INVESTIGATING LONG-DURATION PLASMA OPERATION WITH THE INTERNATIONAL MULTI-MACHINE DATABASE

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Abstract Combined high-fusion performance and long-pulse operation is one of the key integration challenges for fusion energy development in magnetic devices. Addressing these challenges requires an integrated vision of physics and engineering aspects with the purpose of simultaneously increasing time duration and fusion performance. Since the previous 2023 IAEA Fusion Energy Conference, significant progresses have been made in tokamaks and stellarators including very recent achievement in duration and/or performance. These progresses are reviewed by analyzing the experimental data provided by 10 tokamaks and two stellarators. Data have been gathered and coordination have been provided by the IEA-IAEA international CICLOP group (Coordination on International Challenges on Long duration OPeration).

1. INTRODUCTION

Achieving both high fusion performance and long-pulse operation is a key integration challenge in the development of magnetic fusion energy. Addressing these challenges requires a comprehensive approach that integrates physics, operational strategies, and engineering considerations in order to simultaneously extend plasma duration and reliably increase fusion power. The revised ITER baseline, the need to test material properties under high neutron fluence, and the importance of exploring tritium-breeding blanket technology in dedicated long-pulse operation introduce additional challenges that require further R&D, supported by current experiments.

Since the 2023 IAEA Fusion Energy Conference (FEC), significant progress has been achieved in both tokamaks and stellarators, with recent advancements in plasma duration and performance. These developments are reviewed by analyzing the latest experimental data from ten tokamaks (ASDEX Upgrade, DIII-D, EAST, JET, JT-60U, KSTAR, TCV, TFTR, Tore Supra, and WEST) and two stellarators (LHD and W7-X), expanding on the pioneering work of M. Kikuchi [1]. Data collection and coordination have been facilitated by the recently established IEA-IAEA CICLOP group (Coordination on International Challenges on Long duration OPeration). The published database [2], which initially included data up to January 2022, has been significantly updated for the 2025 FEC to incorporate the latest 2023-2025 experiments (up to May 2025) including recent records performance with new entries provided by DIII-D, EAST, JET, KSTAR, WEST, and W7-X. This paper presents the updated dataset, reflecting recent advancements in long-pulse operation across several key research areas.

2. RECENT EXPANSION OF THE OPERATIONAL DOMAIN FOR LONG PULSE OPERATION

Long-Pulse Operation (LPO) in tokamaks and stellarators requires maintaining stable plasma for a duration much longer than the plasma energy confinement time, while approaching plasma–wall interaction timescales where physical processes may still evolve on very long timescales. It is worth noting that developing the ITER baseline scenario with durations exceeding 100 s will already address many of the challenges of LPO, since all processes occurring on shorter timescales must be mastered and controlled to ensure safe operation.

Since the 2023 IAEA FEC, major progress has been achieved in developing and demonstrating successful LPO in tokamaks and stellarators, with new records set for plasma duration and performance. Consequently, the CICLOP database has been significantly updated with (i) an extensive validation of the existing data, and, (ii) new data from DIII-D, EAST, JET, KSTAR, WEST and W7-X (up to May 2025). To further support the analysis, additional variables were introduced, such as the volume-integrated stored energy, the confinement factor relative to H-mode scaling laws, the core radiated power, the core electron density and temperature, and pedestal densities and temperatures. The updated database now comprises a total of 238 plasma pulses (109 pulses in the published database [2]) with a up to 54 entries per pulse (29 entries per pulse) including information on the operational limits categorised in terms of machine/engineering or plasma physics limits.

The latest advances in injected energy, duration, total injected power, and sustained performance are reviewed and compared using the CICLOP database. Fig. 1 show the total injected energy and Fig. 2 display total injected power as function of high-fusion performance duration. The figures on the right hand-side represent a sub-set of the tokamak data with H-mode edge. The dashed lines in Fig.2 correspond to lines at constant injected energy of either 0.01 GJ, 0.1 GJ or 1 GJ. Data corresponding to experiments performed before the 2023 IAEA FEC have symbols in grey to highlight the progress in the last two years. As further discussed in section 3,

- EAST (2025) achieved repeatable long-pulse operation with an H mode edge (up 1066 s) in fully non-inductive conditions utilizing a water-cooled tungsten lower divertor [3-4];
- WEST (2025) demonstrated long-pulse operation (up to 1337 s) with a full ITER-grade tungsten divertor, reaching up to 2.6 GJ of injected energy under non-inductive conditions [5-6]
- KSTAR (2024) achieved a 102 s high-performance long-pulse discharge using its newly installed W-shaped tungsten divertor [7];
- W7-X (2025) achieved 480 s of operation with 1.3 GJ of energy turnover and up to 1.8 GJ for 360 s, using its actively water-cooled C-divertor [8];
- JET (2023) set its own record for injected energy in ELMy H-mode (up to 450MJ, up to 60s in deuterium plasmas) with the inertially cooled tungsten divertor [9-10].

As shown in Fig. 2, operating at high power for long durations is a necessary condition for achieving high-performance long-pulse operation. In this context, the fusion triple product has been calculated for the updated CICLOP database and plotted as a function of the duration of the high-fusion performance phase (Fig. 3). The figure confirms a reduction of fusion performance when increasing duration from ~1 s to 100 s. Because machines of different sizes naturally achieve different values of the triple product, ITER will be the first facility able to simultaneously reach both high fusion triple product (in the range of 7-12 atm s) and long duration (typically above 100s). The performance reduction has been interpreted as a consequence of LPO typically being achieved in smaller devices under conditions dominated by electron heating and reduced density. To further increase core pressure, tokamaks and stellarators must operate at higher ion temperatures and densities by optimizing the density profiles, the ion confinement and ion heating. In this context significant progress have been made recently:

- W7-X (2025) has sustained high-performance hydrogen discharges with record fusion triple products of 0.077 atm s for up to 24 s and 0.06 atm s up to 40s, enabled by quasi-steady-state fueling from its new CW pellet injector, demonstrating a significant improvement in long-duration fusion performance [8].
- JET (2023) maintained balanced core ion and electron temperatures with an H-mode edge for up 60s achieving a fusion triple product, $n_i T_i \tau_E$ of 0.055–0.06 atm s [9].

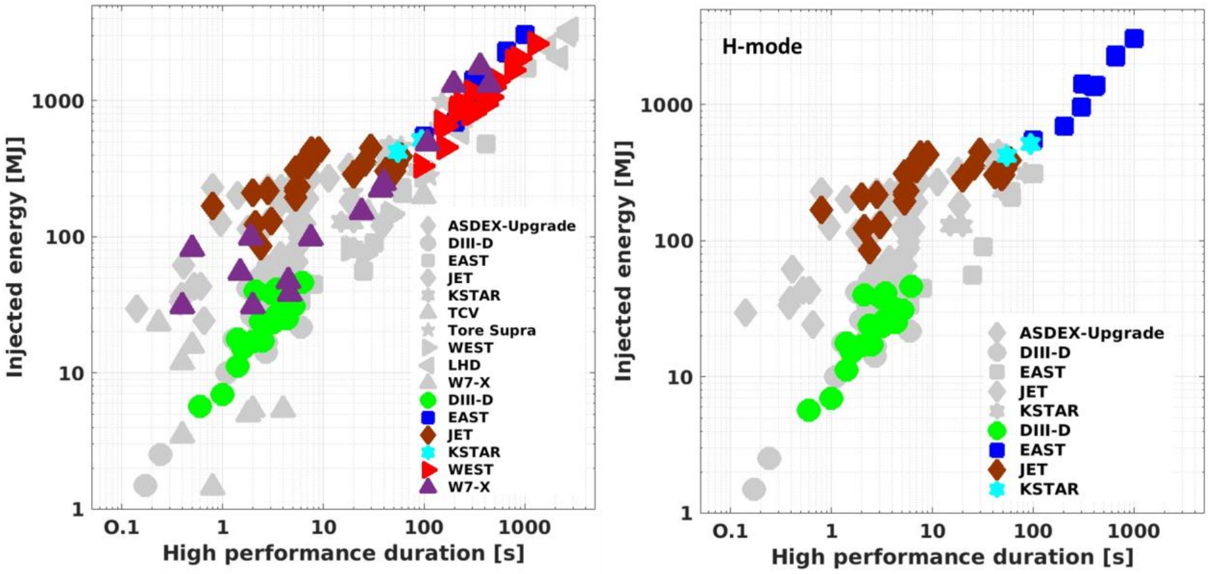

Fig. 1: *(left) Injected energy versus high performance duration for the whole database; (right) for tokamaks H-mode data.*

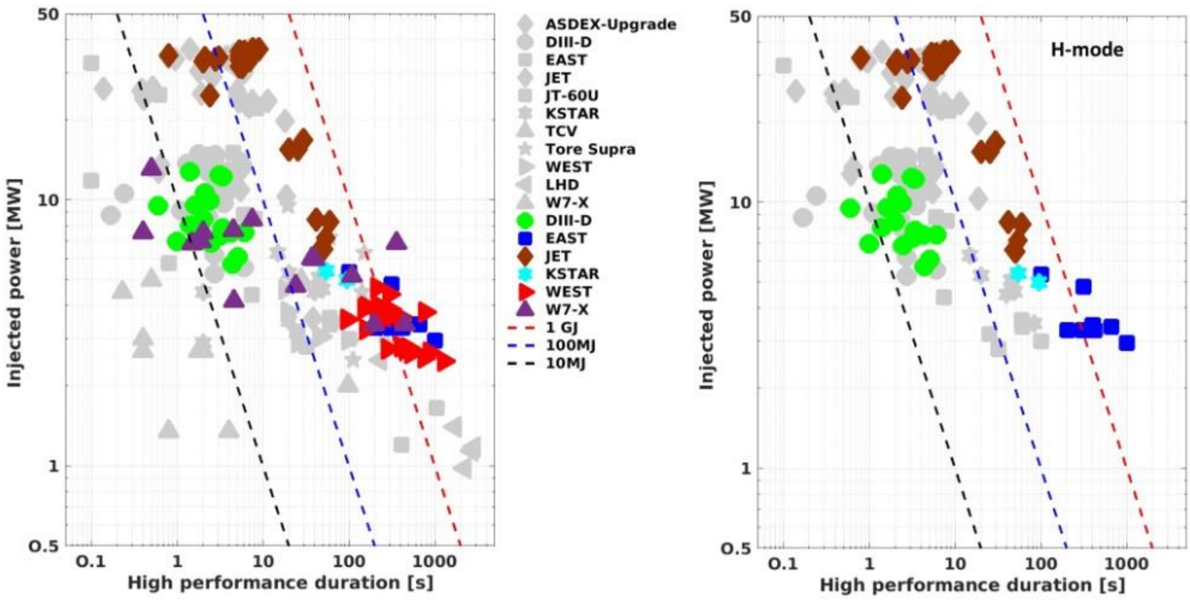

Fig. 2: *(left) Injected power versus high performance duration for the whole database; (right) for tokamaks H-mode data.*

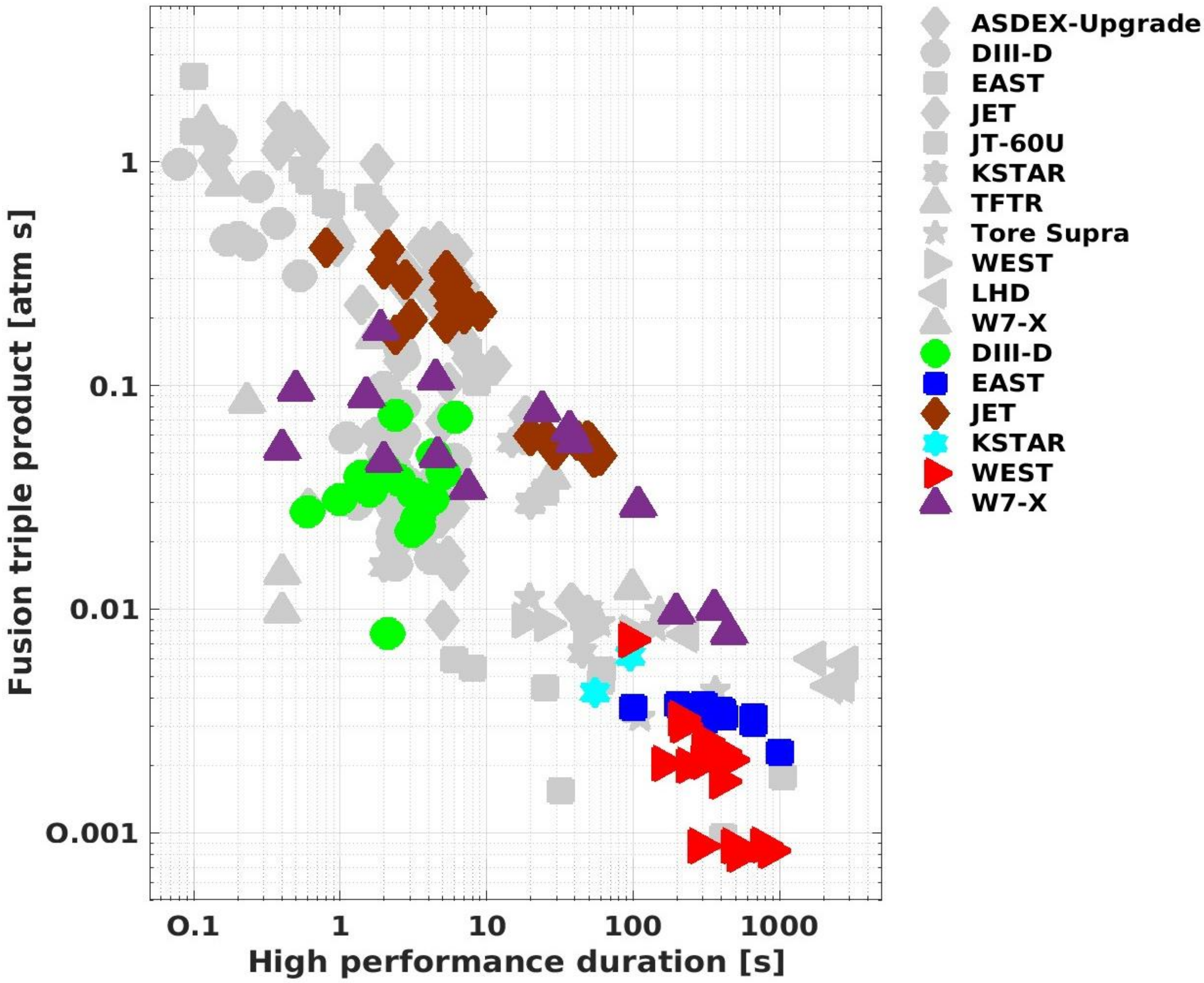

Fig. 3: Fusion triple product, $n_iT_i\tau_E$, in atm s versus the duration of the high-performance phase [s].

3. RECENT PROGRESS ACROSS SEVERAL RESEARCH AREAS

3.1. Long-Pulse Operation in attached conditions

3.1.1. L-mode edge

The high-level objective of the WEST experimental programme is to test and qualify ITER divertor technology under high heat loads and particle fluences (comparable to ITER values), while exploring controlled plasma operation scenarios with tungsten plasma-facing components. Significant and regular progress has been achieved since the full installation and completion of the ITER-grade lower divertor, which is fully representative of ITER's actively cooled divertor technology [5]. In addition, vessel conditioning in WEST is performed using ITER-relevant boronization techniques. In this configuration, pulse durations have been extended from typically 100 s in 2023 up to 1337 s in 2025 as a world record duration [5-6]. The main challenges that had to be overcome to expand the operational limits include: overheating of water-cooled in-vessel components up to their operational limits; outgassing from remote areas; and sporadic events triggered by W-rich flakes entering the plasma core. Predictive simulations using the High-Fidelity Pulse Simulator were performed to define the operational window for fully non-inductive operation, while remaining within all operational limits [11]. These limits include the Greenwald density, infrared safety constraints on water cooling pipes temperature, and the MHD stability of advanced scenarios with reversed magnetic shear. In this context, WEST has demonstrated long-pulse operation (up to 1337 s) with a full ITER-grade tungsten divertor, achieving up to 2.6 GJ of injected energy in non-inductive conditions [5-6]. To reach a stable and reproducible state, a double feedback control system has been integrated with the Plasma facing Component, PFC, protection systems. In this scheme, the transformer flux is maintained at a constant value, while the plasma current (around 230 kA) is controlled by the applied LHCD power (2 MW range). This scenario was found to be extremely robust and resilient to both internal and external perturbations.

Recent enhancements to W7-X include the installation of a pressure water-cooled divertor consisting of CFC blocks welded onto CuCrZr heat sinks, the commissioning of a steady-state pellet injector, and the addition of ten identical cryopumps for active density pumping and control. During the 2025 experimental campaign, W7-X achieved a record in hydrogen plasma of injected energy up to 1.8 GJ for 360 s in attached divertor configuration, with 5 MW of Electron Cyclotron Resonance Heating (ECRH) power [8]. The previous record (Feb. 2023) was 1.3 GJ for 480 s. With the new water-cooled divertor, the surface temperature of plasma-facing components stabilizes rapidly. Crucial for developing long-pulse scenarios is the optimization of divertor configurations and precise strike-line control on the island divertor and/or impurity seeding (c.f. section 3.3). In addition, high-performance discharges with record fusion triple-product values have been sustained for up to 40 s thanks to quasi-steady-state fuelling provided by the new CW pellet injector, combined with improved ion temperature and core electron heating (up to 6 MW) at high density (~1×10^{20} m^{-3}).

3.1.2. H-mode edge

In 2025, EAST [4] achieved a significant milestone by sustaining a record H-mode plasma for over 1066 seconds with lower single-null configuration and total injected energy ≈ 3.05 GJ while the previous longest H-mode duration, reported at the 2023 IAEA FEC, was 403 s [3]. An internal electron transport barrier was formed in combination with an H-mode edge. The toroidal magnetic field was optimized to simultaneously maximize confinement and exploit the synergy between EC and Lower Hybrid waves using a combination of 1.1 MW LHCD and 1.9 MW ECRH. A flux-loop feedback control scheme was developed to maintain zero loop voltage, ensuring fully non-inductive current drive. A grassy-ELM regime was sustained with precise magnetic equilibrium control, minimizing radial drifts of the strike point. The outer strike point was positioned on the horizontal target, enhancing RF power coupling and reducing tungsten sputtering/erosion. Due to the actively water-cooled W-divertor, the divertor surface temperature remained steady below 800 °C. Moreover, the use of low-Z wall coatings (via lithium injection) and real-time lithium powder injections improved particle control and prevented tungsten impurity accumulation throughout the discharge.

KSTAR has achieved a 102 s high-performance long-pulse discharge in ELMy H-mode using its newly installed lower W-shaped tungsten divertor [7]. The primary heating and current drive systems employed for LPO are Neutral Beam Injection (NBI) and ECRH. During LPO, KSTAR has encountered several limits [7]:

(1) rapid temperature rises of the PFCs caused by beam-driven fast-ion orbit losses;
(2) insufficient poloidal flux for sustaining discharges beyond ~102 s;
(3) slow degradation of plasma performance on timescale well above the confinement time (~10^3 τ_E).

During the final months of JET operation in 2023 [9-10], long-duration deuterium discharges were carried out with plasma current I_P = 1.4 MA and toroidal magnetic field B = 1.9 T ($q_{95} \approx 4$). Two types of long-duration H-mode discharges were developed: (i) high-power 30 s ELMy H-mode: 12–14 MW NBI combined with 2 MW ICRH (Ion Cyclotron Resonance Heating); (i) medium-power 60 s H-mode: 4–6 MW NBI combined with 2 MW ICRH. Both scenarios operated in H-mode with regular type-I ELMs, low global radiation levels, and no evidence of core impurity accumulation or deleterious MHD activity. The pulses were stationary from a radiative perspective and exhibited comparable ion and electron temperatures in most cases: ≈ 5 keV in the 30 s high-power cases and ≈ 3 keV in the 60 s medium-power pulses. The experiments achieved the highest total injected energy ever recorded in a single JET pulse (450 MJ) and pushed the operational boundaries of the inertially cooled divertor, with divertor energy of 365 MJ. Despite the moderate total input power, the 60 s discharges reached H-mode with type-I ELMs. Overall, none of the long-pulse experiments exhibited physics limitations; instead, plasma performance and maximum duration were constrained by engineering factors, namely auxiliary power availability, heat exhaust capability, and the lack of non-inductive current drive.

3.2. Long-Pulse Operation: heat exhaust capability

To characterize and compare the heat exhaust capability of the various facilities, a simple proxy has been defined as the ratio of the plasma heating power, P, normalized to the plasma surface, S, P/S. Fig. 4 (left) shows this ratio as a function of the high-fusion performance duration. The highest values of P/S (≈ 0.3 MW/m^2) with a tungsten wall have been obtained on ASDEX-Upgrade. Since 2023, significant progress has been achieved in reaching LPO with duration of about 100 s with P/S ≈ 0.1 MW/m^2, approximately a factor of two below the ITER target.

A second figure of merit, defined in [12], has been explored to characterise the divertor heat exhaust capability of magnetic fusion experiments. It is based on the empirical scaling of the power decay length with the poloidal gyroradius [13]. This parameter is expressed as $P_{sep}B/(q_{95}AR)$, where P_{sep} is the power crossing the separatrix, B the toroidal magnetic field, A the aspect ratio, and R the major radius. For ITER, this figure of merit is ≈ 7-8 MW T/m. This parameter is shown on Fig. 4 (right) versus duration for the CICLOP tokamak database and is up to 3 MW·T·m^{-1}.

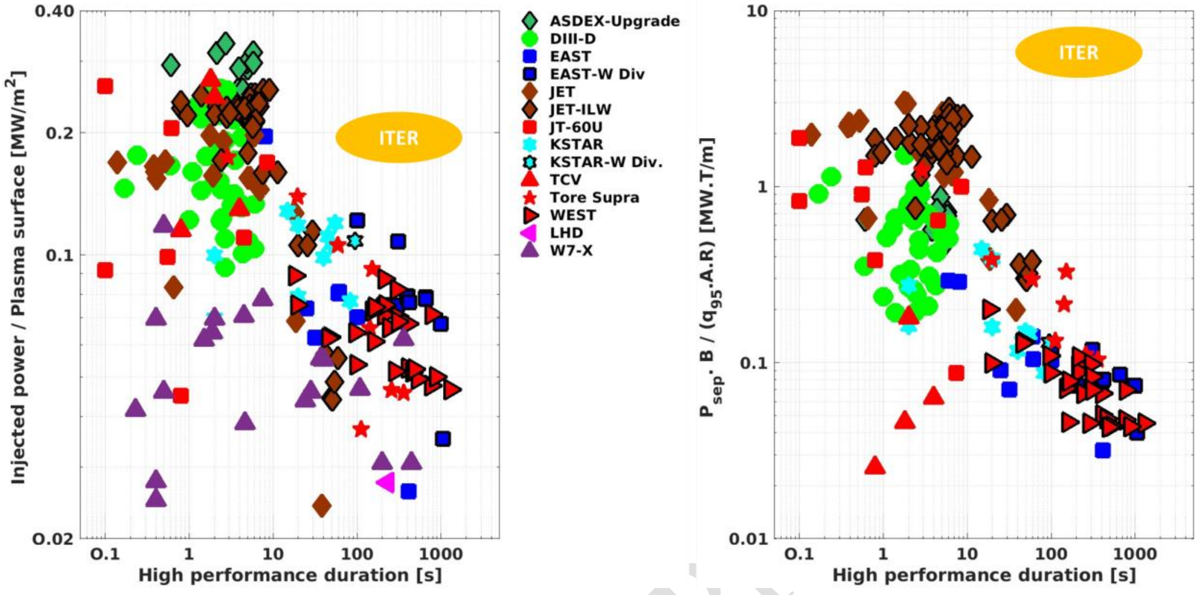

Fig. 4: P/S [MW/m²] (left) and $P_{sep}B/(q_{95}AR)$ (right) vs high performance duration. Experiments performed with a metallic divertor (ASDEX-Upgrade, EAST, JET-ILW, KSTAR and WEST) have symbols with a black contour line.

3.3. Long-Pulse Operation in detached conditions

On W7-X, detached configurations, the power flux on the divertor drops below 1 MW/m² in high-radiating regimes [14-15]. Detachment with intrinsic carbon impurity has been sustained for up to 100 s using a feedback system controlling the plasma density, as radiated power scales with density. With neon seeding, this correlation is decoupled, thereby expanding the operational window. During the 2024–2025 campaigns, detached plasmas lasting up to 320 s with 3–4 MW of counter-ECCD have been achieved without core impurity accumulation—surpassing the 2023 record of ~120 s. On WEST, the X point radiation regime has been sustained for up 35s [16]. On EAST [17], stationary detached ELM-free H-mode plasmas lasting ~50 s have been sustained using feedback control of the target divertor electron temperature with nitrogen seeding. By reducing the plasma current from 400 to 350 kA (q_{95} = 6.8), detachment has been sustained for up to 70 s [17]. On JET, partial divertor detachment has been sustained during the high-power phases using neon seeding in both deuterium and deuterium-tritium (DT) plasmas [18]. JET demonstrated sustained fusion energy production in DT plasmas with an edge radiative mantle as shown in Fig. 5 (next section).

3.4. Extension of duration in deuterium-tritium operation

Fig. 5 summarizes the fusion performance versus duration for DT plasmas obtained in TFTR [19], JET during the Deuterium–Tritium Experiment 1 (DTE1) with the carbon wall/divertor in 1997 [20-21], and during DTE2 and DTE3 with the ITER-Like Wall in 2021 and 2023 [22-24]. Following the installation of the ITER-Like Wall at JET in 2011, along with upgrades to the neutral beam system and diagnostics, major efforts have been devoted to developing stationary and reproducible high-fusion scenarios in deuterium for durations of 5–7 s. These scenarios were successfully transferred to DT plasmas, as shown in Fig. 5, achieving values of $n_iT_i\tau_E$ ranging from 0.45 to 0.50 atm·s sustained for ~5 s. For the three JET DT experimental campaigns, Fig. 5 (right) presents the fusion energy produced during the high-performance phase versus the high-performance duration. The results demonstrate significant progress in fusion energy sustainment, 69 MJ of total fusion energy (a world record) (65MJ over 5 s), and, 22 MJ during 7s a world record for the sustainment of D-T plasmas with a radiative mantle as requested for ITER operation [18]. The pulse durations with the radiative divertor are not limited by plasma physics but by the JET technical capabilities for long duration operation.

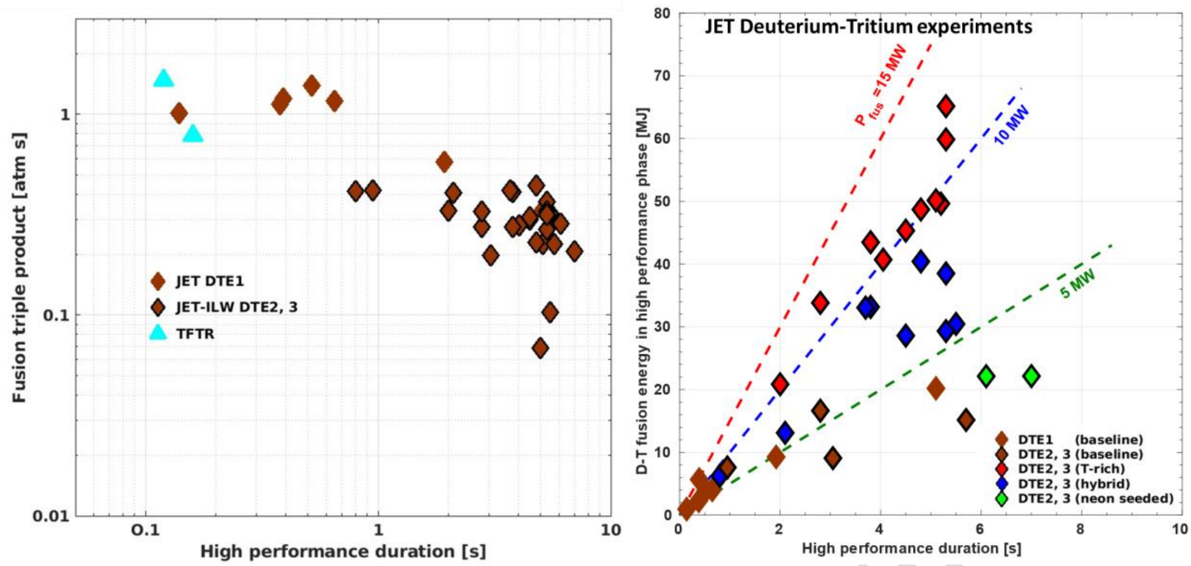

Fig. 5: *(left) Fusion triple product versus high performance phase duration in DT plasmas; (right) fusion energy during the high-performance phase versus high-performance duration for JET DTE1, 2, 3.*

3.5. Long-Pulse Operation at high-normalized pressure

In a steady-state tokamak, both the fusion gain and the bootstrap current fraction must be maximized simultaneously. Thus, to fulfil the requirements of continuous operation and high fusion gain, it is necessary to operate at high normalized beta, β_N. To illustrate this requirement, β_N (and β_p) values have been plotted versus duration for the updated CICLOP tokamak database (Fig. 6, left). The highest β_N values, approaching 4, have been obtained on DIII-D while values close to 3 have been achieved on ASDEX-Upgrade [25]. However, the duration of these discharges is limited by the technical constraints of the facilities. KSTAR has demonstrated steady-state high-β_N operation (β_N ~ 2.7) with a carbon wall and achieved a 25 s-long stationary scenario featuring an internal transport barrier and high ion temperatures (~10 keV) [26]. It is worth noting that ASDEX Upgrade, KSTAR, and, more recently, DIII-D WEST and EAST have extended operational scenarios at high poloidal beta (β_p) (Fig. 6, center), achieving improved energy confinement at plasma densities exceeding the Greenwald density limit for DIII-D [27-28] and EAST [3-4]. EAST, KSTAR and WEST have sustained long pulse (above 100s) at high β_p >2 (Fig. 6, center). The fusion triple products have been plotted versus poloidal beta (Fig. 6, right). At β_p values between 2 to 3, fusion triple products have been obtained at DIII-D with values of 0.05 atm s.

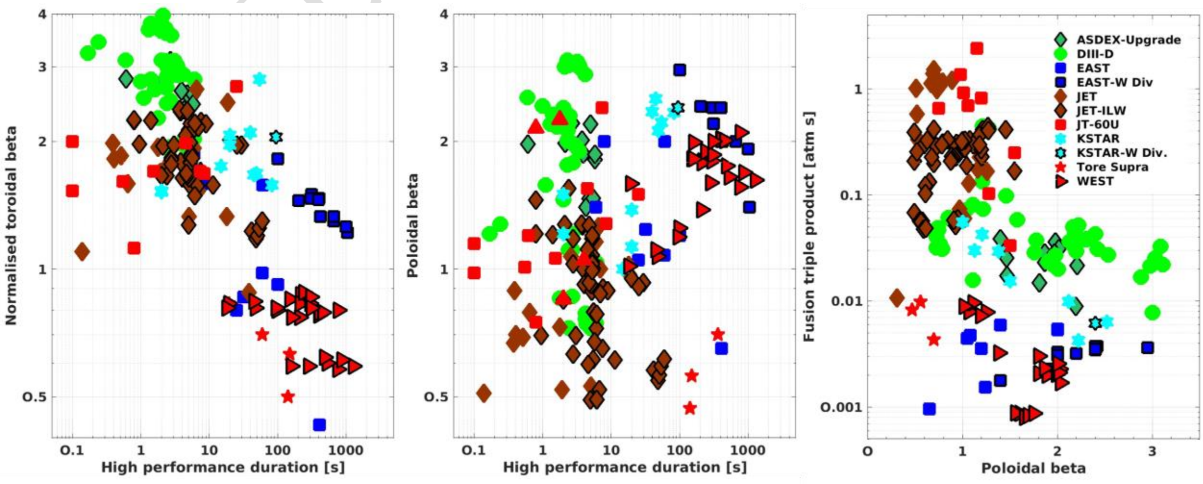

Fig. 6: *β_N (left) and poloidal beta (center) versus duration; (right) fusion triple product versus poloidal beta. Experiments performed with a metallic divertor have symbols with a black contour line.*

4. CONCLUSION

The latest advances in injected energy, pulse duration, injected power, and sustained performance have been reviewed and compared using the multi-machine international CICLOP database. Significant and encouraging progress has been achieved in sustaining high-performance long-pulse operation in both tokamaks and stellarators, supported by superconducting coils, actively cooled components, and various types of metallic walls. These achievements represent a crucial step toward bridging the gap between present experiments and ITER, and they provide a solid foundation for the development of steady-state scenarios in DEMO and future fusion pilot plants.

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