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# Long plasma duration operation analyses with an international multi-machine (tokamaks and stellarators) database

To cite this article: X. Litaudon *et al* 2024 *Nucl. Fusion* **64** 015001

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## Special Topic

# Long plasma duration operation analyses with an international multi-machine (tokamaks and stellarators) database

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Received 30 June 2023, revised 11 October 2023

Accepted for publication 23 October 2023

Published 16 November 2023



## 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. Significant progress has been made in tokamaks and stellarators, including very recent achievement in duration and/or performance. This progress is reviewed by analyzing the experimental data (109 plasma pulses with a total of 3200 data points, i.e. on average 29 data per pulse) provided by ten tokamaks (in alphabetical order: Axially Symmetric Divertor Experiment Upgrade, DIII-D, Experimental Advanced Superconducting Tokamak, Joint European Torus, JT-60 Upgrade, Korea Superconducting Tokamak Advanced Research, tokamak à configuration variable, Tokamak Fusion Test Reactor, Tore Supra, W Environment in Steady-State Tokamak) and two stellarators (Large Helical Device and Wendelstein 7-X) expanding the pioneering work of Kikuchi (Kikuchi M. and Azumi M. 2015 *Frontiers in Fusion Research II: Introduction to Modern Tokamak Physics* (Springer)). Data have been gathered up to January 2022 and coordination has been provided by the recently created International Energy Agency-International Atomic Energy Agency international Coordination on International Challenges on Long duration **OP**eration group. By exploiting the multi-machine international database, recent progress in terms of injected energies (e.g. 1730 MJ in L-mode, 425 MJ in H-mode), durations (1056 s in L-mode, 101 s in H-mode), injected powers, and sustained performance will be reviewed. Progress has been made to sustain long-pulse operation in tokamaks and stellarators with superconducting coils, actively cooled components, and/or with metallic walls. The graph of the fusion triple products as a function of duration shows a dramatic reduction of, at least two orders of magnitude when increasing the plasma duration from less than 1 s to 100 s. Indeed, long-pulse operation is usually reached in dominant electron-heating modes at reduced density (current drive optimization) but with low ion temperatures ranging from 1 to 3 keV for discharges above 100 s. Difficulties in extending the duration may arise from coupling high heating powers over long durations and the evolving plasma-wall interaction towards an unstable operational domain. Possible causes limiting the duration and critical issues to be addressed prior to ITER operation and DEMO design are reported and analyzed.

**Keywords:** nuclear fusion, long plasma duration operation, international multi-machine (tokamaks and stellarators) database

## 1. Introduction and the ‘grand challenge’ of long-pulse operation (LPO)

Controlling fusion plasma for long periods, while gaining experience in steady-state and/or LPO with superconducting magnets, continuous heating and current drive systems, active pumping and fuelling, and active cooling systems that can maintain the plasma-facing components at a stable temperature, is essential for the success of ITER<sup>19</sup> and future fusion reactors.

ITER shall demonstrate the scientific and technological feasibility of fusion energy by combining high-fusion performance burning plasmas (with large values of fusion gain  $Q$ , defined as the ratio of fusion power over the absorbed power

applied externally to maintain a steady state) with LPO [1, 2]. The aims of the ITER experiment are to produce and study burning plasma with a larger fraction of non-inductive current and longer durations in different operational regimes, such as (e.g [1–6]):

- (i) the baseline scenario at plasma current 15 MA and magnetic field 5.3 T, inductively driven burning plasma at  $Q \geq 10$  for an extended duration of 300–400 s,
- (ii) the intermediate or so-called ‘hybrid’ scenario at plasma current  $\sim 12.5$  MA and magnetic field 5.3 T at  $Q \geq 5$  for LPO up to 1000 s,
- (iii) and, ultimately, for steady-state burning plasma conditions in a fully non-inductive current drive regime ( $\sim 10$  MA/5.3 T) for a total duration up to 3600 s.

Access to LPO will be important to demonstrate the availability and integration of essential fusion reactor

<sup>19</sup> In southern France, 35 nations are collaborating to build the world’s largest tokamak, ITER (the “Way” in latin). <https://www.iter.org/>

technologies and to test components such as the tritium breeding blanket modules to close the tritium cycle in a fusion reactor producing electricity on the grid.

LPO in tokamaks and stellarators means addressing control of stable plasma for a duration much longer than the plasma energy confinement time and approaching plasma-wall interaction (PWI) timescales, where physics processes may still evolve over very long timescales: such as first-wall erosion processes, including material ageing, melting, and dust issues (e.g [7–14]). Figure 1 illustrates the characteristic timescales involved in fusion devices when progressively increasing the time duration of operation [15, 16]. To operate fusion reactors for a few hours, even in a pulsed mode of operation, the timescales range from 7 to 8 orders of magnitude, i.e. from typically 100 microseconds for magnetohydrodynamics (MHD) phenomena leading to off-normal transient events, such as disruptions and edge localized modes, seconds for energy and particle confinement times, tens of seconds for internal plasma current radial diffusion time to hundreds of seconds for PWI processes (particle retention in the walls, first-wall particle desorption, erosion, and material redeposition) to several hours for first-wall material ageing processes and thermal equilibrium of the whole device. Among the reported challenges, it is worth pointing out that all these processes occurring over various timescales are inter-connected or nonlinearly coupled. For instance, the long-term erosion of first-wall material will lead to the formation of deposits and potentially trigger disruptions (MHD event). To make progress towards achieving LPO, different physics and technology limits set by these various processes with different timescales (as shown in figure 1) must be overcome. In the case of inertially cooled plasma-facing components, pulse length is limited by the amount of energy these components can store for safe operation. However, R&D on actively cooled components offers the possibility of extending pulse duration, limited by the continuous heat exhaust capability of the whole system. It is worth noting that developing the baseline scenario on ITER with a duration of over 100 s will already address many of the challenges of LPO, as all processes that occur over shorter timescales must be mastered and controlled for safe operation.

LPO is often called the ‘Grand Challenge’ for fusion science [15], requiring pushing the limits of physics and technology integration in a nuclear environment for fusion reactor applications. To address these broad challenges, the development of LPO requires, among other things, a coordinated worldwide effort that encompasses:

- (1) Experimental programs on existing short- (aimed at physics development or proof-of-principle experiments) and long-pulse facilities for tokamaks and stellarators;
- (2) Technology R&D programs (e.g. actively cooled plasma unit components, superconducting magnets, linear facilities);
- (3) Control methods and recovery techniques to maintain a fusion burning plasma within a safe and stable operating domain that can be transferred to ITER and to thermonuclear fusion reactors.

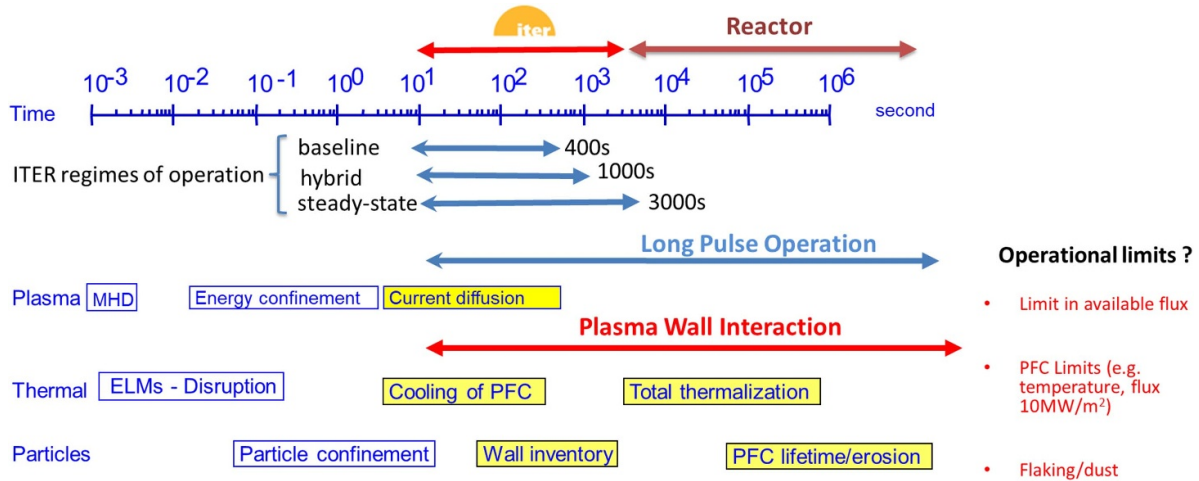
Indeed, the development of fusion energy with stable and safe reactor operation requires:

1. The sustainment and the control of the kinetic and magnetic configuration within the stable boundary domain;
2. Safe and reliable operation of the superconducting magnets;
3. Control of the PWI processes, where all the plasma-facing components are actively cooled and monitored to stay within a safe domain of operation;
4. Efficient fuelling and pumping systems for continuous wave or continuous waveform (CW), control of the fusion fuels, and management of the tritium cycle;
5. Efficient external CW heating and current drive systems for fusion burn access and control;
6. Integrated real-time control for machine protection, safe operation, performance with minimum set of CW diagnostics, CW data acquisition and large data processing tools;
7. CW power supplies, and a heat transfer system for high pulse repetition rate;
8. Comprehensive mastering of the nuclear aspects for operating facilities with high deuterium–tritium 14.1 MeV neutron fluence conditions.

To facilitate the exchange of information at the international level on these overarching challenges, the International Atomic Energy Agency (IAEA) has established a network of experts called CICLOP, which stands for **C**oordination on **I**nternational **C**hallenges on **L**ong duration **O**peration. The medium/long-term objective of the CICLOP group is to promote activities and to collect and disseminate information on the physics and engineering issues of LPO for tokamaks and stellarator facilities. To further strengthen the synergy between tokamaks and stellarators, the group has set up a high-level 0D multi-machine database that combines physics and technology information for quantifying recent and significant progress on LPO. The group has used the CICLOP database to (i) identify gaps in physics and technology for LPO, and (ii) assess the limiting factors in duration or fusion performance in present experiments. Very recently, this activity has been reported and discussed more broadly within the frame of the International Atomic Energy Agency (IAEA) Technical Meeting devoted to Long-Pulse Operation of Fusion Devices [14–16 November 2022, IAEA Headquarters, Vienna, Austria, <https://conferences.iaea.org/event/258/>].

This paper will report on the development and analysis of the 0-D multi-machine database combining recent results obtained in tokamaks and stellarators. Within this context, it will also provide a high-level summary of the discussions and challenges raised during the aforementioned 2022 IAEA Technical Meetings on Long-Pulse Operation of Fusion Devices [<https://conferences.iaea.org/event/258/>]. After this introductory section that provides the broad context, the paper is divided into three main sections devoted to the:

- (i) database description and the operational domain towards LPO with the description of the existing limiting factors (section 2);



**Figure 1.** Characteristic timescales involved in fusion devices of typically ITER size. The exact value for the various timescales depends on the exact machine size, plasma parameters, and first-wall technology.

- (ii) description of the progress towards the sustainment of fusion performance over long duration by continuing and expanding the pioneering work of Kikuchi and Azumi [17] (section 3);

Section 4 will provide a conclusion highlighting the scientific gaps for the development of LPO.

## 2. Operational domain for LPO

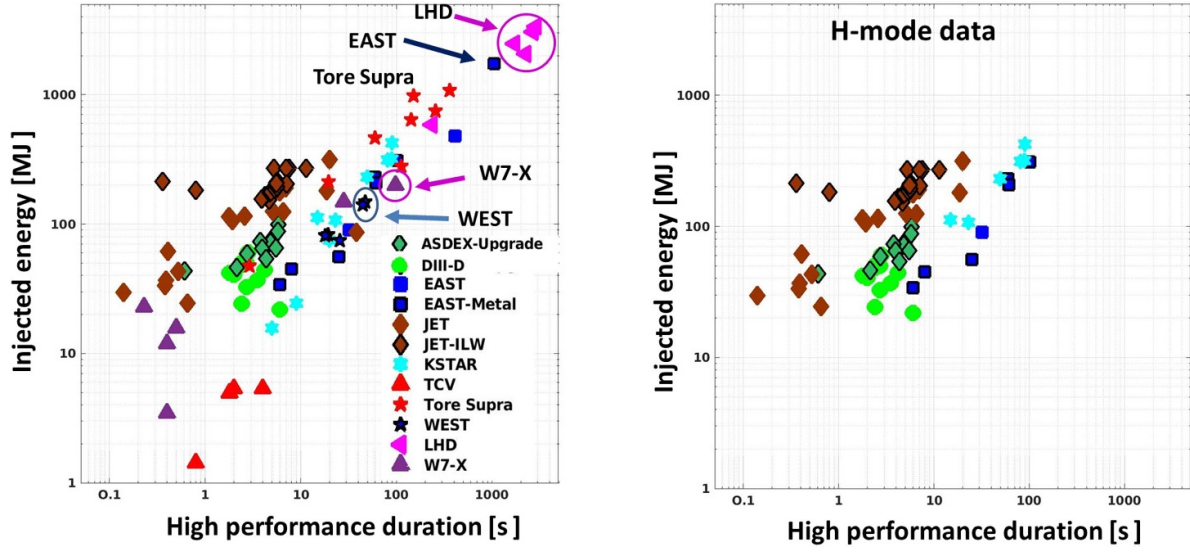
Significant effort has been made by the CICLOP group to collect and validate a 0D multi-machine database with data coming from both tokamak and stellarator experiments. The database includes experimental data provided by ten tokamaks (Axially Symmetric Divertor Experiment (ASDEX Upgrade) with tungsten plasma-facing components, DIII-D, Experimental Advanced Superconducting Tokamak (EAST), Joint European Torus (JET) with a carbon wall, JET with the ITER-like wall (JET-ILW), JT60-U, Korea Superconducting Tokamak Advanced Research (KSTAR), tokamak à configuration variable (TCV), Tokamak Fusion Test Reactor (TFTR), Tore Supra, W Environment in Steady-State Tokamak (WEST)) and two stellarators (Large Helical Device (LHD) and Wendelstein 7-X (W7-X)). The latest references for these facilities are provided by the overview papers published in Nuclear Fusion following recent IAEA Fusion Energy Conferences [<https://iopscience.iop.org/journal/0029-5515/page/Fusion%20Energy%20Conferences>], i.e. ASDEX Upgrade [18], DIII-D [19], EAST [20], JT60-U [21] JET [22–24], KSTAR [25], TCV [26], TFTR [27], Tore Supra and WEST [28, 29], LHD [30], W7-X [31]. In addition, annex 1 provides a list of references per facility for the discharges selected in the CICLOP database together with some indication (when available) of the plasma regime. The database comprises a total of 109 pulses and around 3200 data points (on average 29 data per pulse) and is accessible through an IAEA

web page: <https://nucleus.iaea.org/sites/fusionportal/ciclop/SitePages/CICLOP-DB.aspx>. The pulses have been selected by the scientific contact persons for the respective facility and correspond to the most representative fusion scenarios (cf. annex 1 like ITER baseline scenario, hybrid scenarios, reversed shear scenario, transient hot-ion modes, regimes with internal transport barrier, etc) achieved either in terms of duration or/and fusion performance. In the table of annex 1, an indication is also provided when the discharges have been obtained in fully (or quasi fully non-inductive) conditions (e.g. with zero loop voltage at the plasma edge for tokamaks). For the JT-60U and TFTR data, the pulses are the ones provided by the dataset published by Kikuchi and Azumi [17]. The database of LHD is completely open, and the information on the selected discharges is available from the following link: <https://doi.org/10.57451/lhd.analyzed-data>.

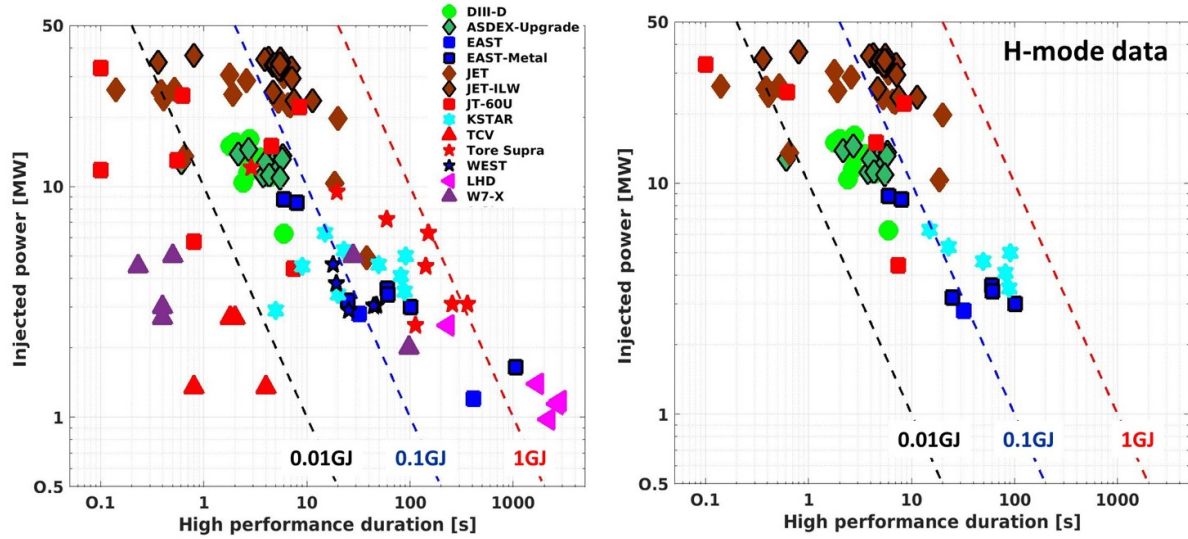
The data provided by the CICLOP's members acting as contact persons for the respective facilities listed above consists of: the facility's name and type (tokamak/stellarator), pulse number, date of the pulse, a published reference, regime of operation with general characteristics (H-mode or L-mode), main fuel species (hydrogen, deuterium, tritium, helium), first-wall and divertor materials, information on the actively cooled components technology, injected additional power [MW], injected energy [MJ], plasma duration [s], duration of the high-performance phase [s], core ion temperature  $T_{i0}$  [keV] and density  $n_{i0}$  [ $10^{20} \text{ m}^{-3}$ ], the plasma energy confinement time  $\tau_E$  [s], plasma current [MA], toroidal field on axis [T], safety factor, major radius [m], minor radius  $a$  [m], elongation, and triangularity. For this reference paper, the data has been collected up to January 2022. The descriptions and definitions of the variables are provided in annex 2.

Figure 2 is the plot of the total injected energy versus the high-fusion performance duration drawn either for the whole database or a subset of the tokamak database with improved confinement based on the formation and sustainment of an edge pedestal transport barrier, namely the H-mode regime. Figure 3 is the same dataset, where the injected power has been





**Figure 2.** (left) Injected energy versus high-performance duration for the whole database; (right) for tokamak H-mode data only. Experiments performed with a metallic wall (ASDEX Upgrade, EAST-Metal, JET-ILW, and WEST) have symbols with a black contour line.



**Figure 3.** (left) Injected power versus high-performance duration for the whole database; (right) for tokamak H-mode data only. The dashed lines correspond to lines at constant injected energy (the injected power times duration) of either 0.01 GJ (black), 0.1 GJ (blue), or 1 GJ (red). Experiments performed with a metallic wall (ASDEX Upgrade, EAST-Metal, JET-ILW, and WEST) have symbols with a black contour line.

plotted versus the high-fusion performance duration for the whole database (3-left) and for the H-mode data (3-right). The dashed lines in figure 3 correspond to lines at constant injected energy (the injected power times duration) of either 0.01 GJ, 0.1 GJ, or 1 GJ. Experiments performed with a metallic wall (ASDEX Upgrade, EAST-Metal, JET-ILW, and WEST) have symbols with a black contour line. The graphs indicate that long-pulse regimes (typically above 100 s) are sustained at reduced power, i.e. typically below 5 MW. This is one of the challenging issues when LPO is operated in the H-mode regime, where the power needs to be above a given power threshold to access and sustain the edge transport barrier with good confinement properties (figure 3(right)). Figures 2 and 3 provide an overview of the operational domain in terms of

injected energy and power obtained in present tokamaks and stellarators, which is further described below.

The record of injected energy (3.3 GJ/2859 s, nearly 48 min) with 1.2 MW of injected power has been obtained on the superconducting stellarator LHD in helium plasmas with an actively water-cooled stainless steel wall and carbon divertor [32, 33] that maintain a constant surface temperature by extracting, in a continuous manner, the heat at the surface of the plasma-facing components. In these experiments, it has been reported that plasma duration and performance were limited by the overheating of some carbon flakes deposited on the plasma-facing components, where the heat could not be properly extracted [32]. The flakes detaching from the plasma-facing components could lead to radiative collapses

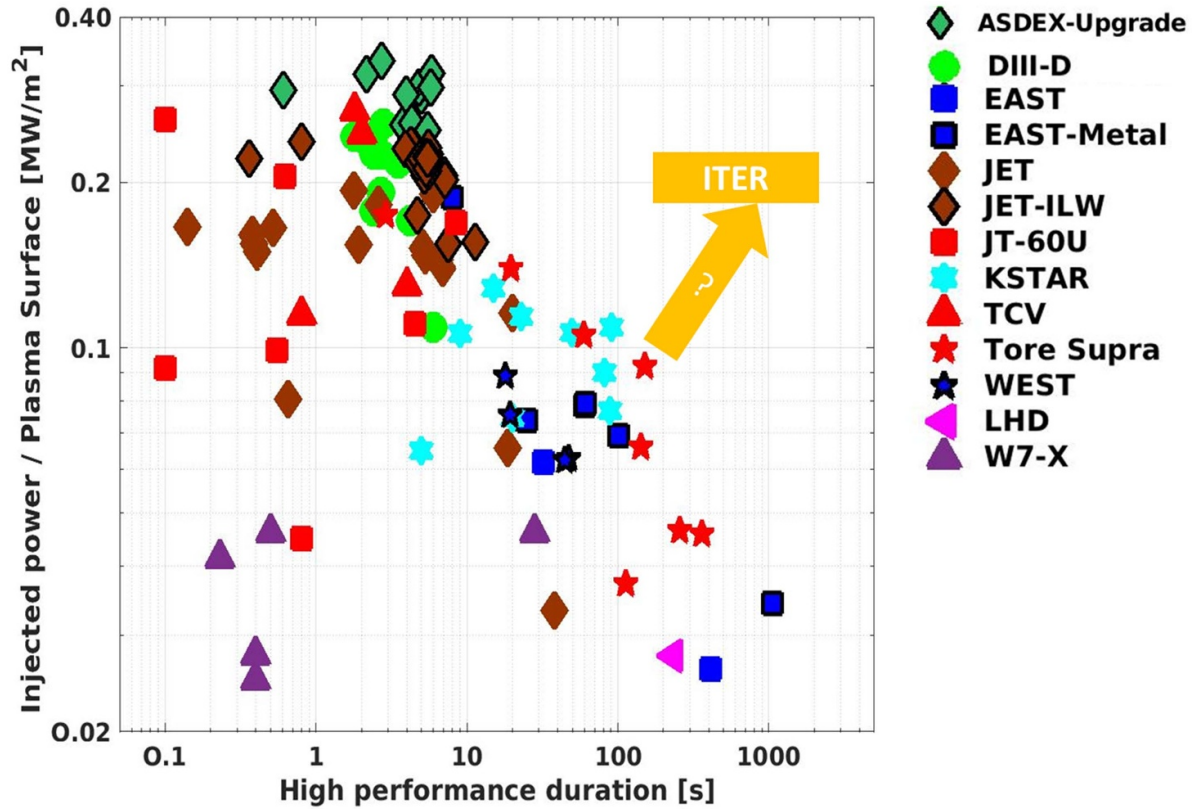
by entering into the hot plasmas and radiating energy. On the stellarator line, W7-X [34] has completed its initial phase of operation from December 2015 up to November 2018, where the energy limit was set by the inertially cooled carbon plasma-facing components. During this phase, plasma durations have been reached up to 100 s by injecting 2 MW of electron cyclotron heating, and were limited by the injected energy that can be accommodated without exceeding a maximum surface temperature set by the inertially cooled system (indeed, without cooling, the surface temperature is continuously increasing with the injected energy) [35]. In a preliminary attempt to reduce the surface temperature and the heat load on the components, operation with detached plasmas conditions has been successfully developed, where 30 s long discharges have been sustained at a power level of 5 MW with a reduction of peak heat flux by a factor of 8–10 compared to attached plasmas [36, 37]. After this initial operational phase, an actively cooled first wall and divertor have been installed during the machine shutdown up to the end of 2021, and scientific exploitation restarted at the end of 2022 with the purpose of significantly increasing the capability to inject high power over long durations [38, 39].

EAST has been recently equipped with an actively cooled tungsten divertor and has reached world record injected energy (1.7 GJ/1056 s) in a tokamak with a total radio-frequency (RF) power of 1.65 MW [40, 41]. In a fully non-inductive regime (without a residual inductively driven ohmic current, where the plasma current is driven by a combination of external current drive sources and the self-generated bootstrap current), the pulse duration was limited by the maximum RF power that could be coupled in LPO (e.g. hot spot formation on the antenna guard limiters or on the divertor). Indeed, it has been recently reported in [41] that ‘the limitations of the long-pulse H-mode plasma are from hot spots in the local area of the lower divertor, due to leading edges, and also from sputtering and erosion from the RF-antenna and guard limiter’. To reach this new record, EAST has implemented major upgrades to enhance the machine capabilities towards LPO in terms of diagnostics and control, heating and current drive systems, and with a lower tungsten divertor to enhance particle and power exhaust capabilities with active water-cooled technology. The previous injected energy record for tokamaks was set by Tore Supra with actively cooled carbon fibre composite tiles on limiters, where 1.07 GJ discharges have been sustained for durations of 6 min and 18 s with 3 MW of lower hybrid powers with no sign of saturation of the fuel retention in the carbon wall [42, 43]. It is worth noting that similar operational limits, as previously discussed for stellarators on LHD, i.e. carbon-flake formation with local overheating leading to disruptions, have also been reported on Tore Supra [44]. The upgrade to Tore Supra, WEST, involved replacing the actively cooled plasma-facing carbon components with a full tungsten divertor to test the ITER divertor technology, which can withstand the high heat and particle fluxes expected in steady-state operation [28]. The WEST experiment also features access to elongated ITER-relevant plasma shapes, improved heating and current drive systems and diagnostic tools, as well as an improved

control system [45] to manage the plasma and its interactions with the tungsten walls. In the first phase of operation (2017–2020), the WEST lower divertor has been equipped with a few prototypes of ITER-grade plasma-facing units integrated into the inertially cooled W-coated components. In this configuration, WEST has already obtained 53 s long discharges sustained with 3 MW of injected power [29, 46, 47]. During this phase, the operational space was limited by the absence of active cooling of the lower divertor. A fully actively cooled divertor with an ITER plasma-facing unit was installed in 2021, and operation started in 2022 with the objective to further extend the pulse duration in a reliable manner and address the consequences of high fluence operation on the ageing of the plasma-facing components [29, 48, 49].

It is worth noting that figures 2(right) and 3(right) indicate that the tokamak database with an H-mode edge is more limited in particular in conditions with metallic walls, i.e. with contributions from the ASDEX Upgrade with an inertially cooled W wall and divertor [50], DIII-D [51] and KSTAR [25] with an inertially cooled carbon wall, EAST when equipped with an actively cooled W divertor, and JET equipped with the so-called ‘ITER-like wall’, consisting of a beryllium wall and a W divertor but inertially cooled. Indeed, long-pulse H-mode operation requires not only additional powers to be coupled above the L to H power threshold but also to be sustained reliably over long durations. Within this context, EAST [52] and KSTAR [25] have sustained H-mode operation for durations up to 100 s with injected energy up to 425 MJ (as in the end of January 2022). This is major progress towards the development of sustained H-mode regimes for the ITER baseline of operation.

Significant effort should be made to develop LPO at high CW injected power, which is extremely challenging in practice since it requires the coupling of high power in a continuous and reliable manner in configurations where the energy could be extracted continuously without exceeding local overheating of some plasma-facing components. 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,  $P$  (including alpha heating for burning plasma conditions like ITER), normalized to the whole plasma surface,  $S$ ,  $P/S$  in  $\text{MW m}^{-2}$ . Figure 4 shows this ratio  $P/S$  calculated for the CICLOP database versus the high-fusion performance duration. This figure highlights the challenge in terms of the heat exhaust capability towards LPO. The highest values of  $P/S \sim 0.3 \text{ MW m}^{-2}$  have been obtained on the ASDEX Upgrade with a tungsten wall, but it is limited in duration by the machine’s technical capability (no active cooling and normal conducting magnet). LPO with duration above 100 s has been reached with  $P/S \leq 0.05 \text{ MW m}^{-2}$  and at reduced power below 5 MW. There is an important gap (a factor 4 is  $P/S$ ) to be filled in the coming years between the presently achieved values and the ITER targets. Indeed, for ITER, the expected ratio of  $P/S$  is in the range of  $0.2 \text{ MW m}^{-2}$ , to be sustained from 100 s up to 3000 s (when adding the externally injected power to the self-alpha heating power produced by the D-T fusion reactions,  $P$  is in the range of 150 MW).



**Figure 4.** Plasma heating power normalized to the plasma surface ( $P/S$  in  $\text{MW m}^{-2}$ ) vs the high-performance duration. Experiments performed with a metallic wall (ASDEX Upgrade, EAST-Metal, JET-ILW, and WEST) have symbols with a black contour line.

The CICLOP group has elaborated, in collaboration with the EUROfusion Operation Network [53], a table summarizing, for both tokamaks and stellarators, the potential limits that should be overcome when reaching high-performance LPO. The objective is to identify, for the pulses in the database, what

are the physics and machine limits that prevent higher fusion performance and/or higher duration, as already discussed. The identified limits are summarized in the table below for both the machine/engineering limits (section A) and the plasma physics limits (section B):

#### A. Machine/engineering limits

1. Limit in available flux
  - 1.1. Max available flux is reached
2. Limit in energy ( $I^2t$  limit) or forces for the coils
  - 2.1. Divertor coil energy limit
  - 2.2. Poloidal field/toroidal field (PF/TF) coils energy limit
  - 2.3. Error field energy coils limit
  - 2.4. Force limits on coils
3. Limit in injected power and/or energy
  - 3.1. Max. energy limit that can be exhausted by the cooling system is reached
  - 3.2. Max. power reached (limit in performance)
    - 3.2.1. Power could not be increased further, e.g. generator limits, reactive power limits
  - 3.3. Max. duration of injected power reached
    - 3.3.1. Energy limits in sub-system components, generator limits, etc
4. Limit in energy and/or temperature for plasma facing components (PFC)
  - 4.1. Limit on wall or divertor temperature or heat flux is reached
  - 4.2. Limit on heating systems: radio-frequency (RF) guards limiters or antenna overheating, ion neutral beam injection (NBI) dumps temperature or energy limit reached, NBI shine through limits
5. Limits in measurements in control system
  - 5.1. Current plasma measurement drift
  - 5.2. Neutron and gamma limits
  - 5.3. Gas limits



## B. Plasma physics limits

1. Limit in MHD stability (current and pressure)
  - 1.1. Pressure/beta limits
  - 1.2. Current instabilities: kink modes and neo-classical tearing modes
  - 1.3. Disruption force
2. Limit in core/pedestal confinement
  - 2.1. Ion temperature clamping in dominant electron-heating regimes
  - 2.2. Limits in pedestal pressure
3. Limit in plasma radiations
  - 3.1. Core impurity accumulations (e.g. W in the core)
  - 3.2. ‘unidentified flying object’ resulting from first-wall erosion, leading to radiative plasma collapses
  - 3.3. C, W Be-flakes overheating, leading to radiative collapses
4. Limit in density
  - 4.1. Uncontrolled density evolution (wall recycling evolution)
  - 4.2. Stability limit approaching density limits
5. Limit in wall/divertor erosion
  - 5.1. Flakes or dust production (that can detached and lead to disruptions)
  - 5.2. Erosion and migration
  - 5.3. Transient events (disruption and edge localised modes (ELMs))

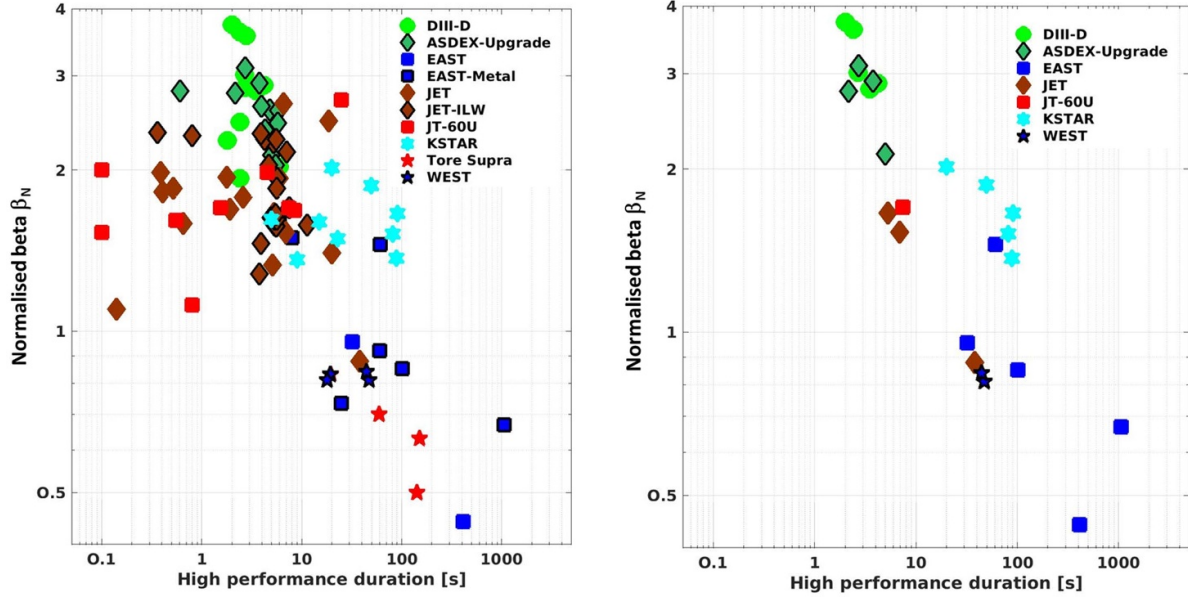
To progress in LPO development, discharges should remain in the stable operational domain, i.e. away from the identified plasma physics and engineering limits. This objective should be achieved by developing integrated control (continuous control) and event handling (asynchronous control) methodology that should steer the plasma pulses away from the limits and maintain the discharge in the stable operating domain, even with the occurrence of unforeseen events (event handling or an exceptional handling strategy) (e.g [54, 55]). This is a scientific challenge for the coming years and, in addition, the development should be carried out using algorithms or methodology that could be transferred to the ITER plasma control system to optimize the operational time on ITER with a limited number of operational pulses in a nuclear and complex environment [55–57].

Within this context, facilities that do not have long-pulse capabilities with active cooling systems and superconducting magnets (AUG, DIII-D, TCV, JET in the CICLOP database of operating facilities) should continue to play a leading role by proposing proof-of-principle experiments, particularly for model validation within (or close to) the operational limits and extrapolation (e.g [50, 58]), by developing physics experiments that provide the scientific basis for extending the duration, and by validating controllers.

For instance, the DIII-D and EAST research programme has been further developed due to a joint collaboration coordinating research towards long-pulse high-performance operation by firstly developing the scenario physics basis on DIII-D over short durations and, secondly, by adapting and extending the scenario to LPO with a metallic first wall on EAST by taking care, in particular, of new aspects related

to core-edge integration and a different heating and current drive mix [59, 60]. The progress in the joint development of a high poloidal-beta tokamak has recently been reviewed in [61]. Indeed, different routes towards steady-state conditions could be explored in short-pulse facilities in a more flexible manner at reduced operational risks before the extrapolation to long-pulse facilities. Inertially cooled components offer a wider and safer operational space, for instance, by avoiding the risks of water leak, leading to a long-shutdown time of the facility, due to local overheating and damage of components operating at high heat flux. Within this context, AUG [18, 50], DIII-D [19, 51, 62], and TCV [26, 63] are actively exploring different access conditions to high normalized pressure ( $\beta_N$  as defined in annex 2) with different magnetic and/or divertor configurations by combining a high fraction of self-generated bootstrap current with an externally driven current, as recently reported at the 2022 IAEA TM on Long-Pulse Operation of Fusion Devices [14–16 November 2022, IAEA Headquarters, Vienna, Austria, <https://conferences.iaea.org/event/258/>].

In a steady-state tokamak reactor, the fusion gain (which is proportional to the toroidal beta) and the bootstrap current fraction (which is proportional to the poloidal beta) have to be maximized simultaneously. Therefore, to fulfil both the requirements of continuous operation and high fusion gain, it is necessary to operate at high values of  $\beta_N$  (cf. for instance, the review [64]). To illustrate this requirement,  $\beta_N$  has been plotted versus the duration of the high-performance phase in figure 5 for the tokamak CICLOP database (figure 5(left)) and for the regimes which are fully (or quasi-) non-inductive (figure 5(right)), as listed in annex 1. High  $\beta_N$  values (approaching 4) have been obtained on DIII-D



**Figure 5.** (left) Normalized pressure,  $\beta_N$ , versus high-performance duration for the CICLOP tokamak database; (right) a similar graph for the discharges in fully (or quasi-) non-inductive conditions.

and the ASDEX Upgrade in non-inductive conditions, but the discharges are limited in duration due to the technical limits of the facilities (no active cooling and no superconducting magnets).

### 3. Fusion performance and long-pulse duration

In this section, the fusion performance is characterized by the fusion triple product, as in the pioneering work of [17],  $nT_{i0}\tau_E$ , where  $n$  is the fuel density in the plasma core,  $T_{i0}$  is the fuel ion temperature in the core, and  $\tau_E$  is the volume-averaged plasma energy confinement time. As indicative values, it has been shown that in deuterium-tritium (DT) mix plasmas [65], and for an optimal range of ion temperature (14 keV), that the minimum value of triple product to be reached is  $nT\tau_E$  is  $4.6 \times 10^{20} \text{ m}^{-3} \text{ keV s}$  (or 0.73 atm s) to achieve the fusion amplification  $Q = 1$  (scientific break-even) and  $29 \times 10^{20} \text{ m}^{-3} \text{ keV s}$  (or 4.6 atm s) for ignition ( $Q = \infty$ ), where the external heating source could be switched off. Optimizing the fusion triple product will maximize the fusion amplification factor,  $Q$ , despite the fact that there is no direct proportionality between the two quantities [65]. For ITER scenarios, as defined in Green *et al* 2023 [4] and recalled in the introduction of this paper, the indicative values of the triple product are in the following ranges:

- (i) baseline scenario (300–400 s):  $nT_{i0}\tau_E \approx 75 \times 10^{20} \text{ m}^{-3} \text{ keV s} = 12 \text{ atm s}$  with  $n \approx 1.1 \times 10^{20} \text{ m}^{-3}$ ;  $T_{i0} \approx 20 \text{ keV}$ ;  $\tau_E \approx 3.4 \text{ s}$ ;
- (ii) hybrid scenario (1000 s):  $nT_{i0}\tau_E \approx 51 \times 10^{20} \text{ m}^{-3} \text{ keV s} = 8 \text{ atm s}$  with  $n \approx 0.9 \times 10^{20} \text{ m}^{-3}$ ;  $T_{i0} \approx 21 \text{ keV}$ ;  $\tau_E \approx 2.7 \text{ s}$ ;

- (iii) Fully non-inductive scenario (3600 s):  $nT_{i0}\tau_E \approx 44 \times 10^{20} \text{ m}^{-3} \text{ keV s} = 7 \text{ atm s}$  with  $n \approx 0.6 \times 10^{20} \text{ m}^{-3}$ ;  $T_{i0} \approx 25 \text{ keV}$ ;  $\tau_E \approx 2.5 \text{ s}$ .

The fusion triple product (in pressure time second, atm s) has been calculated for the CICLOP database and plotted as a function of the duration of the high-fusion performance phase, as illustrated in figure 6(A). The figure indicates a significant reduction of fusion performance by two orders of magnitude when increasing the duration from  $\sim 1 \text{ s}$  to  $100 \text{ s}$ , confirming previous analysis [17, 66]. It is clear that machines of different sizes will achieve different values of triple product, and it is only on ITER that a high-fusion triple product and duration could be obtained simultaneously. Nevertheless, even for a given facility without long-pulse capabilities (like active cooling, superconducting magnets), where high-fusion performance has been reached for short durations and attempts have been made to extend duration up to the maximum technological limits (as in the ASDEX Upgrade, DIII-D, JET, and JT-60U), the extension in duration is obtained at the expense of the fusion performance (cf. the individual points for these facilities). In fact, as will be further detailed below, LPO is usually obtained at reduced power (cf. the previous section), reduced plasma current, and in regimes of low density and high electron temperature to optimize the non-inductive current drive effects with, therefore, low fusion performance or triple product. Within this context, figure 6(B) is the same figure as figure 6(A), where the fusion triple product is plotted as a function of the duration of the high-performance phase, but only for the fully non-inductive regimes, as listed in the table of annex 1. It confirms that, whereas the ASDEX Upgrade, DIII-D, JET, and JT-60U have obtained fully (or quasi-) non-inductive regimes, the operation could not be prolonged due

to the respective technical limits of the facilities. Only EAST, KSTAR, WEST, LHD, W7-X, and the near future JT-60SA will explore, over long durations, high-performance fully non-inductive regimes prior to ITER operation.

Future facilities (such as ITER) will have to face the challenges of the technological and the scientific capabilities to combine high-fusion performance of burning plasma with  $Q$  larger than one for durations well above the confinement time with an actively cooled wall and superconducting magnets. The scientific challenge will be to find an optimal path where plasma conditions will be maintained in a stable manner. Within the context of ITER scenario preparation, JT-60SA, as a satellite facility to ITER, will have the capability to explore LPO up to 100 s with actively cooled plasma-facing components to cope with heat flux of  $10 \text{ MW m}^{-2}$ , and high injected power (40 MW/100 s) together with advanced integrated plasma control for developing an ITER relevant scenario with detached operation by seeding extrinsic impurities like neon or argon [67, 68].

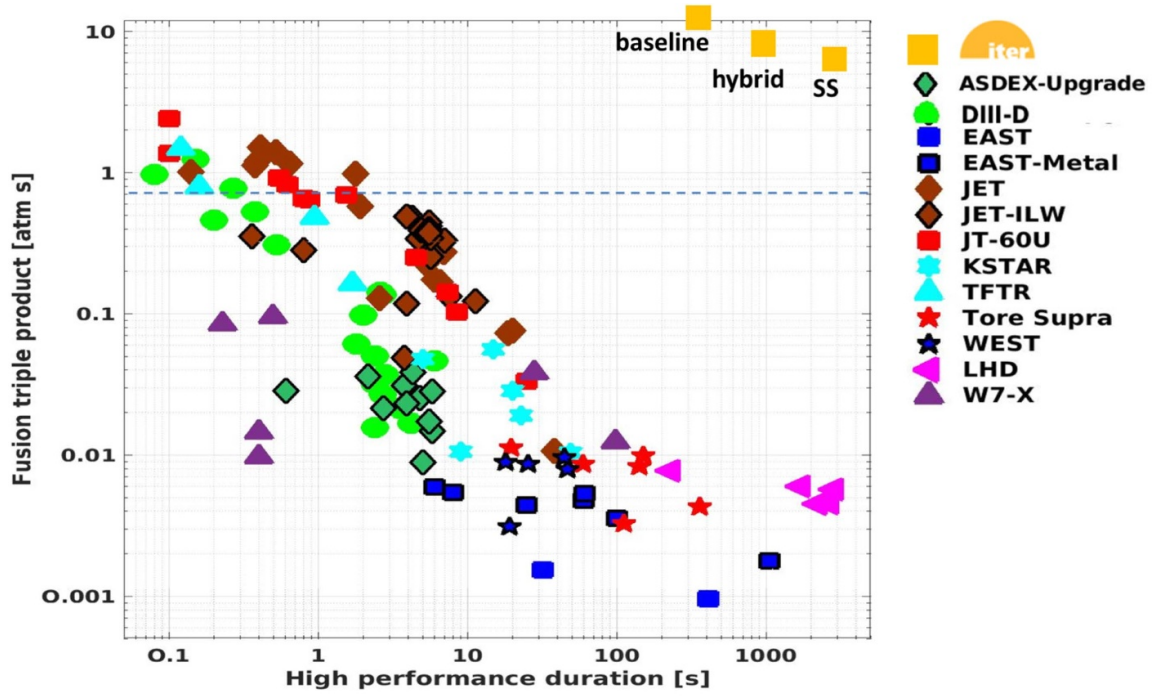
The core ion pressures [atm] and ion temperatures have been plotted separately versus the high-fusion performance duration (figure 7) with a view to better disentangle some of the reasons for the reduction in fusion performance with duration. Indeed, the figures indicate that LPO in superconducting tokamaks or stellarators (EAST, LHD, Tore Supra, WEST, and W7-X) are achieved in a domain with dominant electron heating (using mainly RF heating schemes like lower hybrid heating and current drive, electron cyclotron heating and current drive, and minority ion cyclotron resonance heating) but at reduced density to maximize core electron heating and the non-inductive current drive effect generated by external power sources. Consequently, for durations typically above 10 s, these regimes are obtained at reduced ion temperature ( $\leq 3 \text{ keV}$ ), where the electron and ion fluids are decoupled. In the quest of increasing the fusion performance over long durations, KSTAR has recently sustained a regime with high  $T_{\text{io}} \sim 10 \text{ keV}$  for a duration of 20 s, as shown in figure 7, with low loop voltage to reduce the primary flux consumption [69]. Nevertheless, this regime is also obtained at reduced density (consequently, without a significant increase in the core ion pressure and fusion triple product), where a high fraction of fast ions is reported to be an essential ingredient to improve the thermal performance of the hot plasmas. A route for LPO at higher density and higher core pressure remains to be found in these pioneering experiments. On stellarators, when operated with dominant electron heating and reduced density, a similar low level of ion temperature is reported (typically below  $1.5 \text{ keV}$  in W7-X [70, 71]). This is partly attributed to the lack of ion heating and reduced energy collisional exchange power between the electron and ion fluids that scales like the square of the electron density.

To further increase the core pressure, tokamaks and stellarators should be operated simultaneously at a higher ion temperature and density due to, for example, density profile optimization. A higher ion temperature could be reached by directly injecting higher powers coupled to the ion fluid or, like on ITER or future reactors, by increasing the electron temperature and density to enhance the collisional exchange between the electron and ion fluids to provide a broad ion-heating source. Indeed, ion heating will result from the collisional power transfer from the electrons (externally heated and/or self-heated by the fusion-born alpha particles generated by deuterium–tritium reactions) to the ions that depend on plasma density and electron ( $T_e$ ) to ion ( $T_i$ ) temperature difference as the energy exchange power scales as  $n^2 \times (T_e - T_i)/T_e^{3/2}$ .

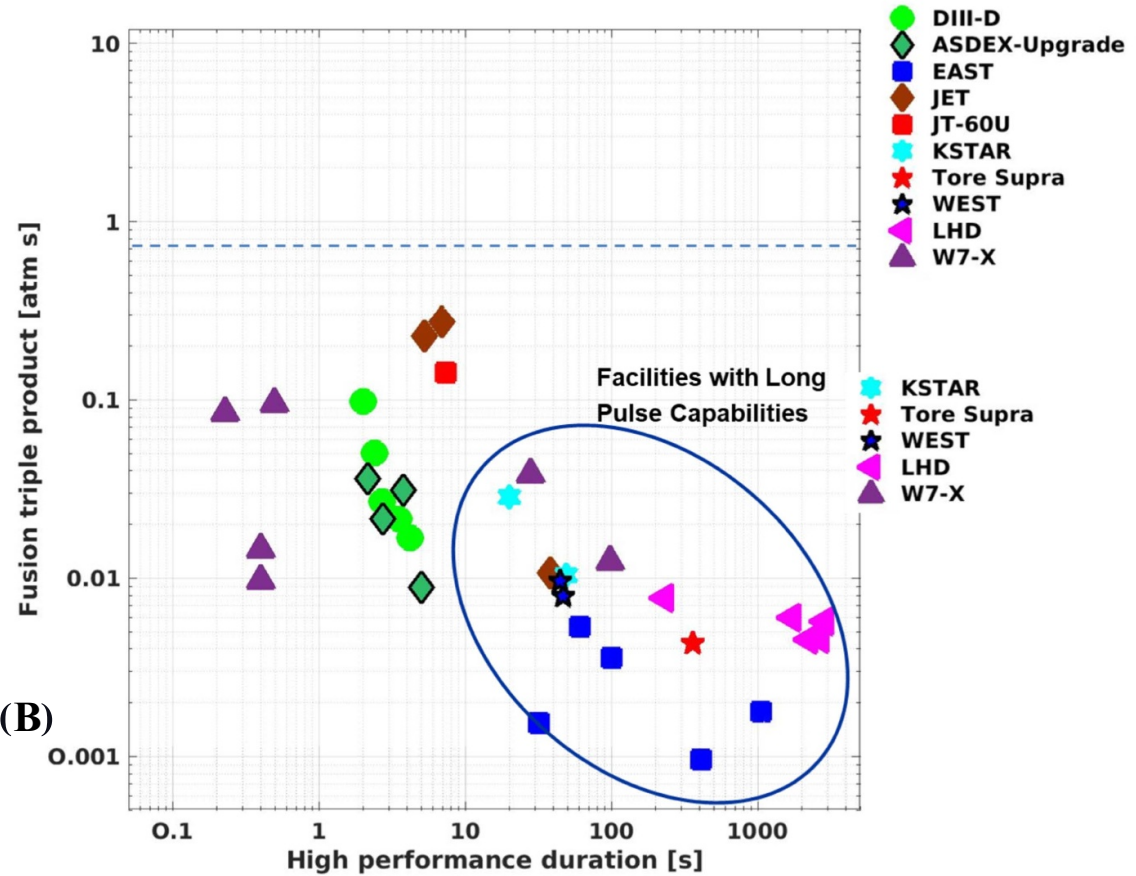
A promising 30 s LPO has been achieved in W7-X at high density in detached divertor regimes and led to a sustained high-fusion triple product, as illustrated in figure 7(top) [36, 37]. Stellarator configurations at high density for long-duration operation are sustained without the need for an external non-inductive current drive (that is usually optimized at reduced density in tokamaks) since the magnetic configurations are set by the external superconducting magnetic field configuration. In tokamaks, operation at high density (close to the density limit) requires operation with a large fraction of self-generated bootstrap current to minimize the need for an external current drive.

Figures 8 and 9 are similar graphs to figure 6 but where a subset of the CICLOP's database has been used by selecting either facilities with a metallic first wall/divertor or experiments performed with a deuterium–tritium mix fuel. Figure 8 indicates that significant progress with metallic wall operation in support with ITER and DEMO has been made on the ASDEX Upgrade, EAST equipped with the tungsten divertor, JET-ILW and WEST in its first phase of operation, but the database is still relatively limited. Within this context, and to expand the expertise and know-how for the preparation of ITER operation, facilities will be upgraded to be equipped with an actively cooled tungsten divertor on WEST for phase II of operation beyond 2022 [29], KSTAR [25], and over a longer term on JT60-SA [49, 67] and W7-X [31] in a second phase of operation after an initial exploitation for both facilities within the wide operational space provided by the carbon plasma-facing components. In addition, new facilities like the Divertor Tokamak Test (DTT), which is being built in Frascati (Italy) [72, 73], will explore long-pulse and high-power operation with an actively cooled tungsten divertor, where applied powers will be increased in a phased approach up to 45 MW [74] to investigate energy and particle exhaust challenges for ITER and future fusion reactors.

The last graph, figure 9, is a summary of the fusion performance versus high-fusion duration for deuterium–tritium



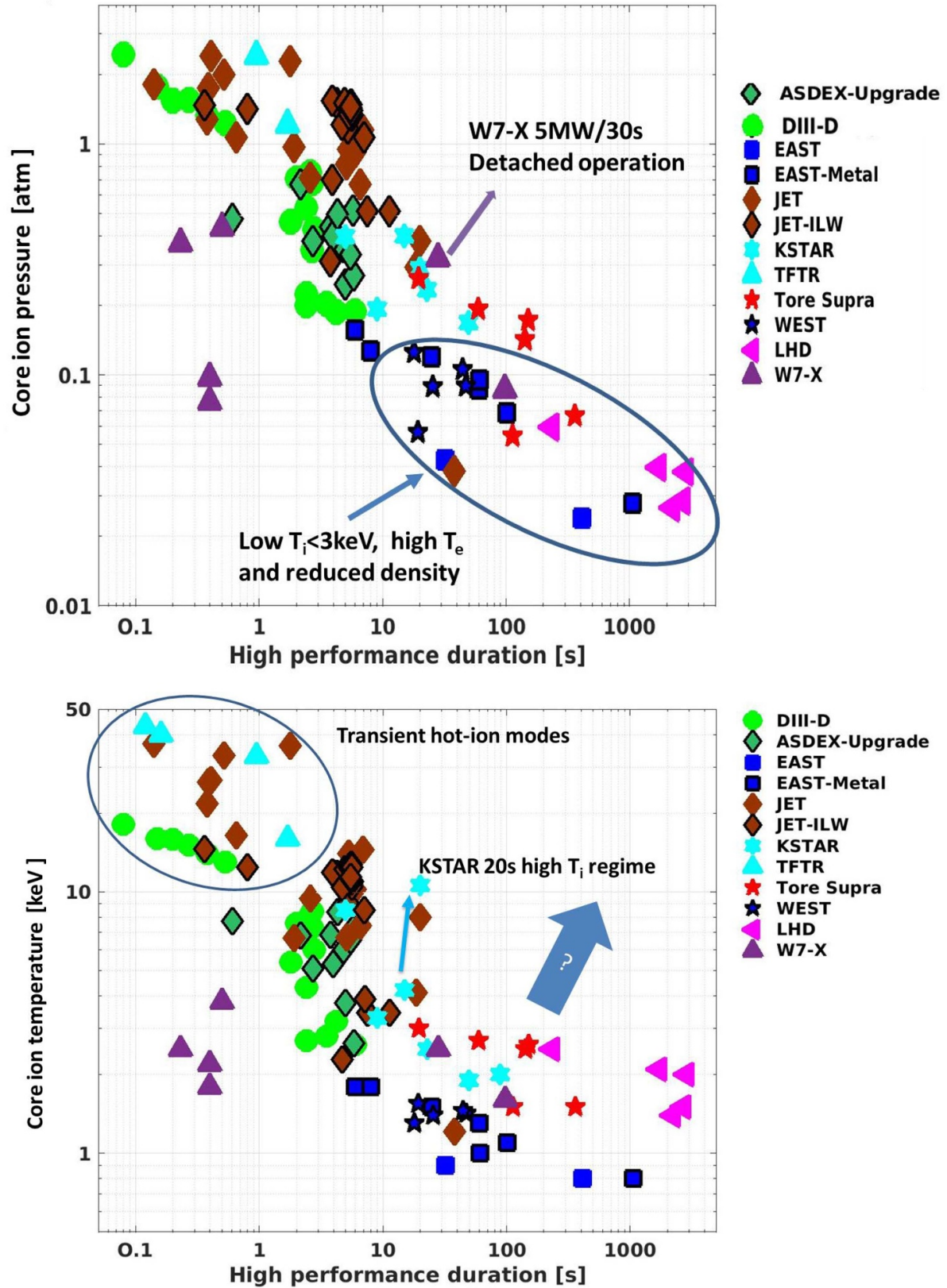
(A)



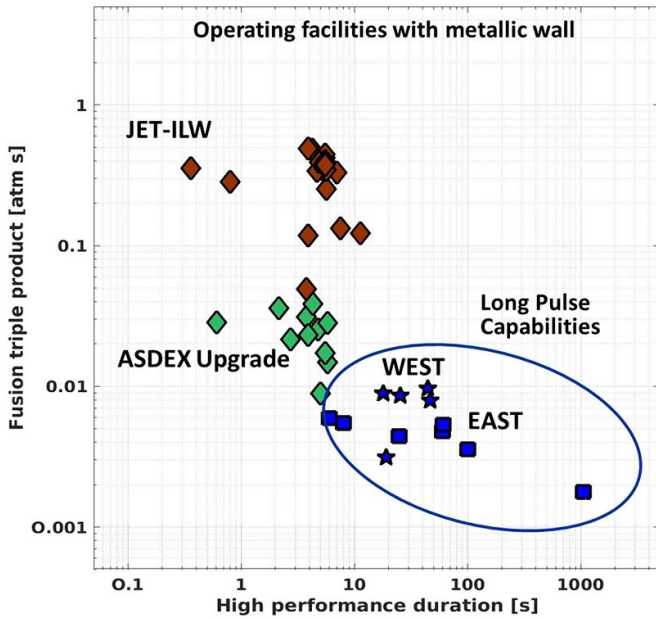
(B)

**Figure 6.** (A) The fusion triple product,  $nT\tau_E$ , in atm s versus the duration of the high-performance phase [s]. Experiments performed with a metallic wall (ASDEX Upgrade, EAST-Metal, JET-ILW, and WEST) have symbols with a black contour line. The horizontal dashed blue line corresponds to the fusion triple product of 0.7 atm s, which is indication of the minimum value to get  $Q = 1$  at the optimum ion temperature, as described in [65]. None of these experiments are at this optimum value. The energy confinement time  $\tau_E$  is in the following ranges for each facility: ASDEX Upgrade: 0.04–0.08 s; DIID-D: 0.07–0.5 s; EAST: 0.04–0.1 s; JET: 0.2–1.1 s; JT-60 U: 0.2–1.1 s; KSTAR: 0.06–0.14 s; TFTR: 0.13–0.3 s; Tore Supra and WEST: 0.04–0.09 s; LHD: 0.13–0.17 s; W7-X: 0.1–0.2 s; ITER: 2.5–3.4 s. (B) The fusion triple product,  $nT\tau_E$ , in atm s versus the duration of the high-performance phase [s] for fully (or quasi-) non-inductive regimes only.



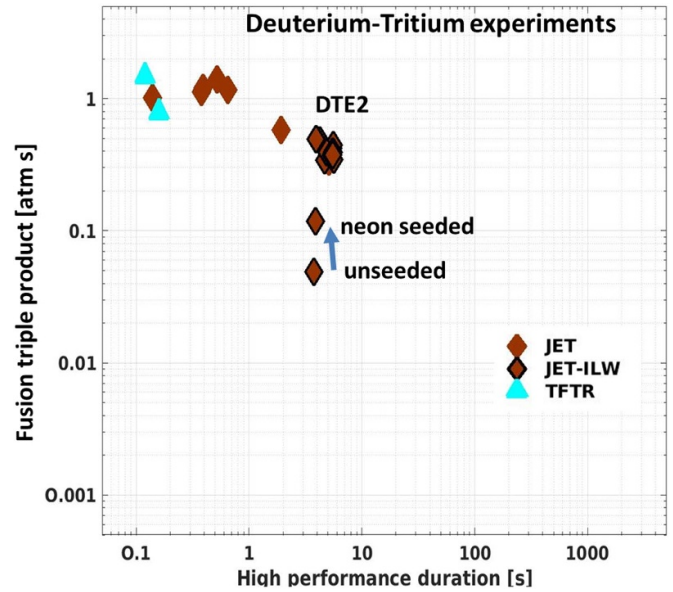


**Figure 7.** (top) Ion plasma pressure,  $n T_{i0}$ , in atm and (bottom) core ion temperature,  $T_{i0}$ , in keV versus the duration of the high-performance phase [s]. Experiments performed with a metallic wall (ASDEX Upgrade, EAST-Metal, JET-ILW, and WEST) have symbols with a black contour line.



**Figure 8.** The fusion triple product,  $nT\tau_E$ , in atm s versus the duration of the high-performance phase [s] for operating machines with a metallic wall/divertor.

mix plasmas obtained in TFTR from 1993 to 1997 [27, 75], JET during the deuterium–tritium experiment 1 (DTE1) with the C-wall/divertor in 1997 [76–79], and DTE2 with the ITER-like wall in 2021 [22, 23, 80, 81]. The TFTR experiments are characterized by high-fusion performance hot-ion modes called ‘super-shot’, where record ion temperatures 20–40 keV (cf. also figure 7) have been reached transiently (less than 1 s) to optimize the fusion power up to 10 MW. These regimes were obtained in a limiter tokamak (i.e. with no divertor), but with intense wall conditioning by injecting lithium pellets into the torus that has resulted in improvements in deuterium–tritium fusion power production [82]. In a transient ELM-free regime, JET during DTE1 has obtained record fusion power (16.1 MW) and high triple product values ( $\sim 1$ – $1.4$  atm s) in a transient manner (less than 1 s) with high ion temperature. ELMy H-mode regimes were sustained during 5 s with a fusion triple product of  $\sim 0.3$  atm s at a level of 4 MW of fusion power but were limited by the heating power duration at a level of 24 MW [77]. Following the installation of the ITER-like wall at JET in 2011, the upgrade of the neutral beam system and diagnostics, significant effort has been made by the JET team to develop stationary and reproducible high-fusion scenarios in deuterium for a duration of 5 s that have transferred in deuterium–tritium plasmas, as shown in figure 9 with values of  $nT\tau_E$  ranging from 0.45 to 0.50 atm s, sustained for a duration of 4.3–5.6 s [22, 23]. Finally, neon-seeded radiative scenarios with (semi-)detached plasmas were tested for the first time in DT plasmas, where the benefit of neon injection in proof-of-principle experiments is shown in figure 9 with the increase in the fusion triple product compared to similar reference pulses



**Figure 9.** The fusion triple product,  $nT\tau_E$ , in atm s versus the duration of the high-performance phase [s] for deuterium–tritium experiments.

but without neon seeding. This integrated ITER-relevant scenario that could be extended up to 10 s with reduced heat flux on the divertor requires optimization in DD and DT and will be further developed during the third DT campaign called DTE3 [83, 84] that is taking place in 2023. To conclude, the sustainment of burning plasmas during LPO will be one of the main and unique scientific objectives of the ITER programme and will fill a major gap in the development of fusion power production.

#### 4. Conclusions

High-fusion performance and LPO are major integration challenges between physics and technology for tokamaks and stellarators on the path towards fusion power-plant development. To address these challenges, a fully integrated vision of the physics and engineering aspects is required to simultaneously increase the time duration and performance of the plasma discharges. As reviewed in this paper and illustrated through the exploitation of a multi-machine 0D database set up by the CICLOP group, significant progress has been made recently towards extending the plasma duration and/or performance. Nevertheless, gaps between the present results and the need to prepare and optimize ITER operation and beyond have been identified by either following the analysis of the multi-machine CICLOP database or highlighted during discussions at the 2022 IAEA Technical Meeting devoted to Long-Pulse Operation of Fusion Devices [14–16 November 2022, IAEA Headquarters, Vienna, Austria, <https://conferences.iaea.org/event/258/>]. The major programme gaps that have been identified are summarized below:

1. LPO is usually performed at reduced power level (below 5 MW), and there is a need to operate at higher power levels in regimes that are compatible with the power handling capacity of the facility. As described in section 2, the power handling capacity of the facility is limited both by the machine/engineering (e.g. limit in energy or temperature for the PFC) and physics limits (e.g. transient events like disruption or ELMs, first-wall erosion).
2. LPO is usually performed in attached divertor conditions, which is not the operating regime for ITER and the reactor. There is a need to sustain a fully detached (or semi-detached) divertor regime for long-pulse duration and address compatibility with the core plasma performance. In addition, the ageing of plasma-facing materials (divertor, first walls, etc) shall be addressed in long-pulse detached regimes, as well as in accidental attached conditions. The long-term impact of the materials' evolution on plasma operation also needs to be assessed.
3. There is a need to further develop operational procedures (e.g. plasma wall conditioning) that are not only valid on a given facility but that can also be transferred to ITER in a safe and reliable manner within the context of nuclear operation.
4. There is a need to further develop and test control methods and plasma performance recovery techniques to maintain the plasma within a stable operational domain (e.g. control in case of loss of divertor detachment that may lead to local overheating of the plasma-facing components). The developed methods should be transferable to ITER and beyond.
5. Finally, LPO should be viewed within the context of operating a nuclear facility with a large level of neutron flux and fluence.

The planned and foreseen upgrades in existing facilities (heating and current drive upgrades, divertor and first-wall modifications), together with the operation of new facilities like JT-60SA and the DTT in support with ITER and reactor design, will provide ideal platforms to address, in a coordinated manner, the scientific challenges required to efficiently address the gaps listed above towards safe and sustained LPO of ITER and fusion powerplants.

## Acknowledgments

The first author would like to acknowledge fruitful scientific exchange with C. Bourdelle and J. Jacquinot during the elaboration of the work. This study has been carried out also with the support and coordination of the IAEA and IEA. The authors gratefully acknowledge the support provided by these as well as by their own organizations to perform this study.

Part of this work has been carried out within the National Key R&D Program of China (Grant No. 2022YFE03050000).

Part of this work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No. 101052200—EUROfusion). Views and opinions expressed are, however, those of the author(s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them.

This work was supported in part by the US DOE under Contracts DE-FC02-04ER54698 and DE-AC52-07NA27344.

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**Annex 1.** The pulse list, associated references, and operation regime per facility.

Facility	Pulse Number	References	Regime of operation	Fully or quasi non-inductive regime (1 = yes, 0 = no)
ASDEX Upgrade	32305	Bock A. <i>et al</i> 2017 <i>Nucl. Fusion</i> <b>57</b> 126 041	Hybrid, advanced scenarios $q_{95} = 5$ with high non-inductive current fraction	1
ASDEX Upgrade	33379	Bock A. <i>et al</i> 2018 <i>Phys. Plasmas</i> <b>25</b> 056115	Advanced scenarios, high $q_{95} = 7$	1
ASDEX Upgrade	36087	Stober J. <i>et al</i> 2020 <i>PPCF</i> <b>62</b> 024012	Counter electron cyclotron current drive (ECCD) $q_{\min} \sim 1.5$	0
ASDEX Upgrade	36663	Burckhart A. <i>et al</i> Experimental Evidence of Magnetic Flux Pumping at ASDEX-Upgrade 28th IAEA FEC 10–15 May 2021 EX-4-1 <a href="https://nucleus.iaea.org/sites/fusionportal/Shared%20Documents/FEC%202020/FEC2020_ConfMat_Online.pdf">https://nucleus.iaea.org/sites/fusionportal/Shared%20Documents/FEC%202020/FEC2020_ConfMat_Online.pdf</a>	Magnetic flux pumping	0
ASDEX Upgrade	37922	Id.	Id.	1
ASDEX Upgrade	38622	Id.	Id.	0
ASDEX Upgrade	38625	Id.	Id.	0
ASDEX Upgrade	38791	Burckhart A. <i>et al</i> , NF 2023 accepted	Id.	1
ASDEX Upgrade	40398	Schramm R. <i>et al</i> , NF 2024 to be submitted	Counter ECCD $q_{\min} \sim 1.5$	0
ASDEX Upgrade	40402	Id.	Id.	0
ASDEX Upgrade	40403	Id.	Id.	0
DIII-D	147634	Holcomb C.T. <i>et al</i> 2014 <i>Nucl. Fusion</i> <b>54</b> 093009	Elevated $q_{\min}$ , double-null shape, high torque	0
DIII-D	155543	Turco F. <i>et al</i> 2015 PoP <b>22</b> 056113	Steady-state hybrid, double-null shape, high torque	1
DIII-D	147044	Jackson G.L. <i>et al</i> 2015 <i>Nucl. Fusion</i> <b>55</b> 023004	ITER baseline scenario, full torque	0
DIII-D	170479	Luce T., APS 2017 contributed	ITER baseline scenario, zero torque	0
DIII-D	171323	Ding S., <i>et al</i> , <i>Nucl. Fusion</i> <b>60</b> (2020) 016023	High rotation Super-H mode	0
DIII-D	174788	Id.	Low rotation Super-H mode	0
DIII-D	161409	Petty C.C. <i>et al</i> 2017 <i>Nucl. Fusion</i> <b>57</b> 116 057	ELM-suppressed steady-state hybrid, single null shape, full torque	1
DIII-D	144903	Solomon W. <i>et al</i> 2013 <i>Nucl. Fusion</i> <b>53</b> 104 019	Zero torque advanced inductive	0
DIII-D	119787	Garofalo TTF talk, 2015	High torque high- $\beta_p$ with internal transport barrier (ITB)	1
DIII-D	154372	Id.	Low torque high- $\beta_p$ with ITB	1
DIII-D	122976	Garofalo A.M. <i>et al</i> 2006 <i>Phys. Plasmas</i> <b>13</b> 056110	High $q_{\min}$ with B- and I-ramps, high torque	1
DIII-D	174791	Snyder P.B. <i>et al</i> , <i>Nucl. Fusion</i> <b>59</b> (2019) 086017	Super-H mode (early phase)	0
DIII-D	174809	Id.	Super-H mode (early phase)	0
DIII-D	171323	Id.	Super-H mode (stationary)	0
DIII-D	87977	Lazarus E.A. <i>et al</i> , <i>Phys. Rev. Lett.</i> <b>77</b> (1996) 2714. Lazarus E.A. <i>et al</i> , <i>Nucl. Fusion</i> <b>37</b> (1997) 7	Negative central shear	0
EAST	43336	Wan B. <i>et al</i> 2013 <i>Nucl. Fusion</i> <b>53</b> 104 006	L-mode LH	1
EAST	41195	Id.	H-mode LH + IC	1

(Continued.)



## Annex 1. (Continued.)

Facility	Pulse Number	References	Regime of operation	Fully or quasi non-inductive regime (1 = yes, 0 = no)
EAST	67341	Gong <i>et al</i> 2017 Plasma Sci. Technol. <b>19</b> 032001	H-mode LH + IC + EC	0
EAST	73999	Wan <i>et al</i> 2019 Nucl. Fusion <b>59</b> 112 003	H-mode LH + IC + EC	1
EAST	70187	Garofalo <i>et al</i> 2018 PPCF <b>60</b> 014043	H-mode LH + EC + IC + NBI	0
EAST	71320	Gao X. <i>et al</i> 2021 Plasma Sci. Technol. <b>23</b> 092001	H-mode LH + NBI	0
EAST	48129	Wan B. <i>et al</i> 2015 Nucl. Fusion <b>55</b> 104 015	H-mode LH + NBI	0
EAST	90949	Huang J. <i>et al</i> 2023 Phys. Plasmas <b>30</b> 062504	H-mode LH + EC: a minute time scale steady-state high $\beta_p$ discharge with dominant electron heating	1
EAST	106915		L-mode LH + EC steady-state high $\beta_p$ with dominant electron heating	1
JET	40847	Gormezano C. <i>et al</i> . FST <b>53</b> (2008) 958	Optimised shear—ITB	0
JET	42733	Gormezano C. <i>et al</i> 1998 PRL <b>80</b> (1998) 5544	ITB + EMLy H-mode	0
JET	42746	Id.	Optimised shear ITB	0
JET	42847	Keilhacker M. <i>et al</i> 1999 Nucl. Fusion <b>39</b> 209	Alpha heating experiment	0
JET	42940	Gormezano C. <i>et al</i> 1998 PRL <b>80</b> 5544	ITB + EMLy H-mode	0
JET	42976	Keilhacker M. <i>et al</i> 1999 Nucl. Fusion <b>39</b> 209	ELM-free H-mode record fusion power DT	0
JET	42982	Horton L.D. <i>et al</i> 1999 Nucl. Fusion <b>39</b> 993	DT H-mode	0
JET	42983	Id.	DT H-mode	0
JET	47413	Gormezano C. <i>et al</i> . PPCF <b>41</b> (1999) B367 and Gormezano C. <i>et al</i> . FST <b>53</b> (2008) 958	Optimised shear—ITB + ELMY H-mode	0
JET	53521	Litaudon X. <i>et al</i> 2002 PPCF <b>44</b> 1057	ITB	1
JET	53697	Litaudon X. <i>et al</i> 2003 Nucl. Fusion NF <b>43</b> 565	ITB	1
JET	56552	Joffrin E. <i>et al</i> 5th IAEA TM on Steady-State Operation, Daejeon, Korea 14–17 May 2007	60 s pulse	1
JET	62065	Joffrin E. Nucl. Fusion 45 (2005) 626–634	20 s H-mode	0
JET	66498	Litaudon X. <i>et al</i> 2007 PPCF <b>49</b> B529	ITB	0
JET	68413	Joffrin E. <i>et al</i> 5th IAEA TM on Steady-State Operation, Daejeon, Korea 14–17 May 2007	20 s hybrid+ ELMy H-mode	0
JET	69093	Litaudon X. <i>et al</i> 2007 PPCF <b>49</b> B529	ITB	0
JET	77922	Joffrin E. <i>et al</i> 23rd IAEA Fusion Energy Conf. IAEA-CN-165/EX/1-1 Daejeon, 11–16 Oct. 2010	Hybrid+ ELMy H-mode	0
JET	85419	Giroud C. <i>et al</i> 2015 PPCF <b>57</b> 035004	Hybrid+ ELMy H-mode	0
JET	96947	Mailloux J. <i>et al</i> 2022 Nucl. Fusion <b>62</b> 042026	Hybrid+ ELMy H-mode	0

(Continued.)

## Annex 1. (Continued.)

Facility	Pulse Number	References	Regime of operation	Fully or quasi non-inductive regime (1 = yes, 0 = no)
JET	96994	id.	Baseline—ELMy H-mode	0
JET	97781	id.	Hybrid + EMLy H-mode	0
JET	99464	Giroud C. <i>et al</i> 29th IAEA Fusion Energy Conference 16–21 October 2023, London, United Kingdom	Reference for 99 621 but without seeding	0
JET	99596	Hobirk J <i>et al.</i> , 2023 Nucl. Fusion Special Issue on JET T & D-T	DTE2 hybrid pulses	0
JET	99621	Giroud C. <i>et al</i> 29th IAEA Fusion Energy Conference 16–21 October 2023, London, United Kingdom	Radiative scenario	0
JET	99869	Hobirk J <i>et al.</i> , 2023 Nucl. Fusion Special Issue on JET T & D-T	DTE2 hybrid pulses	0
JET	99910	id.	Id.	0
JET	99912	id.	Id.	0
JET	99949	id.	Id.	0
JET	99950	id.	Id.	0
JET	99963	M. Maslov <i>et al</i> 2023 Nucl. Fusion Special Issue on JET T & D-T	JET DTE2 hybrid pulses tritium rich	0
JET	99964	id.	Id.	0
JET	99969	id.	Id.	0
JET	99970	id.	Id.	0
JET	99971	id.	Id.	0
JET	99972	id.	Id.	0
JT-60U	26939	Ushigusa K. and the JT-60 Team, Proc. 16th IAEA Conf.F1-CN-64/O1-3 (1996)	Record ion temperature	0
JT-60U	31872	Ishida S. <i>et al</i> 1999 Nucl. Fus. <b>39</b> 1211	Record $Q$ (DT-equivalent)	0
JT-60U	40259	Fujita T. <i>et al</i> 2003 Nucl. Fusion <b>43</b> 1527	Reversed shear	0
JT-60U	21140	Kikuchi M. for JT-60 Team, Proc. 15th IAEA Conf., IAEA-CN-60/A1-2, p31, Vol.1. (1995)	High- $\beta_p$ H-mode	0
JT-60U	21143	Kikuchi M. & Azumi M. 2015 Frontiers in Fusion Research II: Introduction to Modern Tokamak Physics, Springer Int. Publishing	High- $\beta_p$ H-mode	0
JT-60 U	34292	Kamada Y. <i>et al</i> 2001 Nucl. Fusion <b>41</b> 1311	Reversed shear	0
JT-60U	21282	Kamada Y. <i>et al.</i> , Proc. 15th IAEA Conf., IAEA-CN-60/A5-5, p651, Vol.1. (1995)	High- $\beta_p$ H-mode	0
JT-60U	30006	Ishida S. <i>et al</i> 1999 Nucl. Fus. <b>39</b> 1211	High- $\beta_p$ H-mode	0
JT-60U	43046	Ide S. <i>et al</i> 2005 Nucl. Fusion <b>45</b> S48	Reversed shear quasi non-inductive current drive	1
JT-60U	29941	Ishida S. <i>et al</i> 1999 Nucl. Fus. <b>39</b> 1211	High- $\beta_p$ H-mode	0
JT-60U	48158	Oyama N. <i>et al</i> 2009 Nucl. Fusion <b>49</b> 104007	High- $\beta_p$ H-mode	0
KSTAR	16497	Chung J. <i>et al</i> 2018 Nucl. Fusion <b>58</b> 016019	ITB	0
KSTAR	17209		MA shot	0
KSTAR	18306		Long pulse	1

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## Annex 1. (Continued.)

Facility	Pulse Number	References	Regime of operation	Fully or quasi non-inductive regime (1 = yes, 0 = no)
KSTAR	21631	Chung J. <i>et al</i> 2021 Nucl. Fusion <b>61</b> 126051	ITB	0
KSTAR	25860	Han H. <i>et al</i> 2022 Nature 609 269–275	ITB, fully non-inductive	1
KSTAR	27327	Id.	MA shot	0
KSTAR	21735		Long pulse	1
KSTAR	30163		Long pulse	1
KSTAR	30291		Long pulse	1
TCV	18549	Coda S. <i>et al</i> , Plasma Phys. Control. Fusion <b>42</b> (2000) B311	Fully non-inductive, steady-state, ECCD-driven discharge	1
TCV	23585		Fully non-inductive, steady-state, record-duration ECCD-driven discharge sustained by two sets of gyrotrons back-to-back	1
TCV	25645	Coda S. <i>et al</i> , Phys. Plasmas <b>12</b> , 056124 (2005)	Fully non-inductive, steady-state, ECCD-driven eITB discharge	1
TCV	34428	Coda S. <i>et al</i> , Proc. 34th EPS Conf. on Plasma Phys., Warsaw, Poland, Europhys. Conf. Abstr. <b>31F</b> (2007) (D-1.008)	Fully non-inductive, steady-state, bootstrap-driven discharge	1
TFTR	83 546	McGuire K.M. <i>et al</i> , in Proc. 13th International Conf. on Plasma Physics and Controlled Nuclear Fusion Research, Montreal, 1996 (IAEA, Vienna), Vol. 1, p.19.	Supershot, T-NBI	0
TFTR	80539	Hawryluk R.J., Rev. Modern Phys. <b>70</b> (1998) 537	Supershot, highest DT power	0
TFTR	79100	Id.	Supershot	0
TFTR	83207	Id.	Supershot	0
Tore Supra	19980	Martin G. <i>et al</i> . Fus. Eng. Des. <b>51–52</b> (2000) 1007	Limiter	0
Tore Supra	25419	Id.	Limiter	0
Tore Supra	30414	Jacquinot J. <i>et al</i> . NF 43 (2003) 1583	Limiter	0
Tore Supra	32299	Van Houtte D. <i>et al</i> 2004 Nucl. Fusion <b>44</b> L11	Limiter	1
Tore Supra	33898	Jacquinot J. <i>et al</i> . Nucl. Fusion <b>45</b> (2005) S118	Limiter	0
Tore Supra	34181	Id.	Limiter	0
Tore Supra	46569	Dumont R. <i>et al</i> . PPCF <b>56</b> (2014) 075020	Limiter	0
Tore Supra	47979	Id.	Limiter	0
WEST	54178	Yang X. <i>et al.</i> , Nucl. Fusion <b>60</b> (2020) 086012	32 s L-mode	0
WEST	55787	Loarer T. <i>et al</i> . Nucl. Fusion <b>60</b> (2020) 126 046	53 s L-mode	1
WEST	55789	Bucalossi J. <i>et al</i> . Nucl. Fusion <b>62</b> (2022) 042007	50 s L-mode	1
WEST	55953	Tsitrone E. <i>et al</i> . Nucl. Fusion <b>62</b> (2022) 076028	He L-mode	0
WEST	56923	Bodner G. <i>et al</i> . Nucl. Fusion <b>62</b> (2022) 086020	Impurity powder dropper	0
LHD	124617	<a href="https://doi.org/10.57451/lhd.analyzed-data">https://doi.org/10.57451/lhd.analyzed-data</a>	Steady-state high-performance IC + ECH	1
LHD	124612	Id.	Id.	1

(Continued.)

**Annex 1.** (Continued.)

Facility	Pulse Number	References	Regime of operation	Fully or quasi non-inductive regime (1 = yes, 0 = no)
LHD	124530	Id.	Id.	1
LHD	124576	Id.	Id.	1
LHD	124579	Id.	Id.	1
W7-X	20160308.008	Pedersen T.S. <i>et al.</i> , PoP <b>24</b> , 055503 (2017)	Limiter, ECRH, 2.7 MW	1
W7-X	20171 109.045	Jakubowski M. <i>et al.</i> , Fusion Energy Conference (2018)	Test Div., ECRH, 3 MW	1
W7-X	20171 207.006	Wolf R. <i>et al.</i> , PoP <b>26</b> , 082504 (2019), Pedersen T.S. <i>et al.</i> , PPCF <b>61</b> , 014035 (2019)	Pellet, ECRH	1
W7-X	20180918.045	Beidler C. Nature <b>596</b> (2021) 221	Pellet, ECRH	1
W7-X	20181016.016	Jakubowski M. <i>et al</i> 2021 Nucl. Fusion <b>61</b> 106003	Detached, 5 MW, 30 s	1
W7-X	20181017.019	Wolf R. <i>et al.</i> , PoP <b>26</b> , 082504 (2019)	2 MW, 100s	1

Rem:

- 1) The pulses and the data from JT-60U and TFTR are from Kikuchi M. and Azumi M. 2015 *Frontiers in Fusion Research II: Introduction to Modern Tokamak Physics*, Springer International Publishing.
- 2) The database of LHD is completely open and the information on the selected discharges are available from the following link <https://doi.org/10.57451/lhd.analyzed-data>.
- 3) the whole CICLOP database is accessible through an open IAEA web page <https://nucleus.iaea.org/sites/fusionportal/ciclop/SitePages/CICLOP-DB.aspx>



**Annex 2.** Definitions of the CICLOP database variables. The list of data shaded in grey to be provided by the facility contact person is optional.

Provider	Name of the data provider
Date when data provided	Provide the date when the data has been provided in the CICLOP database.
Pulse number	
Date of the pulse	Provide the date when the pulse has been performed.
Reference	Provide a publication or reference to conference where the pulse/regime has been described.
Regime of operation	Describe the regime of operation—precise if it is a fully non-inductive regime.
H-mode or L-mode	L-mode edge = 0; H-mode edge = 1
Main species (H, D, TT, DT, He...)	H-H= 1, D-D = 2, T-T = 3, He-He = 4; H-He = 104, D-T= 103
Injected seeded impurities (argon, neon, nitrogen)	
First-wall material	1 = C, 2 = Mo, 3 = W, 4 = Be, 5 = stainless steel
Divertor material	Id.
Actively cooled divertor or first wall	Precise if operation is performed with actively cooled components.
Max. injected power [MW]	Sum of the total injected power without removing any losses (without removing NBI shine-through) but including the ohmic power.
Injected energy (MJ)	Time integral of the injected power.
Plasma duration (s)	From the plasma initiation to the end with a threshold in plasma current (current above 0.05 MA).
Duration of the high-performance phase (s), durationHP	Duration for providing the time-averaged fusion performance (triple product, beta, etc). Different durations can be provided for a given pulse to get the fusion performance on, for instance, one, two, or three confinement time, one resistive time, etc. The following normalization is proposed: the duration of the high-power phase is calculated as the duration for which the neutron yield is above 80% of the maximum neutron yield. Instead of neutron yield, normalized beta could be used, as for the DIII-D data.
Time-averaged fusion triple product ( $s \times 10^{20} \text{ m}^{-3} \times \text{keV}$ )	Time-averaged fusion triple product on the duration of the high-performance phase, durationHP, defined as the product of core ion density, ion temperature, and confinement time.
ni(0) ( $10^{20} \text{ m}^{-3}$ )	The core ion density time averaged during the high-performance phase.
Ti(0) (keV)	The core ion temperature time averaged during the high-performance phase. The ion temperature is usually measured by charge exchange spectroscopy diagnostics. For WEST, the core $T_i$ is deduced from the reconstructed neutron yield.
tau_E (s)	Energy confinement time averaged during the high-performance phase.
Plasma current (MA)	Plasma current time averaged during the high-performance phase time averaged on durationHP.
Toroidal field on axis (T)	Bo toroidal field on magnetic axis time averaged during high-performance phase.
q <sub>95</sub>	Safety factor at 95% of the poloidal flux deduced from magnetic reconstruction time averaged during the high-performance phase.
Major radius (m)	Time averaged during the high-performance phase.
Minor radius (m)	Time averaged during the high-performance phase.
Plasma elongation radius	Time averaged during the high-performance phase.
Plasma triangularity radius	Time averaged during the high-performance phase.
Poloidal beta	$\beta_p = 2\mu_0 \langle p \rangle / B_p^2$ where $\langle p \rangle$ is the volume-averaged total (thermal and non-thermal) plasma pressure, and $B_p$ is the averaged poloidal magnetic field on the last closed magnetic flux surface. Time averaged on durationHP.
Toroidal beta	$\beta_t = 2\mu_0 \langle p \rangle / B_o^2$ where $\langle p \rangle$ is the volume-averaged total (thermal and non-thermal) plasma pressure, and $B_o$ is the toroidal field on the axis. Time averaged on durationHP.
Normalized toroidal beta	$\beta_N = \beta_t \cdot (I_p/aB_o)^{-1}$ time averaged on durationHP.
Volume-averaged current resistive (s)	Assuming a neoclassical resistive time calculated with volume-averaged density, temperature, $z_{\text{eff}}$ , $a/2$ , major radius, and average $q = (q_{\text{axis}} + q_{95})/2$ . Time averaged on durationHP.

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