation of the ITER Research Plan (IRP) [7, 17, 18, 19] and facilitating licensing aspects. However, this change in turn introduces additional risks related to the quality of vacuum in the ITER tokamak (oxygen gettering) and plasma contamination by tungsten. The latter is critical since concentrations of tungsten tolerable in ITER plasmas are typically ~3 orders of magnitude lower than for beryllium, and the risk that intolerable tungsten concentrations materialize in ITER needs to be minimized to ensure $Q \geq 10$ operation. Associated scenario modelling and dedicated experiments have been carried out at ITER Member tokamaks to evaluate these risks and define mitigation strategies [7].

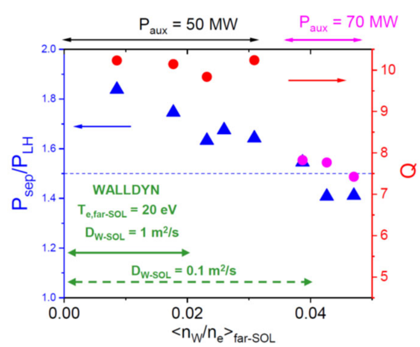

FIG.9. JINTRAC modelled Q and margin of edge power flow above the L-H transition vs. average tungsten density in the far-SOL compared to predictions from WallDYN for 15 MA/5.3 T DT plasmas with additional power heating levels of 50 MW ($Q \geq 10$) and 70 MW for high tungsten edge densities.

Concerning vacuum conditions, a new boronization system (diborane introduction and additional glow discharge cleaning electrodes (GDC)) has been introduced with a phased approached from SRO to DT-1 regarding the number of electrodes. To assess the lifetime of boron layers and to define the arrangement of electrodes, modelling and experimental studies have been carried out that support the chosen configuration and predict a maximum frequency of boronization of two weeks, coincident with the short-term maintenance periods in the IRP. In addition, schemes have been defined to remove the fuel retained in these layers by dedicated plasma operation and ion cyclotron wall conditioning, taking into account JET DT operational experience [20].

The risk of tungsten contamination by increased wall sources and radiative collapse of ITER plasmas has been evaluated through a range of modelling and analysis as well as via new experiments at Member tokamaks. These evaluations consider all phases of the ITER scenarios in the IRP (limiter, L-modes, DD H-modes in SRO and DT H-modes in DT-1) as well as heating and current drive (H&CD) schemes. These studies show that viable scenarios can be produced with a tungsten wall in ITER, especially for the stationary phases of SRO H-modes and $Q \geq 10$ operation (FIG. 9), avoiding uncontrolled tungsten central accumulation. For the lower power phases (limiter, low power L-mode) the operational range is narrower and requires optimization of plasma parameters including H&CD to avoid the collapse of plasma temperature by tungsten peaking [18]. Finally, the new baseline and IRP take advantage of the high resilience of tungsten to transient loads for both the SRO and DT-1/DT-2 phases of the IRP. This includes the use of inertially cooled tungsten PFCs in SRO to facilitate commissioning of control and protection systems with minimized risks (i.e., avoiding the melting of FW and PFCs by disruptions for, at a minimum, $I_p \leq 10$ MA and leaks from FW water-cooled components in this phase), as well as for DT-1 with water-cooled components. For the latter, a thicker layer of tungsten on the PFCs has been included in the design to increase robustness and decrease the risk of leaks following the impact of runaway electrons during disruptions [6, 7, 21].

5.4 Integrated Commissioning Plan

The first ITER integrated commissioning phase (IC-I) has a target duration of 18 months (including 6 months contingency) and will focus on the integrated commissioning of the tokamak components and systems to demonstrate that their performance can support the foreseen experimental programme in SRO. In this phase it will be demonstrated that suitable vacuum conditions for plasma operation can be achieved and that the superconducting magnets meet the requirements to support plasma scenarios up to 15 MA/5.3 T, two of the main outcomes of IC-I. This phase starts with the pump-down of the cryostat to check leak tightness of all components inside the cryostat, such as magnets and thermal shields, the cryostat, and the VV. This is an essential step in the demonstration that the core ITER tokamak and ancillaries meet the requirements necessary for the execution of the IRP. The high-level objectives for the IC-I phase will be achieved sequentially. These comprise wall conditioning (baking and GDC), cooldown of the superconducting coils, commissioning of all systems required for plasma operation such as ECH, ICH, in-vessel coils, DMS, diagnostics, and fuelling systems, and energisation of all the superconducting coils (by far the most demanding activity during this phase). Included also are the commissioning of the plasma control system (PCS), central safety system (CSS), central interlock system (CIS), radiological and environmental monitoring system (REMS), and the diagnostic systems foreseen for SRO. The IC-I phase will be declared successful once its targets are achieved and will be followed by the demonstration of first plasma in SRO.

The IC-II phase (pre-DT-1) will start with cryostat and vacuum vessel closure and start of pump-down and has a target duration of 10 months (including 3 months contingency). It will focus on the integrated commissioning of the available tokamak components and systems, the vast majority of which will be very close to their final DT-2 configuration at this stage, to demonstrate that their performance can support the foreseen experimental programme in DT-1 [8, 19] and eventually demonstrate the Project Specification. Of particular importance are the demonstration of the $Q \geq 10$ fusion power goal and of tritium breeding with the test blanket modules (TBMs) [8, 19]. Besides recommissioning of the systems already available for IC-I/SRO, IC-II will include the integrated commissioning of the key systems and components required for the achievement of the DT-1 goals, such as the fuel cycle for DT operation, TBMs, neutral beam injection (NBI), diagnostics required for DT operation, etc.

In the IC-III phase (pre-DT-2), all installed systems will be re-commissioned as well as any new system that will need to be installed or upgraded for the DT-2 phase (e.g., the 3rd NBI injector).

5.5 ITER Research Plan (IRP) for SRO and DT-1 & 2

(a) Start of Research Operation (SRO) will focus on the demonstration of 15 MA operation in L-mode, commissioning of all required systems, including disruption mitigation, and the demonstration of H-mode plasma operation in deuterium at 2.65 T with a low neutron fluence compatible with post-SRO assembly activities. The overall objective is to develop the operational basis for plasma scenarios for DT-1 and to commission with plasma the key systems required to support them (e.g., Plasma Control System (PCS), Advanced Protection System (APS), Disruption Mitigation System (DMS), etc.). This phase starts with the demonstration of the first tokamak plasma and proceeds to L-mode operation in hydrogen (H), first up to 7.5 MA/2.65 T and then at 7.5 MA/5.3 T with focus on disruption loads and mitigation. This is followed by H-mode scenario operation in DD plasmas up to 7.5 MA at 2.65 T together with an assessment of 7.5 MA/5.3 T operation in DD. This requires commissioning of the available H&CD systems (ECH and ICH with H and ^{3}He minority) up to their nominal coupled power levels for durations of up to 50 s. This DD phase is followed by operation in hydrogen, thus providing an assessment of deuterium retention, up to 15 MA/5.3 T and concludes with a set of experiments to quantify erosion of boron layers and dust formation for safety studies. Of particular importance is the inertially cooled SRO tungsten wall which minimizes risks associated with the development of disruption mitigation severely impacting operation.

(b) First deuterium-tritium phase (DT-1) will demonstrate robust operation in H-mode plasmas in deuterium-tritium (DT) to $Q \geq 10$ and for burn durations ≥ 300 s and tritium breeding within an accumulated fluence of 3.5 10^{25} neutrons (~1% of ITER's lifetime total), which is ITER's first fusion power goal. To meet it, the IRP addresses, gradually and robustly through five campaigns (FPO-1/5), key scientific and technical issues including: self-heating of DT plasmas by alpha particles, operation with efficient tritium management, tritium breeding by performing the TBM Research Program, validation of assumptions in the nuclear safety licence in DT-1 and provision of the basis to define license details for DT-2 phase. Specifically: (i) FPO-1 concentrates on recommissioning of systems in their final configuration with hydrogen plasmas, de-risking disruption mitigation associated with DT plasmas and operation with high power plasmas ($P_{input} \sim 100$ MW) in L-mode (up to 15 MA/5.3 T) and H-mode (7.5 MA/2.65 T) to demonstrate power handling capabilities of PFCs; (ii) FPO-2 starts with the interleaved development of DD and DT H-mode plasmas at 5.3 T, completing commissioning of all systems required for DT operation, with the target to demonstrate $P_{fusion} \geq 100$ MW with $I_p \leq 10$ MA for $t_{burn} \geq 50$ s; (iii) FPO-3 builds upon the FPO-2 results and targets the demonstration of $Q \geq 10$ $P_{fusion} = 500$ MW, with all plasma parameters (including helium exhaust but not the current profile) reaching stationary conditions, which is expected to require $t_{burn} \geq 50$ s; (iv) FPO-4 focuses on the extension of $Q \geq 10$ $P_{fusion} = 500$ MW to $t_{burn} \geq 300$ s including its repeatability; and (v) FPO-5 demonstrates the reproducibility of the $Q \geq 10$ $P_{fusion} = 500$ MW with $t_{burn} \geq 300$ s scenario and high-duty operation (30 m repetition time) with scenarios producing $P_{fusion} \geq 250$ MW and $t_{burn} \geq 300$ s. The latter demonstration is technically important in view of more intense operation anticipated in DT-2 and for demonstration of tritium breeding with TBMs.

(c) Second deuterium-tritium phase (DT-2), with all systems in their final configuration.

The DT-2 objectives are:

- To demonstrate all of the ITER Project's fusion power production goals ($P_{fusion} = 500$ MW with $Q \geq 10$ $t_{burn} \geq$ 300-500 s, in high-duty operation, and long pulse and non-inductive steady-state scenarios with $Q \geq 5$ and burn lengths of 1000 s and 3000 s, respectively).
- To support the ITER Members' fusion reactor programmes including scenario development issues, design basis and operational issues and their TBM programmes, up to neutron fluences of, at least, 0.3 MWyear/m^2.

5.6 Development of Integrated Modelling producing reliable predictions

The Integrated Modelling and Analysis Suite (IMAS) provides standard tools and applications to support the integrated modelling (IM) of fusion plasmas and has been developed in close collaboration with the Members' fusion communities [22-25]. This development has been guided by the need to address plasma scenario issues and to optimize the IRP towards the achievement of ITER's goals. IMAS applications have been applied intensively to develop the new baseline research plan including both scenario and machine configuration assessments. Regarding scenarios, the IM suites JINTRAC, ASTRA and DINA have been applied to model plasma scenarios as well as stationary plasma conditions for all phases of the IRP. These simulations have been key to defining the steps in scenario development for the IRP as well as evaluating other important aspects such as the neutron fluence consumption for SRO and DT-1. As an example, JINTRAC modelling of limiter ramp-up on tungsten surfaces has shown the key role of ECH in preventing tungsten accumulation in the plasma core during this phase of ITER discharges. Using tungsten sources derived from PWI models, JINTRAC has shown that the risks to the $Q \geq 10$ objective with a tungsten FW is low to moderate (FIG. 9). IM has also been essential to assessing the application of H&CD schemes through the use of the IMAS H&CD workflow now also integrated within the JINTRAC suite. Dedicated modelling studies have been performed to assess ICH coupling and near-field sheath effects to quantify the effect of ICH on tungsten sputtering, the revision of ECH operational conditions, and the estimate of NBI shine-through losses for the new IRP.

IMAS continues to be extended to deliver new capabilities and provide high-fidelity plasma simulator (HFPS) capabilities including the simultaneous core-edge-SOL transport and source modelling capabilities of JINTRAC with the free-boundary plasma evolution code DINA [23]. HFPS modelling allows full scenario simulations including all plasma aspects integrated with magnetic and kinetic controls, as well as plasma stability analysis. Besides applications to ITER, such capabilities are also applicable to the Members' tokamaks, enabling validation of the implementation and support to the research community. Looking to the future, IMAS applications are evolving towards adopting more modern code architectures and utilising improved software engineering practices [24, 25]. Collaboration with the fusion community through the sharing of state-of-the-art models optimised for high physics fidelity or fast plasma simulations is key to delivering the needed IMAS capabilities. To facilitate this objective the ITER Organization has launched an open-source software initiative presently under implementation.

6. A PERSPECTIVE ON FUSION RESEARCH AND THE IMPORTANCE OF ITER

Nuclear fusion has long been regarded as the ultimate solution to humanity's energy needs—a technology that could provide abundant power for millennia. The remarkable surge of private sector investment in fusion development, fuelled in part by global concerns over climate change, has led many in the fusion community to publicly suggest that commercialization is only a few years away. While this optimism boosts morale and attracts investment, it risks overlooking major technical and financial obstacles and may foster unrealistic public expectations. Such a gap between perception and reality could ultimately hinder progress toward reducing fossil fuel dependency. As Richard Feynman aptly noted: "For a successful technology, reality must take precedence over public relations, for nature cannot be fooled."

Cost remains one of the most significant barriers to fusion's progress. The ITER project, designed as a stepping stone toward commercial fusion reactors, has undeniably disappointed by demonstrating that the costs associated with building and operating such facilities can be nearly an order of magnitude higher than what is considered desirable for widespread adoption. It is important to recognize that this expense is not purely technical; organizational complexity and political constraints, including the unprecedented international partnership established to deliver the project, have contributed substantially to ITER's budget overruns. These issues are not inevitable, but future fusion projects must clearly prioritize cost reduction from the outset, embedding efficiency into every stage of design and construction. This means involving industry partners early in the process to ensure that manufacturing realities are reflected in design choices, avoiding over-specification that leads to unnecessary expenses, and maintaining practical loading conditions for mechanical and thermal stresses. These latter two elements, in particular, are not necessarily mitigated in more compact designs. Simplifying system interfaces and allowing adequate space for assembly and maintenance are also crucial steps; smaller is not always cheaper, and tightly packed subsystems can lead to cascading design changes that drive up costs. The lessons learned from ITER's manufacturing experience should be systematically consolidated and applied to future designs, both at the component level and in the overall system architecture. The ITER project is fundamentally committed to sharing this return of experience for the benefit of fusion energy development.

Managing the intense power flux and developing suitable materials for the first wall (FW) and all plasma-facing components (PFC) remains another critical challenge. Despite decades of experience on smaller research devices, and a well-established physics understanding of power exhaust, magnetic confinement fusion at the reactor scale must still contend with extremely high power fluxes, both stationary and transient, on PFCs. The problem is common to both compact (e.g., high field or tight aspect ratio) and conventional DEMO devices. In particular, the ratio between peak and average power flux remains problematic, especially when the resulting flux is combined with the need to breed sufficient tritium, employ plasma-compatible materials, control edge instabilities in high confinement plasmas, avoid/mitigate disruptions and ensure that PFCs can withstand both power and neutron flux over time without threatening the integrity of active cooling. Despite decades of research, progress in managing these key issues has not yet demonstrated robust solutions adequate for fusion reactors. Material options remain scarce, and the implementation of liquid metal walls—once considered a promising solution—has not been demonstrated routinely at relevant scales in the tokamak environment. ITER's large size—apart from the requirement to achieve sufficient energy confinement given the limits on toroidal field imposed by the use of conventional superconducting magnets—is itself in part a reflection of the limits imposed by power fluxes in stationary operation. The ITER device is positioned in an operating space where power exhaust should be manageable (assuming edge magnetohydrodynamic instabilities are adequately mitigated), and it will offer an excellent reactor scale platform on which to test heat exhaust strategies for larger tokamaks with much higher fusion power; this ITER contribution is reinforced by the increase in additional heating power in the new baseline. Some of these strategies may also be applicable to more compact tokamak reactor designs (namely high field or tight aspect ratio tokamaks), but the power handling in these devices is likely to be significantly more challenging in view of the reduced surface areas over which power is deposited.

To address these challenges, research and development efforts must face reality. It is sensible to rely on thoroughly tested materials for the foreseeable future, even if this results in higher activation levels; reduction of activation should occur during subsequent optimization stages. Rather than seeking more compact solutions, there is a strong argument for the focus to be on reducing power fluxes to manageable levels across both the divertor and FW. Achieving this would benefit more compact tokamak reactor designs and, further, could enable cost reductions by utilizing stainless steel rather than exotic alloys and costly manufacturing methods. Simplifying design through the use of more economical materials and manufacturing techniques may prove to be more beneficial than merely reducing size.

For those willing to follow this development route, consideration should moreover be given to the potential for fusion reactors to facilitate the incineration of limited amounts of what would otherwise be stored or disposed of as nuclear waste (spent fuel); this could increase breeding ratios by augmenting neutron multiplication within the breeding blanket and provide the power gains which could allow operation at lower fusion power and hence more manageable FW power fluxes.

Tritium breeding also presents an often understated challenge. While fusion is frequently publicly portrayed as relying on unlimited natural resources, tritium breeding from lithium is technically extremely complex. Breeding ratios significantly above unity must be achieved. Maximising usable blanket surface area outside the divertor region, while ensuring adequate shielding—particularly for the inboard region—demands blanket thicknesses exceeding those employed in ITER, typically around 1.3 metres, which also imposes limits to the minimum size of a tokamak reactor. Here again, the uncomfortable reality is that despite decades of research and development, there is currently no breeding blanket design that offers sufficient breeding capacity, fits within ITER's dimensions, and withstands its full spectrum of operational conditions. The majority of explored concepts present challenges related to feasibility, manufacturability, reliability, supply chain logistics, and cost, as recognised by multiple international assessments (see for example [2]).

The choice of blanket coolant adds further complications, since each option comes with its own set of technical and safety challenges. Liquid metal coolants encounter substantial magnetohydrodynamic complications; molten salt (specifically, FLiBe) presents considerable corrosion risks to containment materials; and helium cooling is largely reserved for long-term objectives due to limitations such as power flux management, insufficient nuclear shielding, and heat transport efficiency. Currently, water is considered the most viable coolant option under high-intensity loading, although safety considerations preclude its use with lithium. The most promising strategy at this time entails utilizing lithium in ceramic or oxidized forms, employing lead as a neutron multiplier, and water as the principal coolant. Continued advancement within this field will necessitate a robust technology program to address unresolved issues beyond the scope of the Test Blanket Program, which constitutes a major objective of the ITER mission. Dedicated, smaller-scale fusion devices serving as volumetric neutron sources [26] could enable testing at neutron flux and fluence levels pertinent to future reactors, and could also support the development of a qualification database for fusion core components, advanced sensors, instrumentation, and control systems.

Reliability, Availability, Maintainability, and Inspectability (RAMI) are essential for commercial fusion viability. Even if infrastructure costs and all scientific and material challenges are resolved, a fusion reactor will not automatically become an economically viable power plant. A single subsystem failure can halt the entire plant, and commercial fusion demands extremely high reliability. Fusion plants must ultimately deliver power 70–80% of the time to compete with other energy sources, but PFCs under intense neutron and heat flux make this a formidable requirement. No human can enter a fusion reactor once activated; all maintenance must be remote, modular, and fast. Extended downtimes for blanket replacement or divertor repair would cripple availability. Regulators will require continuous, non-destructive inspection methods to ensure long-term safety, making advanced diagnostics and embedded monitoring systems essential. RAMI considerations are central to fusion power engineering; addressing them with rigor equal to that given to the plasma physics is necessary for fusion plants to achieve the performance required to compete in energy markets.

Looking ahead, the way forward for fusion energy requires a grounded, realistic approach. The past twenty years have brought significant advances in physics, materials, and manufacturing, particularly through the experience gained from designing and now constructing ITER. However, not all fronts have progressed as hoped, and as discussed, some areas require renewed focus. Developing dedicated technology facilities for testing breeding blankets and other components under intense neutron irradiation, exploring new plasma confinement scenarios that alleviate the power exhaust issue, and consolidating manufacturing insights from ITER are critical steps. At the component level, revisiting the manufacturing experiences of key components with input from those who executed the work can yield valuable lessons. At the system level, re-examining and optimizing the overall design based on accumulated experience will likely lead to improvements, such as adjusting magnetic field strength and aspect ratio for more compact, stable, and cost-effective reactors.

7. CONCLUSION

In summary, the ITER Project is progressing strongly, building on a broad range of robust reforms and a new baseline, which, in their execution over the past two years, emphasizing a nuclear quality culture, have resulted in unprecedented levels of performance in terms of schedule and cost efficiency, remarkable for a project of ITER's complexity and scope. As was envisioned, Baseline 2024 is proving to be is a comprehensive and feasible plan for assembly, integrated commissioning and operation, well-attuned to deliver the key objectives of ITER as early as possible. The completion of most major components plus significant progress on all remaining components is being complemented by the installation and commissioning of plant support systems. ITER's successes, its current and anticipated capabilities, and the lessons learned to date are of seminal value to the global fusion community.

Fusion energy's promise is real, but its realization depends on addressing cost, materials, tritium breeding, and RAMI challenges with pragmatic, experience-driven strategies. By consolidating lessons learned from ITER and focusing on realistic, scalable solutions, the fusion community can move closer to making fusion a practical and transformative energy source.

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