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# THE 2024 NEW BASELINE ITER RESEARCH PLAN

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**Abstract**

A new baseline (NB) has been adopted to ensure a robust achievement of ITER Projects' goals, including modifications to the configuration of tokamak and ancillaries, or their phased installation, to minimise operational risks. In the NB, the ITER research plan (IRP) includes three phases for scientific exploitation: (a) start of research operation, with 40 MW of Electron Cyclotron Heating (ECH) and 10 MW of Ion Cyclotron Heating (ICH), with focus on the demonstration of 15 MA operation in L-mode, commissioning of all required systems, including disruption mitigation, and demonstration of H-mode plasma operation in deuterium; (b) DT-1, with 60–67 MW of ECH, 33 MW of neutral beam injection (NBI) and 10–20 MW of ICH, to demonstrate robust operation in high confinement H-mode plasmas in DT up to $Q \geqslant 10$ and for burn durations of 300–500 s within an accumulated neutron fluence of ~1% of the ITER machine's lifetime total, and; (c) DT-2, with up to 67 MW of ECH, up to 49.5 MW of NBI and up to 20 MW of ICH, with the tokamak and ancillaries in their final configuration to demonstrate routine operation in DT plasmas at high Q, and the $Q \geqslant 5$ long-pulse and steady-state scenarios to the final neutron fluence, and to perform R&D on nuclear fusion reactor issues. The logic, basis and evaluations carried out to support the NB IRP are summarized.

## 1. INTRODUCTION

The 2024 new baseline (NB) has been developed by the ITER Project to ensure a robust achievement of the Projects' goals, in view of past challenges. The 2024 baseline includes modifications to the configuration of the ITER device and its ancillaries, e.g. change from beryllium to tungsten (W) as first wall (FW) material, modification of the heating and current drive (H&CD) mix, etc., as well as additional testing of components (e.g. toroidal field coils) or phased installation, start with inertially cooled FW wall before later installation of the final actively water-cooled plasma facing components (PFCs), to minimize operational risks [1, 2]. The NB IRP (see Fig. 1) includes two integrated commissioning and three scientific exploitation phases:

a) First integrated commissioning (IC-I) which focuses on the integrated commissioning of the tokamak components and systems installed in the first assembly phase to demonstrate that their performance can support the foreseen experimental programme in the first scientific exploitation phase (Start of Research Operation). In IC-I it will be demonstrated that suitable vacuum conditions for plasma operation can be achieved in the vacuum vessel (VV) and that the superconducting magnets can meet the requirements to support plasma scenarios up to 15 MA/5.3 T, to cite two of the main outcomes of IC-I. The time allocated to IC-I is 18 months.

b) Start of Research Operation (SRO) with 40 MW of ECH and 10 MW of ICH, which will focus on the demonstration of 15 MA operation in hydrogen H L-mode plasmas, commissioning of all required systems, including the disruption mitigation system (DMS), and the demonstration of H-mode plasma operation in deuterium (D) at 2.65 T. The time allocated to SRO is 27 months.

[1] See A. Loarte, et al., 2025 Plasma Phys. Control. Fusion 67 065023

c) Second integrated commissioning (IC-II) which focuses on the integrated commissioning of the tokamak components and systems, with a vast majority in final configuration, to demonstrate that their performance can support the foreseen experimental programme in the second scientific exploitation phase (First Tritium phase) including the demonstration of the Q ≥ 10 fusion power goal and of tritium (T) breeding with the Test Blanket Modules (TBMs). The time allocated to IC-II is 10 months.

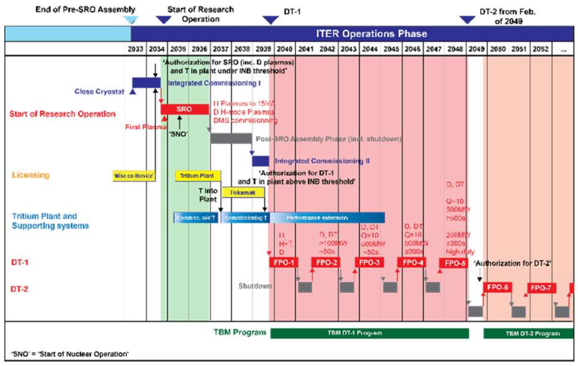


*Figure 1. Operational plan for the execution of the NB IRP to the demonstration of the Q ≥ 10 500 MW fusion power goal in the 1st Deuterium-Tritium phase and the initial campaigns of the 2nd Deuterium-Tritium phase.*

d) First deuterium-tritium phase (DT-1) with 60-67 MW of ECH, 33 MW of NBI and 10-20 MW of ICH, which will demonstrate robust operation in high confinement H-mode plasmas in DT to Q ≥ 10 and for burn durations ≥ 300 s within an accumulated fluence of 3.5 $10^{25}$ neutrons (~ 1% of the ITER machine's lifetime total). The time allocated to DT-1 allows for 5 experimental campaigns of 16 months' duration.

e) Second deuterium-tritium operation phase (DT-2) with up to 67 MW of ECH, up to 49.5 MW of NBI and up to 20 MW of ICH, with ITER tokamak and ancillaries in their final configuration to demonstrate routine operation in DT plasmas at high Q and the Q ≥ 5 long-pulse and steady-state scenarios to the final neutron fluence (0.3 MWy/$m^2$ or 3.0 $10^{27}$ neutrons). The time allocated to DT-1 allows for 5 experimental campaigns of 16 months' duration.

The NB IRP has been developed with strong involvement of experts from all ITER Members, including the International Tokamak Physics Activity, under the coordination of the ITER Organization. The NB IRP describes the objectives, scientific and technical deliverables for each of the three operational (as well as the integrated commissioning) phases. It includes the experimental strategy proposed to be followed to the achievements of the Project's goals in consistency with the configuration of the ITER device together with its ancillary and plant systems and the required licencing steps in every phase. In this paper we focus on the scientific exploitation phases.

## 2. START OF RESEARCH OPERATION (SRO)

The objective of this phase is to develop the operational basis for the plasma scenarios to be later employed for fusion power production in DT-1 and to commission the plasma key systems required to support them (e.g. Plasma Control System (PCS), Advanced Protection System (APS), Central Interlock System (CIS), Disruption Mitigation System (DMS), etc.). The two main operational scenarios to be demonstrated are 15 MA / 5.3T in H L-mode plasmas and D H-mode plasmas up to 7.5 MA at 2.65 T.

The configuration of the ITER tokamak and ancillary systems includes the most core tokamak components in final configuration (VV, superconducting magnets and power supplies, in-vessel coils and power supplies, W divertor, etc.) while the W FW and blanket shield modules are inertially cooled (no water cooling). Similarly, other ancillaries are in partial configuration: fuelling systems and fuel processing to support operation with hydrogen (H) and deuterium (D) plasmas, ECH system with one equatorial and three upper launchers delivering 40 MW to the plasma, ICH system with one antenna delivering 10 MW to the plasma, a partial set of Glow Discharge Cleaning (GDC) electrodes for wall cleaning and boronization and a partial set of diagnostics.

The SRO phase starts with the demonstration of the first tokamak plasma (FP), which requires all tokamak, plant, and auxiliary systems to operate in an integrated way under the PCS satisfying their respective requirements for plasma operation. Then the space for plasma operation is gradually expanded in terms of plasma current ($I_p$), toroidal field ($B_t$) and heating power ($P_{AUX}$) to demonstrate the SRO main goals. Fig. 2.a. summarizes the steps in terms of plasma scenarios to be developed in SRO while Fig. 2.b summarizes the main experimental blocks and the foreseen operational time. Minimization of operational risks and their consequences in SRO, as commissioning of systems and operational range expansion take place, is a key driver for the structure of the IRP in this phase. In addition, key information to confirm DT-1 detailed licencing requirements as well as to guide decisions on upgrades must be obtained at this stage. To this end, the experimental programme in SRO starts in with H L-mode scenarios and $I_p \leq 7.5$ MA and then explores D plasma in H-mode up to 7.5 MA to finally conclude with the development of H L-mode plasmas up to 15 MA/5.3 T. This operational strategy, together with an inertially cooled W FW and limited neutron fluence in D operation, ensures that risks associated with commissioning of control/protection and DMS systems are minimized, as well as that the post-SRO assembly activities can proceed with acceptable doses to workers.

In the first steps after FP in H at 2.65 T, SRO proceeds to diverted plasma operation first at 3.5 MA (step 1) and then up to 7.5 MA (step 2). In these two first steps, commissioning of basic plasma control and protection systems, the DMS, ECH, boronization, wall conditioning and diagnostic systems is carried out interleaved with the expansion of the operational range. This phase concludes with a first attempt to H-mode operation in H plasmas at 2.65 T. Similarly, error fields due to ITER's intrinsic design features and assembly inaccuracies will be identified at this stage and the strategy for their correction will be defined. This will be then later applied to new plasma conditions as the operational range in SRO expands in successive steps. Specific tests will be carried out to quantify the maximum duration of the heated flattops with an inertially cooled FW. Unmitigated disruption/VDE loads will also be characterized at this stage, since it is expected that such loads will not lead to melting of W plasma facing components (PFCs) for the scenarios explored [3], and control schemes to prevent that disruptions/VDE loads damage the water-cooled divertor will be developed (see below). Before start of D operation, $B_t$ will be increased from 2.65 to 5.3 T in a single step, thus ensuring appropriate central heating with ECH to avoid uncontrolled W accumulation. H L-mode scenario development, commissioning and, chiefly, disruption load characterization and mitigation are the focus of R&D in this phase at 5.3 T. Since the main electromagnetic forces on in-vessel components during disruptions scale as $I_p$ x $B_t$, these components will be exposed to forces only 50% lower than the maximum at 15 MA during this phase. Should issues arise human assisted in-vessel repair/replacement activities can proceed before in-vessel activation levels increase.

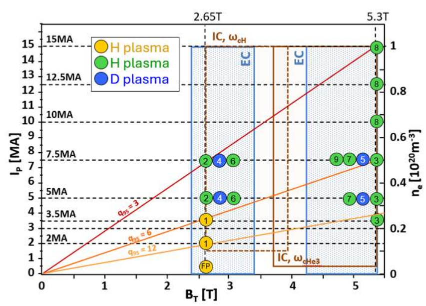


*Figure 2.a. Main operational scenarios for SRO starting from First Plasma (FP). The blue and brown areas display the ECH and ICH operational spaces, respectively. The numbered circles represent the chronological evolution of operation towards $I_p$ = 15 MA $B_t$ = 5.3 T.*

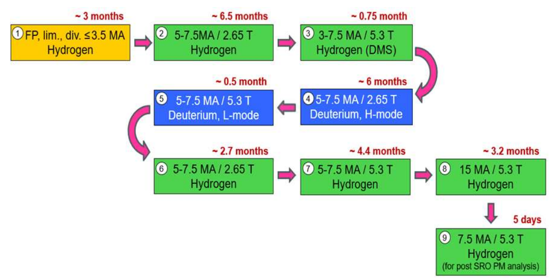


*Figure 2.b. SRO main operational/experimental blocks and foreseen time allocation.*

The main purpose of D operation in SRO is the demonstration of H-mode scenarios and their control up to 7.5 MA/2.65 T including Edge Localized Mode (ELM) control, disruption mitigation, control of divertor and FW power loads and W influxes, etc. At this stage routine operation of ECH and ICH will be required and specific tests of ICH coupling and heating efficiency in H-mode will be carried out, together with assessment of the ensuing ICH-related W influxes. The H-mode scenarios will be initially developed at $I_p \leq 5$ MA, since at this level ELM loads are not expected to melt the W divertor monoblocks. Scenario development will include a range of scans in H-mode plasma parameters by using H&CD, fuelling and impurity seeding schemes, as well as of the plasma clearance to the FW to assess the relative impact on divertor versus W FW sources. Of particular importance at this stage, before increasing $I_p$ to 7.5 MA, is the development of control schemes for ELMs and resulting W influxes, as well as for the critical phases of these scenarios such as H-mode termination and ramp-down. To study the capabilities of the ITER ELM control coils to provide ELM control in these scans, studies of plasma response to the externally applied fields have been carried out for a range of wall clearances, whose results are summarized in Fig. 3. These studies show that, assuming that $I_{coil} \leq 30$ kAt are required to provide ELM control for ~ 5 MA reference H-modes, $I_{coil} \leq 60$ kAt should provide the same degree for the configuration with the largest gap to the FW at the midplane [4].

Once ELM control, disruption mitigation and other required control schemes are demonstrated at 5 MA, $I_p$ will be increased up to 7.5 MA/2.65 T ($q_{95}$ = 3) and H-mode operation will be explored. This will require operation at high coupled power levels since the power required for H-mode access at 7.5 MA/2.65 T is ~ 50% of the maximum coupled power ($P_{LH} \geq 20$ MW vs. $P_{ECH}$ = 40 MW, $P_{ICH}$ = 10 MW). Use of H&CD will allow specific experiments in D plasmas to demonstrate efficient ICH heating (H-minority and D second harmonic heating) providing input to the decision to upgrade the system to 20 MW for operation in DT-1 (see Fig. 4). Similarly, the DMS will be commissioned in H-mode plasmas for the first time with pedestal temperatures in the multi-keV range in ITER,

*See A. Loarte, et al., 2025 Plasma Phys. Control. Fusion 67 065023

addressing issues related to penetration of the shattered pellets [5]. To conclude the D phase in SRO, $B_t$ will be increased to 5.3 T to complete the assessment of ICH coupling/heating/W sources in D plasmas with $He^3$ minority heating, as foreseen to be used in DT-1, and to determine the D H-mode threshold at 5.3 T.

The last part of SRO is executed in H plasmas and starts with a D-to-H change-over which will allow, for the first time, the determination of in-vessel fuel retention in ITER, as well as the demonstration of fuel removal schemes (Ion Cyclotron Wall Cleaning (ICWC) and dedicated tokamak operation), which will be used to define the strategy for T removal in DT-1. This will be followed by further scenario development and commissioning with plasma in H L-modes up to 7.5 MA/2.65T and then up to 7.5 MA/5.3 T. The need for additional operation in H plasmas at this stage is caused by the limited time scheduled in SRO before D operation. D operations must be completed ~ 12 months in advance of post-SRO human entrance into the VV to allow for activation to decay. Once scenario development is satisfactory and all control/protection schemes are commissioned to the level required to support 15 MA/5.3 T operation, the expansion of the operational spaces beyond 7.5 MA/5.3 T will start.

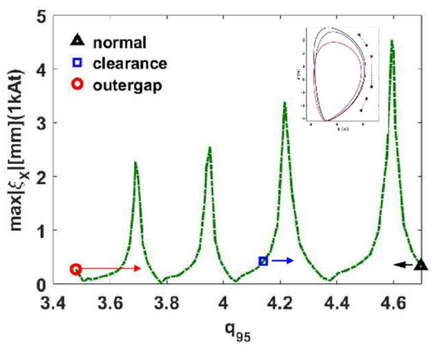


*Fig. 3. Modelled X-point response in mm per kAt in the ELM control coils showing the optimum $q_{95}$ for maximum resonant amplification when the wall clearance is increased. For normal clearance 30 kAt are expected to lead to ELM suppression corresponding to an X-point displacement of 13. 5 cm. A reduction of a factor of 2 in plasma response is found from the nominal configuration to that with an outer wall gap of 45 cm requiring 60 kAt to produce a 13.5 cm X-point displacement [4].*

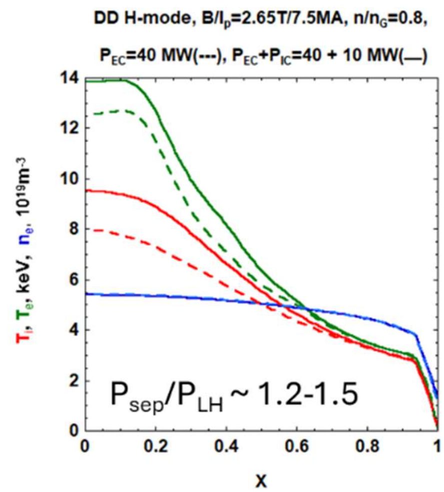


*Fig. 4. 7.5 MA/2.65 T D H-modes with pure electron heating (ECH) and with electron (ECH) + ion heating (ICH 10 MW providing 7.4 MW of ion heating with n = 2 resonance on D) showing the effect of direct ion heating on $T_i$.*

Expansion from 7.5 MA/5.3 T to 15 MA/5.3 T will be done in H L-mode plasmas scenarios in several steps (presently foreseen $\Delta I_p$ = 2.5 MA). At each of these steps, experiments to characterize L-mode plasmas will be carried out together with the development of the scenarios themselves and retuning of the associated control/protection schemes. With increasing $I_p$ and decreasing $q_{95}$, vertical position and MHD control is expected to become more challenging and, thus, the need to retune control schemes at every step. Similarly, at high $I_p$ levels disruption loads approach W melting and the generation of runaway electrons (REs) will increase. The robustness provided by the inertially cooled W FW wall to disruption loads will be used to minimize the risk that such loads impact on the water cooled divertor. For this purpose, detection of disruption and effective mitigation becomes essential together with the use of the vertical stability (VS) in-vessel coils to increase the likelihood of upwards vertical plasma movement during disruptions. This requires actuation of the in-vessel VS coils at least ~ 10 ms before the start of the current quench (see Fig. 5). Unmitigated disruption loads on the W FW will be characterized to the highest possible $I_p$, which is expected to be limited by forces on the VV and in-vessel components or by significant melting of the W FW. With respect to the later, state of the art simulations [3] indicate that, even for 15 MA/5.3 T plasmas, melting of the W FW will be restricted to small areas during the disruption current quench. If confirmed, this opens the way to disruption load characterization at the highest $I_p$ of 15 MA towards the end of this phase, if electromagnetic loads allow. Demonstration of disruption mitigation up to 15 MA/5.3 T in L-mode (and up to 7.5 MA/2.65 T in H-mode) including RE mitigation/avoidance is a major contribution to de-risking later DT-1 operation since $T_e \geq 10$ keV can be achieved in these plasmas, thus, providing a hot-tail source for RE and avalanche amplification similar to that in $Q \geq 10$ scenarios (disruption risks and their retirement/mitigation in SRO are summarised in Tab. 1).

SRO concludes with a series of experiments to characterize erosion/deposition, dust formation and overall impurity dynamics associated with the use of boronization and, possibly, impurity seeding for divertor power load control. To this end diborane ($^{10}B$-enriched) will be used for boronization and $^{21}Ne$ for impurity seeding and their distribution on PFCs, after exposure to a specific series of tokamak discharges, will be measured post-SRO and compared with modelling. The output from these experiments/modelling will be used to refine predictions for DT-1 related to dust formation by boron and fuel retention in boron deposits.

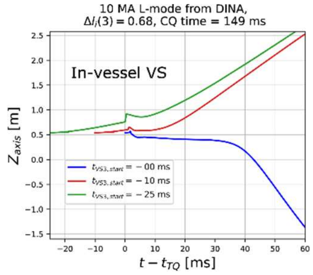


*Fig. 5. Plasma movement after thermal quench for a 10 MA disruption following the energization of the in-vessel VS coils to move the plasma upwards (at 10 and 25 ms before the start of the current quench). In the absence of energization of VS coils this plasma would move downwards.*

| Disruption Phase | CQ | TQ | RE Hot tail | RE β decay (T) | RE Compton e⁻ |
|---|---|---|---|---|---|
| DT | 15 MA | 350 MJ | 10-20 keV | Y | Y |
| SRO | 15 MA<br>**Mitigation for DT-1 fully demonstrated in SRO** | 60 MJ<br>***Mitigation for DT-1 partly demonstrated in SRO*** | 10-20 keV<br>**Mitigation for DT-1 fully demonstrated in SRO** | N<br>*Mitigation for DT-1 not demonstrated in SRO* | N<br>*Mitigation for DT-1 not demonstrated in SRO* |

*Table 1. SRO achievable disruption mitigation goals versus the need for DT-1.*

## 3. FIRST DEUTERIUM-TRITIUM PHASE

The objectives of this phase are to demonstrate the first fusion power Project's goal, namely reproducible operation at $P_{fusion}$ = 500 MW with Q ≥ 10 and $t_{burn}$ ≥ 300 s, as well as high-duty operation (one pulse every 30 min) with $P_{fusion}$ = 250 MW and $t_{burn}$ ≥ 300 s. These objectives are foreseen to require 5 operational campaigns (FPO-1 to FPO-5) to be achieved within a total neutron fluence of 3.5 $10^{25}$ (~1% of the total ITER fluence) while minimizing operational risks. Such requirements have a significant impact on the way the DT-1 IRP is structured (see Fig. 6).

The configuration of the ITER tokamak and ancillary systems includes the vast majority of all systems and components in final configuration including the water-cooled W FW, GDC electrodes, the fuel cycle and the fuelling and diagnostic systems, etc. The ECH system counts with two equatorial and three or four (depending on post-SRO upgrades) upper launchers delivering 60 – 67 MW to the plasma, the ICH system has one antenna delivering 10 - 20 MW (operational by FPO-3 depending on post-SRO upgrades) to the plasma and the NBI system has two heating beams delivering 33 MW (at 870 keV for H in the ion sources) and the diagnostic beam to support charge-exchanged spectroscopy measurements. In addition, TBMs will be installed in two ports and this, together with NBI magnetic field reduction system, is expected to affect error fields.

DT-1 starts with FPO-1 which, to a large degree, aims to reproduce the scenarios already developed in SRO but with the components/systems in final configuration, including their commissioning with plasma as well as of the control and protection systems associated with them. In addition, FPO-1 is structured to de-risk operations for DT-1 and to minimize the consequences of such risks for the IRP should they materialize at this stage. The first phase of FPO-1 will cover L-mode scenarios up to 15 MA/5.3 T and with additional heating up to ~ 103 - 110 MW. This will require the re-identification and correction of error fields, commissioning of all H&CD systems at maximum power and for lengths of ~ 50 s together with the associated control and protection systems (e.g. shine-through, FW and divertor loads, MHD control (sawteeth, etc.), …). Operation at these levels of additional heating allows assessing the performance of the water cooled PFCs (FW and divertor) under plasma power fluxes at similar levels to those in Q ≥ 10 scenarios and will increase the needs for disruption mitigation. Note that, by this stage, the number of unmitigated disruptions must be kept very low since all in-vessel components are water-cooled. Should issues be identified with PFCs, they can be more easily resolved at this stage since in-vessel activation levels are expected to be low (H operation) facilitating repair/replacement activities. The next step in FPO-1 is to de-risk, as far as possible, disruption mitigation which, at this stage, focuses on addressing RE avoidance/mitigation including the effect of the fast electrons from T β-decay. To this end, a series of experiments will be performed for $I_p$ = 7.5 – 15 MA H+T plasmas to demonstrate disruption mitigation while the risk of RE generation (quantified by the product of β-seed and avalanche gain) is gradually increased (see Fig. 7) by a combination of $I_p$ and T concentration steps. Together with the use of ECH central heating at $I_p$ = 15 MA, producing $T_e$ ≥ 10 keV, this will allow the demonstration of disruption mitigation with similar (or higher) sources of hot electrons and avalanche gains to those expected in Q ≥ 10 scenarios, with the exception of those produced by Compton scattering of γ photons produced from in-vessel activation. Operation with T plasmas will also allow the quantification of T retention and of the efficiency of the foreseen removal strategies.

FPO-1 concludes with the demonstration of D H-mode operation up to 7.5 MA/2.65 T with additional power levels up to 103-110 MW. This expands significantly the performance achieved in SRO and will require the

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commissioning and routine use of control and protection systems in this scenario such as: FW and divertor heat load control, ELM control, β control, sawtooth and Neoclassical Tearing Mode (NTM) control, possibly, Alfvén Eigenmode (AE) control, etc. Specific commissioning of neutron diagnostics well as of burn control is foreseen at this stage taking advantage that $P_{AUX}/P_{LH} \sim 5$ can be achieved in these scenarios allowing feedback of a significant fraction of $P_{AUX}$ on plasma parameters to emulate the dependence of fusion power on them.

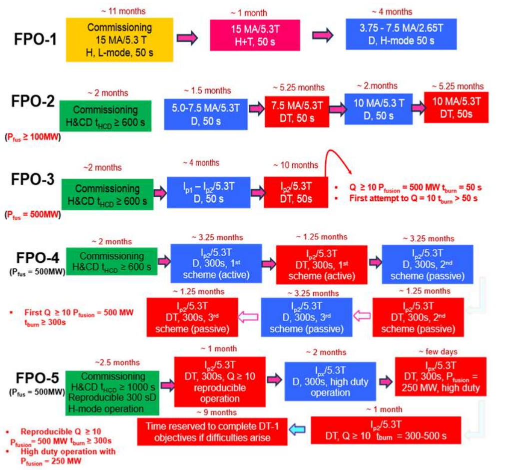


*Figure 6. Sequence of main experimental research activities and foreseen time allocation for the DT-1 phase which includes five operational campaigns (FPO-1 to FPO-5). Note that in FPO-4 several approaches will be followed to extend the burn length; these include the use of H&CD systems to control the current profile and, thus, MHD stability (active) while others are based on pre-forming the current profile before the high Q phase is accessed (passive).*

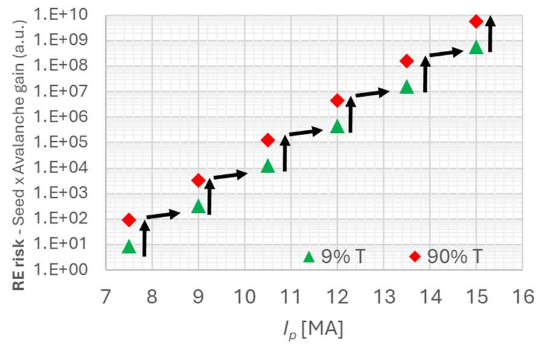


*Figure 7. Stepwise experimental approach to demonstrate RE avoidance during DMS pulses with T, with progressive increase in RE generation risk with increments in Ip and T concentration.*

In FPO-2 the development of DT scenarios towards Q ≥ 10 for burn duration of ~ 50s starts. For neutron fluence economy reasons this is performed by interleaving D and DT scenarios so that basic scenario development and adjustment of control and protection systems are performed in D scenarios with low fluence consumption. R&D in DT scenarios will be focused on fusion power optimization and retuning of control schemes rather than in full scenario development and control scheme adjustments. This D development path is enabled by the larger levels of additional heating power available for DT operation in the NB compared to previous baselines [6]. FPO-2 starts with D H-mode development at 5 MA/5.3 T and expansion to 7.5 MA/5.3 T ($q_{95}$ = 6) which will require re-optimization of control schemes (especially ELM control) as well as of actuators (H&CD). This development path will allow the assessment of plasma confinement in high $q_{95}$ H-modes early in DT-1 in view of their potential for high Q operation in ITER, which typically requires $H_{98}$ ~ 1.3-1.6. Once robust 7.5 MA/5.3 T D H-modes scenarios are demonstrated, DT operation will commence. The initial experiments will contain T at trace level (few %) to characterize T transport and to optimize core plasma T fuelling as well as control schemes (incl. diagnostics and actuators) for D/T mix control. Since these plasmas are D-dominated, the effect of T at trace level on their performance is expected to be small. In follow-up steps H-mode access scenarios with increasing levels of T up to ~ 50 % will be developed. These will be followed by full H-mode DT scenarios with the goal to optimize their fusion performance for timescales of ~ 50 s and tests of burn extension up to 300 s will also be carried out. This development will confirm the potential of $q_{95}$ = 6 plasmas for operation with $P_{fusion}$ > 100 MW, which is expected to require $H_{98}$ > 1.5. In the second phase of FPO-2 a similar development in D and DT plasmas will be repeated for 10 MA/5.3 T plasma, first in D and then in DT (~50-50) with the aim to demonstrate $P_{fusion}$ > 100 MW for $t_{burn}$ ≥ 50 s even with H-mode conventional confinement $H_{98}$ = 1.0. For this development at 10 MA all control schemes developed thus far, including burn control, will be put in practice. The IRP includes a limited exploration of confinement optimization (and, if possible, achievement of $H_{98}$ ≥ 1.5) for 10 MA/5.3 T and burn extension to $t_{burn}$ ≥ 300 s, since 10 MA/5.3 T is the candidate scenario for steady-state operation with $P_{fusion}$ = 400 MW and Q ≥ 5 in DT-2. The extent to which the optimization can be performed will be determined by the time required and the associated neutron fluence consumption, since the priority for DT-1 is to demonstrate the Q ≥ 10 goal as soon as possible. From the results obtained in FPO-2, integrated plasma scenario models within the Integrated Modelling and Analysis Suite (IMAS) will be validated in D and DT H-mode plasmas at 7.5 and 10 MA and used to predict the conditions ($I_p$, $P_{AUX}$, $\langle n_e \rangle$, etc.) for which Q ≥ 10 will be obtained in DT plasmas in FPO-3 as well as the corresponding "D-equivalent" plasma scenario.

FPO-3 focuses on the achievement of $P_{fusion}$ = 500 MW Q ≥ 10 with $t_{burn}$ ≥ 50 s with the aim to demonstrate stationary conditions for all core and edge plasma parameters, plasma-wall-interaction processes as well as helium exhaust. The choice of 50 s is based on integrated IMAS core-edge plasma simulations which indicate that 50 s of stationary burn are required for the helium density in the core plasma to become stationary (see Fig. 8). Note that in this timescale the current profile is still evolving, and this means that deleterious phenomena associated with it (specific MHD instabilities being triggered, degradation of confinement by decreasing magnetic shear, etc.) may be avoided in these short burn scenarios but may materialize when they are extended. The strategy to develop the $Q \geq 10$ is similar to that in FPO-2 by interleaving D and DT operation but now with the additional requirement to identify the conditions in D plasmas that will lead to $Q \geq 10$ in DT. The first step will be to confirm that the conditions identified in FPO-2 for the "D-equivalent" plasma scenario for $Q \geq 10$ can be achieved in practice. If they can be achieved then DT operation will follow. If they cannot, further refinement of IMAS scenario modelling will be carried out and a new "D-equivalent" plasma scenario identified and, then DT operation will follow, as illustrated in Fig. 9. Besides neutron fluence economy, this approach allows identifying plasma scenarios that can deliver the $Q \geq 10$ goal with $I_p < 15$ MA which, if they can be confirmed, would reduce the disruption risk and facilitate their mitigation compared to scenarios at 15 MA. Operation in DT during this phase will focus on performance optimization and tuning of control schemes to maximize Q with specific emphasis on H&CD optimization (e.g. maximizing ion heating), MHD control (sawteeth, NTM, AE, etc.) as well as core-edge integration. At this stage, first tests of operation with the fuel cycle operating in the reactor relevant regime of direct recycling (i.e. removing impurities from the recycled fuel but not separating DT) will be carried out by providing DT for gas fuelling and separate D and T for core fuelling with pellets. By the end of FPO-3 the first attempt (expected to be unsuccessful) to burn extension will be performed.

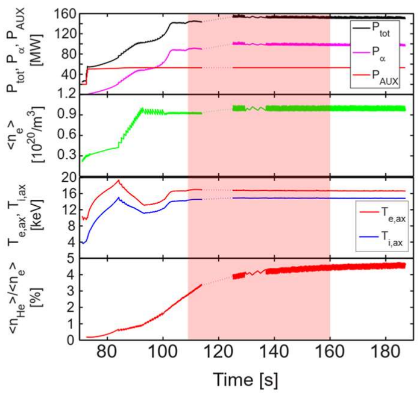


*Figure 7. JINTRAC core-edge integrated simulations of $Q \geq 10$ access in ITER. From top to bottom time evolution of: a) Total heating power ($P_{tot}$), Alpha heating power ($P_{\alpha}$) and auxiliary heating power ($P_{AUX}$), b) Line average density ($<n_e>$), c) Central electron ($T_{e,ax}$) and ion ($T_{i,ax}$) temperatures and, d) core plasma helium concentration ($<n_{He}>/<n_e>$). The band highlighted in red marks the interval of ~ 50 s from the achievement of $Q \geq 10$ to that of stationary helium concentration [7].*

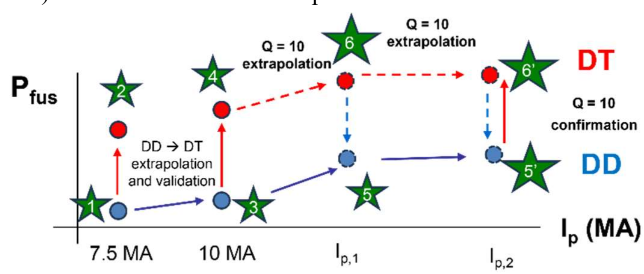


*Figure 8. Experimental steps towards $Q \geq 10$ in FPO-3 following the steps (1-4) in FPO-2. Full arrows correspond to experiments that will be carried out while dashed lines correspond to modelling predictions. This figure describes the case in which $Q \geq 10$ DT predictions are made (6) and the corresponding DD predictions (5) are not verified by experiment. In this case the models will be re-validated with the results of step 5 and a new prediction for $Q \geq 10$ conditions (6') will be made together with the corresponding DD predictions (5'). Once step 5' is verified, the programme in FPO-3 will proceed to step 6. If step 5' is not verified, the cycle of model re-validation and prediction will be repeated.*

FPO-4 focuses on the achievement of $P_{fusion}$ = 500 MW $Q \geq 10$ with $t_{burn} \geq 300$ s, thus, demonstrating stationary plasma parameters including the current profile. The main issue to be addressed at this point concerns the impact of the relaxation of of the current profile on MHD stability and core transport which, in ITER, takes place in timescales of 100's of seconds (Fig. 9). On the basis of present experiments [8] and modelling, the relaxation of the current profile is expected to lead to the triggering of tearing modes associated with the current profile "well" formed in the inner side of the pedestal, since the bootstrap current builds-up in short timescales (few seconds in ITER) which are much shorter than global current profile diffusion (see Fig. 10). Again, for reasons of neutron fluence economy, the development of such scenarios to demonstrate stable operation will be done first in D plasmas and, if successful, they will be reproduced in DT, following which Q will be optimized. To achieve stable MHD plasmas active stabilization of tearing modes will be tried first by applying ECCD in the "well" region near the pedestal top. If unsuccessful, or if the resulting Q is low because of the required peripheral EC power, then passive schemes will be explored first in D and then, once demonstrated in DT plasmas. These include controlling the evolution of the plasma parameters in the L-mode $I_p$ ramp-up and in the H-mode entrance phase to ensure that the current profiles evolves along a stable path to stationarity. Once $Q \geq 10$ for $t_{burn} \geq 300$ s is demonstrated further assessments of DT operation with direct recycling in the fuel cycle will be carried out.

FPO-5 concludes the DT-1 R&D programme; emphasis at this stage is given to reproducible operation with $P_{fusion}$ = 500 MW $Q \geq 10$ with $t_{burn} \geq 300$ s and to high-duty operation in DT (30 min repetition time) with $P_{fusion}$ = 250 MW and $t_{burn} \geq 300$ s. In preparation for DT-2 priority will be given to operation of the fuel cycle in the direct

*See A. Loarte, et al., 2025 Plasma Phys. Control. Fusion 67 065023

recycling mode. The main objective of high duty operation is to demonstrate the capability of the ITER systems and components to operate with high pulse repetition rates in DT, in advance DT-2. This high-duty operation also allows the demonstration of T breeding with the TBMs. For this R&D, specific DT plasma scenarios will be developed and few operational days are allocated to demonstrate 2-shift operation in high-duty in FPO-5. Any remaining time and neutron fluence after high-duty operation is demonstrated will be dedicated to extend the $P_{fusion}$ = 500 MW with Q ≥ 10 scenario up to $t_{burn}$ ≥ 500 s and/or to address fusion reactor issues planned for DT-2, which do not depend on post-DT-1 upgrades.

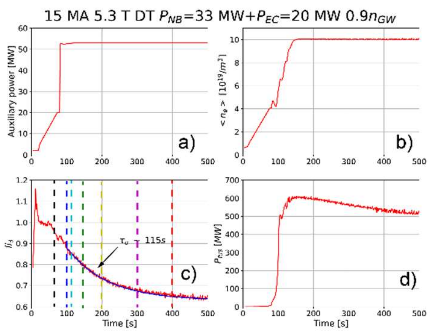


*Figure 10. Time evolution of plasma parameters in JINTRAC core-edge integrated simulations of Q ≥ 10 plasmas in ITER: a) Auxiliary power, b) Electron average density, c) Plasma internal inductance ($l_{i3}$), d) Fusion power [7].*

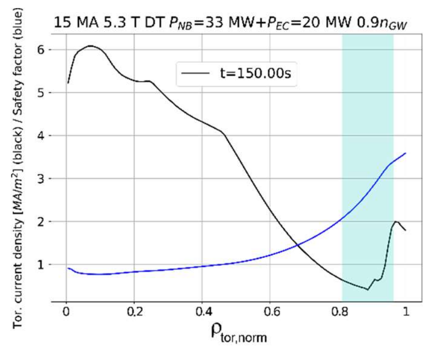


*Figure 11. Current ($j_{plasma}$) and q profiles versus square root toroidal flux at the start of the Q ≥ 10 phase with stationary plasma parameters (except current profile). This corresponds to t = 150 s in Fig. 10 [7].*

## 4. SECOND DEUTERIUM-TRITIUM PHASE

The objectives of this final phase of the IRP are to demonstrate all fusion power Project's goals, namely: high duty operation in Q ≥ 10 scenarios with $t_{burn}$ up to 500s, long pulse ($t_{burn}$ ≥ 1000s) and steady-state operation ($t_{burn}$ ≥ 3000s) with Q ≥ 5 as well as reactor-orientated R&D studies up to the end-of-life neutron fluence of 3 $10^{27}$. The experimental programme is foreseen to expand 5 operational campaigns (FPO-6 to FPO-10) with the initial phase of the FPO-6 campaign giving more emphasis to Q ≥ 10 R&D and then proceeding to interleaving long-pulse, steady-state scenario and fusion reactor research as the FPO-6 and later campaigns progresses. Obviously, the details of the DT-2 campaigns will depend on the outcome of DT-1 research together with the detailed licencing requirements for DT-2. The ITER tokamak and ancillary systems will reach their final configuration at this stage. The major system expected to become available by DT-2 is the 3rd NBI system enabling operation with 49.5 MW, which is required for steady-state operation with Q ≥ 5. In addition, other H&CD upgrades that may not have been implemented in DT-1 could be completed at this stage, namely the ECH system with two equatorial launchers and four upper launchers delivering 67 MW to the plasma and one ICH antenna delivering 20 MW.

## 5. SUMMARY

A new IRP has been developed with strong involvement of experts from all ITER Members, including the International Tokamak Physics Activity, under the coordination of the ITER Organization. The IRP defines the main experimental approaches to be followed to achieve the Project's goals consistent with the new baseline. The IRP takes into account progress in understanding since the 2016 ITER baseline was developed and is structured to address and retire ITER operational risks as soon as feasible and to minimize their impact on experimental availability should the risks materialize. To achieve the successful implementation and execution of the new baseline IRP, once machine assembly is completed, strong collaborations between the ITER Organization and the ITER Member's fusion research institutes in both the experimental and theory/modelling are essential.

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

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