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Required R&D in existing fusion facilities to support the new baseline ITER Research Plan

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Abstract

The report provides the description of a selected set of issues for which R&D in present experiments is required to support the refinement or consolidation of the New Baseline ITER Research Plan (IRP) [Loarte 2025]. The selected set includes issues related to completion of systems' designs, specific choices and options to be explored in the early stages of the IRP and strategic assumptions on the development of the IRP experimental programme towards high Q operation.

Keywords: ITER Research Plan (IRP); SRO (Start of Research Operation); DT-1; DT-2; R&D; Disruption mitigation; Disruption Mitigation System; Boronization; H-mode operation; Stationary and power load control; ITER scenario development;Edge Localized Mode (ELM); Electron Cyclotron Heating (ECH); Ion Cyclotron Heating (ICH); Diagnostics.

1. Introduction

Following the public release of the new baseline ITER Research Plan (IRP)) [Loarte 2025], the ITER Organization (IO) has identified a series of issues for which R&D is required to support the refinement or consolidation of the IRP. The issues identified cover a wide range of areas, including:

- R&D for design completion (in particular for the Disruption Mitigation System (DMS), Diagnostics, Boronization, etc.);
- Disruption characterization, prediction and avoidance;
- Stationary H-mode plasmas, ELMs, ELM control and impact on H-mode and power fluxes;
- Characterization and control of stationary power fluxes;
- Plasma material/component interactions and consequences for ITER operation;
- Start-up, ohmic and L-mode scenario development;
- Conditioning, fuel inventory control;
- Basic scenario control and commissioning of control systems;
- Transient phases of scenarios and control;
- Complex scenario control during stationary phases;
- Validation of scenario modelling and analysis tools;
- Heating and Current Drive and fast particle physics;
- Long pulse/enhanced confinement scenario issues.

The issues are grouped into three categories:

- Category 1. The outcome of R&D can have major impact on system design or on the IRP (e.g. modifying the overall experimental strategy in each phase or the objectives of the phases themselves);
- Category 2. The outcome of R&D is expected to have medium impact on system design or on the IRP (e.g. modifying significant details of the experimental strategy to achieve objectives in each phase);
- Category 3. The outcome of R&D is expected to optimize details of the IRP experimental strategy to achieve objectives in each phase by providing relevant experience.

From these areas, the IO has proposed a set of high priority Category 1 and 2 issues to focus the on-going R&D programmes at experimental facilities in the next 3 years given their impact in the IRP. The selected set includes issues related to completion of systems' designs, specific choices and options to be explored in the early stages of the IRP, and strategic assumptions on the development of the IRP experimental programme towards high Q operation, including:

- Support of DMS baseline design (Shattered Pellet Injection);

- Resolution of diagnostic design issues;

- Boronization-related issues: conditioning, T retention and removal, dust production, etc.;

- ECH + ICH heated D H-modes for operation in SRO;

- Electron and Ion Cyclotron Wall Conditioning (ECWC and ICWC) for use in SRO for wall conditioning and fuel removal demonstration;

- ECH-assisted, ICH-assisted and ohmic start-up for SRO;

- Disruption loads characterization in SRO (including runaway electrons on Tungsten plasma facing components) ;

- Strategy for ELM control;

- n = 1 and n = 2 error fields and correction;

- Tungsten wall and divertor sources and SOL transport;

- Tungsten transport in edge-pedestal and core plasma including the effect of ELM control;

- Divertor lifetime appropriateness to allow operation up to, at least, end of DT-1 with the first tungsten divertor;

- 3-D field ELM suppressed H-mode as integrated scenario for ITER SRO and high Q scenarios;

- No-ELM/small ELM regimes as integrated scenario for ITER high Q scenarios (and potential demonstration in SRO);

- $H_{98} > 1$ H-mode scenarios with potential to provide $Q \geq 10$ with $I_p < 15$ MA in ITER;

- Specific issues for $Q \geq 5$ steady-state scenarios in ITER with NBI + ECH;

Table 1 below includes all R&D issues identified for the IRP refinement and consolidation (Categories 1 to 3).

Acronyms

AC	Alternating Current
AE	Alfvén Eigenmodes
AR	Argon
B	Boron
CQ	Current Quench
CS	Central Solenoid
CX	Charge Exchange
CXRS	Charge Exchange Recombination Spectroscopy
D	Deuterium
DMS	Disruption Mitigation System
D/T	Ratio of Deuterium to Tritium
DT	Deuterium-Tritium
DT-1	First DT phase of IRP
DT-2	Second DT phase of IRP
EC	Electron Cyclotron
ECCD	Electron Cyclotron Current Drive
ECH	Electron Cyclotron Heating
ECRH	Electron Cyclotron Resonance Heating
ECWC	Electron Cyclotron Wall Conditioning
EL	Equatorial Launcher
ELM	Edge Localized Mode
ELMy (H-mode)	High confinement plasma (H-mode) with repetitive ELMs
FILD	Fast Ion Loss Detector
FW	First Wall
GDC	Glow Discharge Cleaning
H&CD	Heating & Current Drive
H/D/T	Hydrogen/Deuterium/Tritium
He	Helium
HFS	High Field Side
H-mode	High confinement mode
H_{98}	Energy confinement normalized to ITER-98(y,2) scaling law
ICCD	Ion Cyclotron Current Drive
ICE	Ion Cyclotron Emission
ICH	Ion Cyclotron Heating
ICRF	Ion Cyclotron Range of Frequency
ICRH	Ion Cyclotron Resonance Heating
ICWC	Ion Cyclotron Wall Conditioning
IMAS	Integrated Modelling and Analysis Suite
IR	Infrared
IRP	ITER Research Plan
LFS	Low Field Side
LIBS	Laser Induced Breakdown Spectroscopy
LIDS-QMS	Laser Induced Desorption-Quadrupole Mass Spectrometry
L-mode	Low confinement mode

MHD	Magnetohydrodynamics
NBI	Neutral Beam Injection
Ne	Neon
NIST	National Institute of Standards and Technology
NTM	Neoclassical Tearing Modes
NTV	Neoclassical Toroidal Viscosity
PCS	Plasma Control System
PCSSP	Plasma Control System Simulation Platform
PF	Poloidal Field
PFC	Plasma Facing Component
Q	Fusion power gain
Q	Safety factor
RE	Runaway Electrons
RF	Radio-Frequency
RMP	Resonant Magnetic Perturbation
RWM	Resistive Wall Mode
SBI	Solid Boron Injector
SPI	Shattered Pellet Injection
SRO	Start of Research Operation
T	Tritium
TF	Toroidal Field
TQ	Thermal Quench
UP	Upper Port
UL	Upper Launcher
VDE	Vertical Displacement Event
VS	Vertical Stabilisation
VUV	Vacuum Ultra Violet
W	Tungsten
Z	Nuclear charge
3-D	Three dimensional

2. Table

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
A. R&D for design completion						
A.1	SPI-single injector. Pellet injection optimization for RE avoidance (incl. TQ and CQ mitigation)	Optimum pellet size, injection direction (HFS-like injection from UP), composition (Ne/H fraction for plasmoid suppression)	1	Range of tokamaks with different sizes and plasma parameters to allow extrapolation to ITER and with appropriate measurement capabilities. Requires modelling effort to assess beneficial effects of different injection geometries	Goal of the injection to maximize the electron density potentially compromised by plasmoid drift and rocket effects. R&D will support decisions to finalize the DMS design, in particular whether upper port injection can be used to suppress the drifts. Requires modelling support. R&D will help to decide on optimum pellet diameters for different target plasmas. In addition, R&D will assess the impact of pre-existing instabilities and pre-SPI impurities (effect of seeding). Post-TQ injection and the impact density transport during the CQ must also be studied. R&D should address optimum pellet composition for relaxed thermal energy dissipation for W-first wall. More details on DMS physics basis and remaining issues (https://user.iter.org/?uid=6APD5G)	From SRO onwards
A.2	SPI-single injector demonstration for runaway mitigation	Access to RE benign termination in ITER, decision on whether to abandon high-Z injection scheme for RE energy dissipation and explore alternative mitigation techniques	1	Range of tokamaks with different sizes and plasma parameters to allow extrapolation to ITER and with appropriate measurement capabilities	Study the injection conditions to access RE benign termination through the recombination of the companion plasma with scaling laws and modelling. Identify boundaries for RE benign termination mitigation (e.g. pressure limit). Study the compatibility of the benign termination with RE avoidance conditions (A1). This can impact the decision on the pellet diameters for the DMS design. More details on DMS physics basis and remaining issues (https://user.iter.org/?uid=6APD5G)	From SRO onwards
A.3	SPI-multiple injections	Determination of effectiveness of multiple injections from same location to achieve RE avoidance with optimum TQ, CQ (incl. wall loads) compared to single injections (incl. timing requirements).	1	Range of tokamaks with different sizes and plasma parameters to allow extrapolation to ITER with at least two injectors from the same/similar locations (toroidal separation not required) and with appropriate measurement capabilities	Tokamak with dual injection capability can contribute Sensitivity of jitter requirement for large range of target plasmas to be tested	SRO
A.4	DMS – alternative injections techniques	Demonstration of the feasibility of the technique to inject material in a tokamak and comparison of mitigation efficiency with SPI	1	Single tokamak demonstration and with appropriate measurement capabilities	Risk mitigation R&D in case reference injection technique cannot meet mitigation requirements	From SRO onwards
A.5	DMS – alternative disruption mitigation strategies	Exploration of disruption mitigation by schemes other than massive injection of D_2 and high Z impurities	1	Single tokamak demonstration and with appropriate measurement capabilities	Risk mitigation R&D in case reference mitigation approach cannot meet mitigation requirements	From SRO onwards

A.6	Laser Induced Desorption for in-situ T retention measurement	Demonstrate LID-QMS as quantitative in-situ diagnostic measurement for T retention in B co-deposits at divertor	1	Demonstration in tokamak with W environment and boronization	Required to provide in-situ measurements of T retained in divertor Boron co-deposits (most likely after each operational day or at each two-week maintenance period)	For tests in SRO and use in DT-1
A.7	Single crystal mirror testing	Performance of single crystal mirror with/without active cleaning	1	Demonstration in boronized environment	Required for evaluation of performance of ITER diagnostics using plasma facing mirrors. Release of hydrogen from hydrides to be monitored if possible	SRO
A.8	Laser Induced Breakdown Spectroscopy	Demonstrate LIBS as quantitative measurement for T retention in Boron co-deposits on main wall	1	Proof of principle demonstration in boronized W wall tokamak environment	Can provide an in-situ measurement of T retention in the first wall during shutdown by installation in a robotic arm	DT-1
A.9	In-Vessel Lighting System	Demonstrate that ICWC or ECWC can be used as a quasi-stable light source between tokamak plasmas in the presence of toroidal field for vessel illumination	1	Tokamaks with capabilities to produce ICWC or ECWC discharges and visible camera imaging	The In-Vessel Lighting system will provide survey of the plasma facing components between pulses. This study will provide key input to preliminary design, e.g. to assess if illumination can be supplied via existing plasma production systems (ECWC or ICWC)	SRO (captive components) / DT-1
A.10	Charge Exchange Samples	Design active surface of the CX samples	1	Tokamaks with metallic PFCs, boronization and suitable sample exposure capabilities at main chamber	Provides input to diagnostic design/optimization	DT-1
A.11	Passive (divertor) spectroscopy: Identification of which lines best to use for ionization front and divertor Ti measurements (no Be)	Explore data for (strong) spectral lines in the 300- 1200nm range. Mapping lines to species and ionization stage. IO verification/benchmarking against modelling (SOLPS, EIRENE + ADAS)	1	Tokamaks with metallic PFCs and/or boronization and/or extrinsic impurity injection (Ne, Ar, Xe), CXRS and passive visible spectroscopy or similar views	Could include (esp. for Ne, Ar and potentially B) an overview/summary collected from literature as well (not necessarily new experiment when the data already exists but still providing the data). Provide data to IO for an 'experimentally observed lines database' specifically for lines observed in fusion plasmas (relating it to T_e/n_e values)	SRO
A.12	Passive (divertor) spectroscopy: Identification of which lines best to use for W influx/erosion during all phases of the ITER tokamak discharge	Explore data for (strong) spectral lines in the 300- 1200nm range. Mapping lines to species and ionization stage. IO verification/benchmarking against modelling (SOLPS, EIRENE + ADAS)	1	Tokamaks with metallic PFCs and passive visible spectroscopy or similar views	Provide data to IO for an 'experimentally observed lines database' specifically for lines observed in fusion plasmas (relating it to T_e/n_e values)	SRO
A.13	Develop gamma-ray-based alpha particle measurements without Be (e.g. with B or Ne)	Determine reaction rates that would lead to sizeable gamma ray emission in ITER DT plasmas while meeting scenario requirements (e.g. < 1 % Ne or < 0.2 % B)	1	Devices with suitable gamma ray detectors and fast particle populations that can be used to quantify the technique	Provide assessment to IO on the viability of alpha particle measurements by the radial gamma ray camera for DT plasmas in the new baseline	DT-1
A.14	Protection of FW/blankets elements and gaps from ECH power	Assess how to avoid non-absorbed ECH power reaching unsuitable wall areas (inc. possible minor design changes and operational guidelines)	2	Tokamak with installed EC system	Provides input based on experience on how to best protect the first wall and the components behind gaps from non-absorbed ECH power	SRO and DT-1

A.15	In-situ coating with solid boron injection	Explore the viability and efficacy of a solid boron (B) injection (SBI) system in mitigating the risks to Q ≥ 10	2	Tokamaks with metallic PFCs (W preferred) with SBI system	A SBI systems is being considered for ITER. Input is required on the boron mass injection requirements for ITER, deposition and redistribution of coating layers dependent on plasma scenario, duration of coating effects, properties of coating layers and hydrogenic species retention, layer flaking, integrated modelling and model validation	DT-1
A.16	Neutron diagnostics	Demonstration of measurement capabilities for time of flight 14 MeV neutron spectrometer	2	Tokamaks with sufficient 14 MeV production	Provides input to diagnostic design to provide D/T ratio from neutron measurements	DT-1/DT-2
A.17	IR measurement with reflections in metallic environment	Demonstration of reflection-robust IR temperature measurements of plasma facing components	2	Tokamaks with metallic PFCs and suitable IR systems	Provides input to diagnostic design/optimization and data processing to minimize consequences of reflections on PFC surface temperature determination	SRO
A.18	Radiation Tolerant Detectors	Demonstrate of compact long-life detectors for X-ray and VUV	2	X-ray sources combined with neutron and gamma ray sources	Extend the operating capability and availability of these systems	DT-2
A.19	CXRS with W and B: 1/ Nuisance W (and Xe) lines in CXRS measurement 2/ Usefulness of B V line (594nm) for CXRS	Explore data for (strong) spectral lines in the 300- 1200nm range. Mapping lines to species and ionization stage. IO verification/benchmarking against modelling (SOLPS, EIRENE + ADAS)	2	Tokamaks with metallic PFCs and/or boronization and/or extrinsic impurity injection (Ne, Ar, Xe), CXRS and passive visible spectroscopy or similar views	Could include (esp. for Ne, Ar and potentially B) an overview/summary collected from literature as well (not necessarily new experiment when the data already exists but still providing the data. Provide data to IO for an 'experimentally observed lines database' (Like the NIST database, but specifically for lines observed in fusion plasmas (relating it to Te/ne values)	DT-1
A.20	Two Wavelength Thomson scattering	Demonstration of a working 2-wavelength Thomson Scattering system on a high temperature device.	3	Experience in Thomson scattering as well as appropriate facilities such as high electron temperature device and suitable experts	Extend the dynamic range in temperature of a classic Thomson scattering system and enable an auto-calibration procedure	DT-1
A.21	Pressure gauges	ZrC electron emitter deployed and tested for ITER relevant conditions	3	Vacuum facility with magnetic field, nuclear reactors, magnetic confinement devices	Extend pressure range and total run time. Evaluate radiation hardness	SRO
A.22	Demonstrate FILD front end options	Demonstration of compatibility with ITER operation (FILD)	3	Demonstration of scintillator in suitable facility with n / gamma exposure	Required to verify ITER lifetime	DT-2
A.23	Demonstrate quantitative measurements of confined and lost fast ions with ICE or other potential techniques	Demonstrate quantitative inference of fast ion distribution information (both lost and confined) from ICE spectra or other techniques not presently considered in ITER	3	Tokamaks with capabilities to measure ICE and/or develop other techniques for measurements of confined and/or lost fast ions that can be applied to ITER	Demonstrate the ability to make quantitative measurements of confined and/or lost fast ions in ITER using ICE or of other techniques that can potentially be adopted in ITER	DT-1

B. Implementation of the ITER Research Plan						
B.1. Disruption characterization, prediction and avoidance (for mitigation see Section 1)						
B.1.1	Disruption/VDE thermal load characterization	Identify operational range for unmitigated disruptions not causing melting. Develop tools/models for thermal load characterisation based on available diagnostics. Characterization of thermal loads during TQ and CQ (magnitude, time dependence and distribution)	2	Tokamaks with metallic walls to minimize radiation from carbon during CQ	Determines plasma operational range in which unmitigated disruptions do not cause melting to PFCs and contributes to the determination of an operational disruption budget.	SRO
B.1.2	Disruption/VDE mechanical load (current flow) characterization including toroidal rotation	Characterization of halo currents during disruptions and VDEs (magnitude, time dependence and distribution, including rotation)	2	Range of tokamaks with a range of vessel conductivities to determine influence of vessel/PFC current path versus plasma physics	Determines plasma operational range in which unmitigated disruptions do not cause category II forces in ITER and contributes to the determination of an operational disruption budget	SRO
B.1.3	Runaway electron load characterization	Characterization of power deposition to PFCs by runaway plasmas (magnitude, time dependence and distribution, including magnetic to kinetic energy conversion in termination) for unmitigated, benign (low-Z) and high-Z termination.	2	Range of tokamaks that can produce reliable runaway beams, vary their terminations and measure power fluxes	Determines plasma operational range in which unmitigated runaway beams do not cause PFC melting and contributes to the determination of an operational disruption budget	SRO
B.1.4	Quantify RE production in vertically moving plasmas dominated by the avalanche process	Determine the impact on plasma vertical movement on the effective field applied to the plasma during disruptions and its consequences for RE avalanche amplification	2	Range of tokamaks that can produce reliable runaway beams dominated by avalanche processes with static plasmas and that can control the speed of their vertical plasma movement	Determine whether the intrinsic vertical movement of disrupting plasmas in ITER will lead to significant reductions in the maximum RE current that can be generated in disruptions	SRO and DT-1
B.1.5	Disruption detection	Development of disruption detection schemes that are portable across tokamaks	2	Range of tokamaks performing systematic experiments to emulate ITER-like disruptions and to demonstrate routine application of detection scheme	Reliable detection schemes are essential for the practical implementation of disruption mitigation (TQ, if possible, if not at least for CQ mitigation and RE avoidance)	SRO
B.1.6	Disruption prediction	Development of disruption predictors that are portable across tokamaks and require minimum re-training	2	Range of tokamaks performing systematic experiments to emulate ITER-like disruptions and to demonstrate routine prediction of disruptions	Reliable predictors are essential for the practical implementation of disruption mitigation	SRO
B.1.7	Disruption avoidance	Development of active operational schemes to avoid disruptions in ITER	2	Range of tokamaks performing systematic experiments to emulate ITER-like plasmas with ITER-like actuators	This involves schemes such as: - to recover plasma thermal stability in high radiative fraction conditions - Pre-emptive application of localized H&CD to prevent growth of MHD that eventually trigger disruptions.	All campaigns

B.1.8	SPI-multiple injections	Assessment of viability of staggered injection scheme.	2	Range of tokamaks with different sizes and plasma parameters to allow extrapolation to ITER with at least two injectors from the same/similar locations (toroidal separation not required) and with appropriate measurement capabilities	Robustness of staggered injection scheme across different target plasmas included seeded plasmas. Investigate large density losses during pre-TQ and CQ posing threat for effectiveness of RE avoidance	SRO
B.1.9	SPI-multiple injectors	Determination of effectiveness of multiple injection from different spatial locations to achieve RE avoidance with optimum TQ, CQ (incl. wall loads)	2	Range of tokamaks with different sizes and plasma parameters to allow extrapolation to ITER with, at least, two injectors toroidally well separated and with appropriate measurement capabilities	Sensitivity of jitter requirement for large range of target plasmas to be tested.	SRO
B.1.10	ITER runaway beam scenario	Develop strategy for RE beam formation for DMS testing during SRO	2	Any tokamak with capability to test RE beam scenarios provoked through high-Z driven disruption	Needed for testing DMS efficacy for RE dissipation during SRO	SRO
B.2. Stationary H-mode plasmas, ELMs, ELM control and impact on H-mode and power fluxes						
B.2.1	Characterization of H-mode impurity transport in ITER-like low grad-n pedestal plasmas	Determine whether impurity density profiles become hollow in the pedestal for conditions in which temperature screening is dominant as expected in ITER pedestal plasmas	1	Tokamaks that can obtain high spatial/time resolution pedestal measurements and vary the relative gradients of n and T in the pedestal	To determine the structure of the pedestal impurity density impurity profiles in ITER including interactions between low-Z and high-Z impurities, and roles of turbulence and neoclassical transport near the pedestal and separatrix	DT-1, SRO This affects both ITER phases but has largest implications on DT-1 since there will much less freedom to vary n_{sep} by gas puffing due to the need to control divertor power fluxes in when SOL powers and plasma currents will be highest
B.2.2	H-mode confinement with electron heating/low input torque	Evaluate H-mode confinement in ITER-like plasmas and compare with ion heated/high torque input and explore possible optimization	2	Tokamaks with appropriate heating systems to provide the required heating mix and torque input	To refine predictions of expected H-mode confinement in ITER and develop schemes for its optimization (incl. deleterious MHD avoidance/control). Required to improve predictions in presence of ITER-like central/off-axis localized ECCD which impacts q profiles at the core.	SRO, DT-1 (this affects all ITER phases. The largest implications are on DT-1 as it impacts fusion power production and plasma self-heating. Specific issues for SRO are dealt separately below),

B.2.3	H-mode access and confinement of D plasmas with electron heating and low plasma density	Evaluate H-mode access and confinement in ITER-like D plasmas with dominant electron heating	2	Tokamaks with appropriate heating systems to provide the required dominant electron heating and low-density operation in D with low core electron/ion thermal coupling	To refine predictions of expected H-mode access power and confinement with emphasis in the SRO phase, including a range of impurity sources	SRO (DT-1 to a lesser extent) The plasma density in SRO H-mode will be restricted by the available power and in DT-1 at high plasma current values
B.2.4	Effects of ion isotope and impurities on the L-H and H-L transition power threshold	Establish improved prediction of L-H and H-L transition power threshold for various types of ITER scenarios for a range of impurity contents	**2**	Tokamaks with various fuel mix and low-Z/high-Z impurity injection	To understand the discrepancy in the isotope effect on L-H and H-L confinement transition power and the role of impurity on the L-mode edge turbulence and L-H/H-L transition threshold power. It id likely to require to consider small/no-ELM H-modes to separate the confinement dependency near threshold on ELMs	DT-1 (also affects the range of Ip on which SRO DD H-modes can be accessed)
B.2.5	Pedestal parameters in H-mode plasmas with low grad-n/low ν*	Determine limits to pedestal plasma parameters including transport and MHD stability for plasmas with low grad-n/low ν* in H-mode plasmas	2	Tokamaks that can produce a range of density gradients in pedestal by controlling edge neutral sources	To refine H-mode pedestal plasma predictions in ITER and to determine whether pedestal transport and MHD stability will be similar - dissimilar to that in present experiments	SRO, DT-1 This affects all ITER phases but has largest implications on DT-1 since it impacts the maximum pedestal pressure and overall confinement that can be achieved.
B.2.6	Validation of H-mode impurity transport in ITER-like core plasmas	Identify experimental conditions with ITER-like neoclassical/turbulence W core transport \and validate models showing that uncontrolled W accumulation fur to neoclassical transport effects is not possible in these conditions	2	Tokamaks that can obtain ITER-like core plasma conditions in terms of neoclassical and turbulence transport of impurity species	To determine the structure of the core impurity density profiles in ITER including interactions between low-Z and high-Z impurities, and roles of turbulence and neoclassical transport at the core with the goal of validating transport models in these conditions	DT-1/SRO (this affects both ITER phases but has largest implications on DT-1 since the achievement of Q ≥ 10 can be also affected)
B.2.7	ELM control by 3-D fields with low input torque	H-mode plasmas with low input torque at moderate ratios of P_{input}/P_{LH} to quantify effects on the plasma pedestal and core transport	2	Tokamaks with in-vessel ELM control coils that can operate with a wide range of input torques	The normalized torque from ITER H&CD systems is low and this may require special optimization of the applied 3-D fields to avoid excessive slowing down of plasma rotation and mode locking	DT-1 (SRO and DT-2 are also affected but to a lesser degree because plasma current in H-modes is lower allowing a larger degree of 3-D field optimization)
B.2.8	Evaluation of small/no ELM H-mode regimes potential to provide high Q operation in ITER (including H-mode entrance and exit phases)	Establish whether regimes with small/no-ELMs alternative to the suppressed ELMy H-mode can provide requirements for high Q in ITER with relevant physics	2	Tokamaks which can access such regimes preferably with high-Z plasma facing components	This is to determine if the plasma parameters required for high Q operation in ITER as well as intrinsic device features (e.g. H&CD level and mix) are expected to be compatible with small/no-ELM regimes observed in present experiments including	DT-1 (SRO may be used to test such regimes but this is not strictly needed)

		parameters (collisionalities, radiative fractions, etc.)			the transition from L- to H-mode and from H- to L-mode	
B.2.9	Impurity (W) exhaust for ELM control by 3-D fields in stationary H-modes and its optimization	Determine core impurity (W) exhaust by 3-D fields and its optimization with respect to main ion particle transport. Validation of simulations of impurity and core transport with 3-D fields (when necessary, with low-medium Z impurities)	2	Divertor tokamaks equipped with in-vessel ELM control coils that can explore ELM control in a range of H-mode conditions and perform the required impurity and main plasma measurements	Provides basis for impurity exhaust in ELM controlled regimes by 3-D fields in ITER and its possible optimization in SRO for application in DT-1/DT-2	DT-1 (SRO is also impacted but to a lesser extent because the impact of W accumulation on plasma performance is less important, and the risk of disruptions is lower because of the lower plasma current H-modes)
B.2.10	Requirements for ELM control by 3-D fields in stationary H-modes and effects on confinement and its optimization	Determine physics basis for the requirements for ELM control in ITER and quantify consequences for pedestal plasma, core energy and particle transport, fast particles and possible optimization by tuning of 3-D fields to plasma conditions	2	Divertor tokamaks equipped with in-vessel ELM control coils that can explore ELM control in a range of H-mode conditions	Provides basis for the strategy to explore ELM control by 3-D fields in ITER in SRO for application in DT- 1 / DT-2	DT-1 (SRO is also impacted but to a lesser extent because the impact of 3-D fields on plasma confinement has no operational consequences)
B.2.11	Control of ELM divertor power flux by mitigation	Determine relation between ELM divertor power flux/wetted area and degree of ELM mitigation/pedestal plasma parameters by ITER-like ELM control schemes (3-D fields, pellet pacing and vertical plasma oscillations)	2	Tokamaks that can mitigate ELMs with ITER-like schemes and obtain high spatial/time resolution divertor IR measurements	To determine whether ELM mitigation can provide ELM divertor power flux control or only control of the total ELM divertor energy density	DT-1 (SRO is also impacted but to a lesser extent because lower currents will be explored in H-mode)
B.2.12	Impurity (W) exhaust by mitigated ELMs in ITER-like pedestal plasmas	Determine efficiency of core plasma (W) impurity exhaust by mitigated ELMs in ITER-like pedestal plasmas with low grad-n (i.e. dominant neoclassical temperature screening in the pedestal) by ITER-like ELM control schemes (3-D fields, pellet pacing and vertical plasma oscillations)	2	Tokamaks that can mitigate ELMs with ITER-like schemes and achieve ITER-like pedestal conditions (low grad-n) and perform the required impurity measurements	To determine whether ELM mitigation can provide core (W) impurity exhaust for conditions with flat or hollow impurity profiles at the pedestal.	SRO, DT-1

B.2.13	Compatibility of plasma fuelling by peripheral pellet deposition and ELM suppression by 3-D fields including impurity sources/contamination	Optimize pellet injection (while maintaining peripheral deposition) and applied 3-D fields to achieve ELM suppression and avoid ELM triggering following pellet injection and evaluate consequences for pellet fuelling efficiency and impurity sources/plasma contamination	2	Tokamaks with HFS pellet injection that can provide peripheral pellet deposition and in-vessel ELM control coils that can achieve ELM suppression for gas fuelled H-modes	To determine optimization of 3-D fields and pellet injection to provide core plasma fuelling while avoiding triggering ELMs and evaluate consequences for impurity sources/plasma contamination	DT-1 (SRO is also affected but required fuelling rates are lower and thus there is more margin for fuelling and to optimize the 3-D fields because H-modes will only be explored up to ~ 7.5 MA)
B.2.14	Impact of pellet-associated transient fluxes on impurity sources/plasma contamination	Quantification of pellet induces sputtering and impurity contamination of fuelling pellets and ELM-pacing pellet	2	Tokamaks with W PFCs operating H-modes and pellet injection capabilities	To define if ELM-mitigated H-modes by pellet pacing can be a viable way to control ELMs from the W impurity source/plasma contamination point of view at low plasma currents in ITER	SRO (note pellet fuelling and ELM suppression, more relevant to DT-1, dealt with in B2.13 above)
B.2.15	Impact of NTV on plasma rotation	Quantification of the impact on plasma rotation of residual n = 1 and n =2 error fields and of n = 3 and 4 on plasma rotation for ITER-relevant plasmas with low input torque	2	Divertor tokamaks that can operate in H-mode and can apply a range of 3-D fields for various n's and study plasmas with a range of input torques/toroidal rotations	To provide a validated physics-based prediction for the effect of residual n = 1 and n =2 error fields and of n = 3 and 4 applied for ELM control on plasma rotation in ITER-relevant plasmas and identify the way to mitigate their possible deleterious effects	SRO and DT-1
B.2.16	Isotopic effects on impurity transport and impurity control in H-mode	Characterize impurity transport and the schemes for impurity control for various hydrogen isotopes in H-mode plasmas	3	Divertor tokamaks that can operate in H-mode with a range of hydrogen isotopes and perform impurity transport studies over a range of impurity species	To determine if impurity transport and core impurity control is essentially different in H H-modes versus D and DT H-modes (beyond the differences stemming from the different plasma parameters achieved) and the implications for the D and DT operational strategy which will be based on that of SRO H-modes	SRO An attempt to H H-modes will be performed in SRO and compared to D H-modes. This will be used to prepare the control strategy for DT in DT-1
B.2.17	Characterization of H-mode pedestal particle transport versus particle source to establish density pedestal	Determine particle transport physics in H-mode pedestal and role of neutral source over a range of H-mode conditions	3	Tokamaks that can obtain high spatial/time resolution pedestal measurements and vary the edge neutral source	To determine effectiveness of gas and neutral recycling to fuel ITER H-mode plasmas and range of conditions over which pellet fuelling will be required	DT-1/SRO (this affects both ITER phases but has largest implications on DT-1 as higher plasma densities and pedestal temperatures will be explored leading to lower neutral fuelling efficiency)
B.2.18	T and D transport and core DT mix control by peripheral pellet fuelling	Evaluate transport from D and T injected by pellets with ITER-like peripheral deposition and implications for DT mix control	3	Tokamaks with HFS pellet injection that can provide peripheral pellet deposition for simultaneously for two hydrogen isotopes (D and T being optimum)	Required to determine the required fuelling rates by pellet injection to control DT mix and its optimization by separate T and D pellet injection or by mixed DT pellets	DT-1

B.2.19	First wall ELM power fluxes by mitigation	Determine level of ELM power fluxes to the first wall for mitigated ELMs by ITER-like ELM control schemes (3-D fields, pellet pacing and vertical plasma oscillations)	3	Tokamaks that can mitigate ELMs with ITER-like schemes and obtain high spatial/time resolution first wall IR measurements	To determine whether first wall fluxes for mitigated ELMs with acceptable ELM divertor power fluxes will also provide acceptable first wall power fluxes. Eventually these may be reduced by increased wall clearance (either if fluxes are excessive or if the associated W wall erosion and core plasma contamination is excessive)	SRO (this mostly affects the H-mode development strategy in SRO since at higher plasma currents in DT-1 very small/no-ELMs H-modes will be required)
B.2.20	Optimization of plasma shape in ITER to reduce W wall source/plasma contamination in SRO H-modes	Determine how plasma shape needs to be modified to reduce W wall source/core plasma contamination in H-mode and dependency on ELM type	3	Tokamaks equipped with a high Z wall and can produce H-modes with a large range of plasma shapes/wall gaps	To determine to which level varying wall gap is expected to reduce wall source and plasma contamination in initial H-mode operation and what are the critical shape parameters/wall gaps to be achieved to assess the operational range in which these are achievable in ITER	SRO (this affects the H-mode development strategy in SRO since at higher plasma currents in DT-1 no large shape variations are possible in ITER nor Type I ELMs are allowed)
B.2.21	Isotopic effects on H-mode plasma scenarios	Determine H-mode plasma characteristics with a range of hydrogen isotopes in comparable conditions (same β_N, same P_{inp}/P_{L-H}, etc.)	3	Divertor tokamaks (preferably with W divertor) that can operate in H-mode with a range of hydrogen isotopes	To refine predictions of H-mode plasma parameters in the initial phases of DT-1 based on the SRO hydrogenic H-modes to optimize the initial DT-1 programme	DT-1 (The experiments in H and D will be performed in SRO and on this basis the DT-1 operational strategy will be elaborated)
B.3. Characterization and control of stationary power fluxes						
B.3.1	Divertor power flux deposition width in ITER H-mode plasmas	Determine scaling of the power flux deposition width with H-mode parameters to high B_{pol}^{sep}/low edge collisionality/ρ^*, its dependence on isotope (H/D/T) and divertor conditions	2	Tokamaks that can explore H-mode plasmas over a range of parameters (pedestal and divertor) isotopes H/D/T and species and, in particular, reach as high as possible high B_{pol}^{sep}/low edge collisionality/ρ^*, close to ITER values and can determine accurately divertor power fluxes	Determine at which point in the Research Plan control of divertor power fluxes by impurity seeding will be required to remain under the engineering limits and which gain may be expected by increasing divertor density and increasing divertor recycling	DT-1/SRO (Impact on SRO is lower because power levels are lower, and range of I_p in H-modes is lower)
B.3.2	Effect of plasma response on divertor power fluxes with 3-D fields for ELM control and 3-D field optimization	Determine the effect of plasma response to 3-D fields on the spatial structure and magnitude of the toroidally asymmetric divertor power fluxes with 3-D fields for ELM control and optimization to maximize wetted area	2	Tokamaks equipped with in-vessel ELM control coils that can explore ELM control in a range of H-mode conditions and with a range of plasma responses and can perform the required divertor power flux measurements	Required to determine the spatial structure of divertor power fluxes in ITER and to identify the physics basis on which to extrapolate experimental results to ITER taking into account the plasma response expected to be required for ELM control in ITER	DT-1 (Impact on SRO is lower because power levels are lower and plasma densities are lower due to lower range of I_p in H-modes)

B.3.3	Radiative H-modes with Ne, Ar and mixed impurities and impact on H-mode performance	Determine physics basis to maximize divertor radiation in ITER-like plasmas and evaluate consequences for plasma performance in H-mode with ITER-like edge plasmas (pedestal collisionalities and l_q)	2	Tokamaks that can explore radiative H-mode plasmas over a range of parameters (pedestal and divertor) and species and, in particular, reach as high as possible B_{pol}^{sep}/low edge collisionality/ρ*, as close as possible to ITER values. Tokamaks should be equipped with diagnostics to accurately determine divertor power fluxes	Required to determine optimum impurity specie (or impurity mix) for efficient radiative divertor operation and to evaluate consequences of radiative divertor operation on H-mode performance	DT-1 (These will be the reference operating plasmas for divertor power load control in this phase due to higher power levels and I_p. These plasmas will be explored in the SRO phase but may not be essential to meet the objectives depending on divertor W source)
B.3.4	Compatibility of peripheral pellet fuelling with radiative H-modes	Determine possible limitations to radiative divertor operation due to density transients following pellet injection causing radiative collapses and optimize pellet injection and radiative H-mode conditions for integrated operation	2	Tokamaks with HFS pellet injection that can provide peripheral pellet deposition and radiative H-mode plasmas over a range of pedestal parameters	Required to determine possible limitations to H-mode radiative divertor operation caused by core fuelling by pellets in ITER or to the range of pellet sizes that can be used for core fuelling due to compatibility with H-mode radiative divertor operation	DT-1 (These will be the reference operating plasmas for divertor power load control in this phase due to higher power levels and I_p. These plasmas will be explored in the SRO phase but are not essential to meet the objectives)
B.3.5	Effect of 3-D fields for ELM control on divertor power fluxes in radiative H-modes	Determine the effects of 3-D fields on radiative divertor operation in H-modes for a range of plasma conditions and applied 3-D fields with a varying degree of plasma response and optimization of radiative H-mode plasmas and applied 3-D fields	2	Tokamaks equipped with in-vessel ELM control coils that can explore radiative H-modes and ELM control in a range of H-mode conditions and with a range of plasma responses and can perform the required divertor power flux measurements	Require to determine the degree to which radiative divertor operation will be effective in reducing divertor power fluxes across the divertor target in ITER and whether rigid rotation of the 3-D field structure is required to smooth off-separatrix peak fluxes	DT-1 (Impact on SRO is lower because power levels are lower, and plasma densities are lower due to lower range of I_p in H-modes and thus radiative divertor operation with 3-D fields may not be required to meet the objectives in this phase)
B.3.6	Wall power/particle fluxes in ITER H-mode plasmas	Determine physics mechanisms leading to wall power/particle fluxes in H-mode and their dependence on plasma edge/divertor conditions isotope (H/D/T/divertor geometry (vertical vs. horizontal)	3	Tokamaks that can explore H-mode plasmas over a range of parameters (pedestal and divertor) isotopes H/D/T and can determine accurately wall power/particle fluxes	Determines the expected level of stationary interaction of H-mode plasmas with the ITER wall and the resulting stationary power fluxes and W wall erosion.	DT-1/SRO (Impact on SRO linked to the development of H-modes at low current). Note link to B.2.18 and B.2.19

B.4. Plasma-material/component interactions and consequences for ITER operation						
B.4.1	Main chamber W sources	Determine the net erosion of W wall taking into account both sputtering by the plasma (main ion, seeded impurities, neutrals and fast neutrals) and redeposition	1	Tokamaks with W wall that can determine gross erosion and net erosion after a set of well controlled experiments to make quantitative assessment	Required to benchmark models that predict main chamber W sources in ITER	DT-1
B.4.2	W transport from far-SOL into the pedestal	Determine W transport properties in SOL, through experiments and simulation tool-chains with high level of fidelity at least for non-transient conditions	1	Tokamaks with far-SOL flow measurements, possibility of localized impurity source (e.g. W laser blow-off), charge state resolved spectroscopy	Required to validate W transport simulation tool-chains against measurements of present devices, reducing uncertainties in underlying model parameters (e.g. plasma flows, W source, response of core, etc.)	DT-1
B.4.3	Formation of fuzz by He/W interaction and critical fuzz thickness	Exposure of W PFCs to He minority plasmas, measurement of fuzz and impact on plasma operation	2	Tokamaks with W divertor PFCs operating with He minority plasmas (preferably H-modes to include synergies with ELMs/suppressed ELMs/small ELMs)	Required to determine if fuzz is expected to grow on ITER divertor target during DT plasma operation and to which thickness it can grow. It should also be assessed whether such thickness is expected to affect plasma operation or not.	DT-2 (also relevant for DT-1 but divertor fuzz growth expected to be lower due to lower He fluences)
B.4.4	Power fluxes to castellated PFCs	Determine power fluxes to castellated structures in stationary plasmas and during ELMs over a range of conditions and identify dominant physics processes	2	Tokamaks that can expose castellated W structures to H-mode plasmas in a range of conditions and provide the necessary measurements (power fluxes, currents, etc.)	Required to evaluate power fluxes to the ITER divertor and possible melting or W material deterioration due to high surface temperatures near edges	DT-1/SRO (For DT-1 stationary fluxes are most important since no large ELMs are possible while for SRO ELM loads are likely to be the issue)
B.4.5	Tolerable W damage on surface and macrobrush edges for tokamak operation	Experimentally determine the tolerable level of surface damage/edge damage of divertor macrobrushes to affect tokamak operation (from H-mode confinement deterioration to increased disruptivity due to uncontrolled W influxes in stationary conditions or following ELMs)	2	Tokamaks that can expose pre-damaged castellated W structures to H-mode plasmas in a range of conditions and provide the necessary measurements (power fluxes, impurity influxes, etc.)	Required to provide guidance for the tolerable W damage level for high confinement, low disruptivity (due to W influxes) H-mode operation. This may eventually limit the maximum value of the divertor power flux and/or the number of H-mode that can be performed without ELM suppression.	DT-1 or SRO depending on the tolerable W damage level for H-mode operation
B.4.6	W operation above recrystallization and implications for tokamak operation	Determine the consequences for the W divertor material properties of sustained operation above the recrystallization temperature and assess possible synergistic effects with plasma exposure and consequences for tokamak operation	2	Tokamaks or laboratory facilities that can expose W components to plasma power/particle fluxes for sufficient lengths of time to cause significant W recrystallization while controlling component temperature (i.e. by water cooling). Tokamak experiments with water cooled components are preferred because they can also assess consequences for operation.	Required to determine an operational W recrystallization budget in ITER and thus power fluxes levels and exposure times consistent with a give degree of W surface recrystallization found compatible with appropriate tokamak operation	DT-1 (Impact on SRO is lower because power levels are lower and, thus power/particle fluxes, and plasma discharges are shorter)

B.4.7	W surface modification by high plasma fluence exposure and implications for tokamak operation	Determine the modification to W surface by plasma exposure to ITER-like fluences (and power fluxes, if possible) and evaluate the consequences for tokamak operation	2	Tokamaks or laboratory facilities that can expose W components to plasma power/particle fluxes for sufficient lengths of time to achieve ITER-like accumulated fluences and determine changes to W surface. Tokamak experiments are preferred because they can also assess consequences for operation	Required to determine whether there are additional limits to divertor power fluxes in ITER beyond those linked to engineering PFC limits and W recrystallization due to long term plasma exposure modification of the W surface	DT-2 (Impact on SRO is lower because power levels and plasma density are lower, and thus power/particle fluxes, and plasma discharges are shorter. For DT-1 the impact will be lower than DT-2 because of shorted discharges)
B.4.8	Splashing of W under transients	Determine detailed physics mechanisms leading to splashing of W PFCs under transients in tokamak experiments	2	Tokamaks that can perform controlled experiments of W melting by transients and diagnose dynamics of molten material	Required to determine expected damage to W PFCs in ITER under transients that cause melting and thus contribution to the determination of the required degree of transient mitigation	SRO/DT-1 (Impact on SRO can be significant since ITER plasmas can generate transients that melt PFCs starting from relatively low levels of plasma current due to disruptions. For ELMs, it is expected that no melting/splashing will routinely occur in SRO since Ip will be limited. For DT-1 it is assumed that uncontrolled transients will be rare, but plasma energies will be higher so that a few uncontrolled transients could cause significant splashing)
B.4.9	Melt damage and impact on operation	Determine the impact of melt damage magnitude and spatial distribution on tokamak operation (from H-mode confinement deterioration to increased disruptivity due to uncontrolled W influxes in stationary plasmas or following ELMs)	2	Tokamak facilities that can expose pre-damaged W components to plasma discharges in a range of conditions with appropriate diagnostics of the exposed component to assess consequences for operation	Required to determine tolerable level of W PFC melt damage for reliable ITER operation thus contributing to the determination of the required degree of transient mitigation	SRO and DT-1 (Note temporary inertially cooled first wall for SRO)

B.4.10	Dust production	Determine dominant processes for dust production from metallic PFCs with boronization by tokamak operation to provide physics basis for evaluation in ITER	2	Tokamak that can perform experiments to determine contribution to Boron and W formation of plasma operation and transients with the appropriate diagnostics	Required to perform the evaluation of dust production in ITER operational phases for confirmation by experimental measurements in advance of DT-1. This may impose additional transient mitigation requirements beyond those dictated by PFC lifetime considerations.	SRO or DT-1 (Impact on SRO can also be significant since ITER plasmas can potentially generate dust by stationary operations and transients in this phase. Depending on whether dust is produced mostly by transients or stationary operation and on the dependence of dust production on transient magnitude, dust production could be larger in the DT-1 campaigns or in SRO)
B.4.11	Impact of vapour shielding on power flux in ITER transients	Determine reduction of power fluxes to PFCs under large transients due to the formation of vapour shield	2	Tokamak or laboratory facilities that can produce sufficiently energetic transients to drive the formation of vapour shields and can diagnose power fluxes to the exposed PFCs under these conditions	Required to determine expected damage to W PFCs in ITER under transients for which vapour shielding is expected, thus contributing to the determination of the required degree of transient mitigation	SRO or DT-1 (Impact on SRO can also be significant as ITER plasmas can potentially generate transients leading to vapour shield in this phase. Due to the larger plasma thermal energies and transient power fluxes on PFCs vapour shielding, consequences may be larger for DT-1 but because the number of uncontrolled transients is lower in DT-1, as mitigation schemes should operate routinely, it may be the case that the largest impact is on SRO where these schemes are developed)

B.4.12	W divertor erosion under controlled ELMs	Determine the net erosion of W divertor by controlled ELMs taking into account both sputtering by the plasma (main ion and seeded impurities) and redeposition during the ELMs themselves	2	Tokamaks with W divertor that can determine ELM-resolved gross erosion and net erosion after a set of well controlled experiments to make quantitative assessment, as net erosion is not likely to be measurable ELM resolved	Required to determine accumulative effects of ELMs on W divertor erosion lifetime	DT-1 and DT-2 (this impacts all phases but it is more likely to have larger impact in the DT phases if operation with controlled/small ELMs (and not full ELM suppression) is viable in ITER
B.4.13	W wall erosion under controlled ELMs	Determine the net erosion of W wall by controlled ELMs taking into account both sputtering by the plasma (main ion and seeded impurities) and redeposition during the ELMs themselves	3	Tokamaks with W wall that can determine ELM-resolved gross erosion and net erosion after a set of well controlled experiments to make quantitative assessment, as net erosion is not likely to be measurable ELM resolved	Required to determine accumulative effects of ELMs on W wall erosion	DT-1 and DT-2 (this impacts all phases but it is more likely to have larger impact in the DT phases if operation with controlled/small ELMs (and not full ELM suppression) is viable in ITER
B.4.14	ICRF driven W sources	Consolidate ICRF-specific impurity sputtering estimates in ITER-like conditions	3	Tokamaks with W wall and antenna limiters that can determine ICRF-specific impurity production	Comparison of RF sheath calculations with experiment, quantify RF specific impurities vs background impurities, effect of PFC geometry	DT-1
B.5. Start-up, Ohmic, L-mode scenario development						
B.5.1	RF assisted plasma start-up	Perform experiments to optimize RF assisted start-up and to validate models for ITER	2	Tokamaks that can perform ECH and ICH -assisted start-up over a range of conditions (e.g. applied electric field, etc.) and can diagnose the plasma in this initial phase	Required to design experimental strategy to optimize plasma start-up in ITER and minimize experimental time dedicated to it. Make ITER plasma initiation more robust	SRO because these are the first phases in which ECH assisted start-up will be used, while also ICH assisted start may proof to be a robust way to initiate plasmas
B.5.2	Ohmic plasma start-up	Perform experiments to achieve ohmic plasma start up in ITER-like conditions (i.e. electric field) and identify the range of parameters over which this can be most reliably achieved Optimization of experimental strategy to achieve Ohmic start-up by tuning of PF operation. Validation of models to describe plasma transport in initial ohmic phase and tokamak electromagnetic model used for ITER PF optimization	2	Tokamaks that can perform ohmic assisted start-up over a range of conditions (e.g. applied electric field, etc.) and PFCs, and can diagnose the plasma in this initial phase.	Required to determine if ohmic start-up is possible with a W first wall in ITER	SRO

B.5.3	Need for central heating to control W in L-mode	Determine the required level of central heating to avoid W accumulation in L-mode plasmas versus divertor plasma conditions	2	Tokamaks with W PFCs that can vary the level of central heating in L-mode power while spanning a range of divertor conditions	Required to determine the required level of central heating (ECH and ICH) along with the ramp-up and ramp-down of I_p and density both at full-field and half-field.	SRO (DT will be impacted in a similar way as the optimum path in L-mode is expected to be developed in SRO for H plasmas and tuned for DD and DT plasmas in DT-1)
B.5.4	Plasma transport in ramp-up phase	Determine density, temperature and plasma transport in the ramp-up phase of ohmic and L-mode heated plasmas to validate models for ITER	2	Tokamaks that can explore various levels of additional heating and ramp-rates and determine the temperature and density evolution and plasma transport in the ramp-up phase	Required to optimize current profile evolution during the ramp-up phase of ITER scenarios by tuning of density, heating power and current ramp-rate to control W and achieve suitable q profile at the end of the ramp (for long pulse $Q \geq 5$ scenarios) or to reduce flux consumption in ramp-up (for $Q \geq 10$ scenario).	SRO (SRO and DT-1 can also be affected but the degree of optimization required for ITER plasma scenarios is much higher in DT-2)
B.6. Conditioning, boronization, fuel inventory control						
B.6.1	ICWC and ECWC conditioning	Perform ICWC and ECWC conditioning and determine requirements for effective cleaning and reliable plasma start-up including recovery from unmitigated and mitigated disruptions	2	Tokamaks that are equipped with ICRH or ECRH systems and can perform ICWC or ECWC on a routine basis for general machine conditioning and disruption recovery and diagnose the conditioning plasma	Without an effective cleaning technique, the experimental programme could be significantly slowed down (e.g. recovery from disruptions or disruption mitigation)	SRO (Both techniques are available in SRO)
B.6.2	Removal of hydrogen by baking and GDC	Determine effectiveness of ITER pre-operational campaign conditioning cycles based on baking and GDC for a range of glow conditions	2	Tokamaks that can perform similar conditioning cycle as foreseen in ITER including plasma facing materials and baking temperature	Required to optimize the pre-operation plasma conditioning cycle and the of the GDC glow	All phases, as this is the conditioning sequence foreseen before starting of plasma operation in all phases
B.6.3	Efficiency of T removal by plasma operation from Boron co-deposits in routinely unexposed areas	Evaluate the efficiency of fuel removal in divertor Boron co-deposits (by isotopic exchange and/or thermal desorption by plasma heating) not routinely exposed to large plasma particle/power fluxes by location of the plasma separatrix in these locations.	2	Tokamaks with W PFCs using routine boronizations where B co-deposits in remote area can be accessed by the plasma with suitable modifications to the magnetic configuration	Required to refine the T housekeeping strategy in ITER. If effective this could be adopted for routine ITER cleaning pulses (operation with raised strike points on the B co-deposits) avoiding the need of more complex T removal techniques.	DT-1
B.6.4	ICWC fuel removal from Boron co-deposits	Evaluate the efficiency of fuel removal in divertor Boron co-deposits by ICWC	2	Tokamaks that can apply ICWC on Boron co-deposits with a range of thicknesses and can diagnose the resulting outgassing	Required to refine the T housekeeping strategy in ITER, It is required to determine the need and frequency of ICWC to maintain a low level of in-vessel T retention as removal effectiveness can depend on characteristics of Boron co-deposits.	DT-1

B.6.5	Boronization by GDC	Evaluate the need for boronization in ITER, optimum application of boronization, boronization layer lifetime and methods to extend boron layer lifetime	2	Tokamaks with metallic PFCs (W preferred) that can perform GDC boronization	Required to consolidate boronization strategy	SRO
B.6.6	Fuel retention due to boronization	Evaluate where does boron deposit following plasma operation, content (hydrogen, oxygen, etc.) in the boronized layers	2	Tokamaks with metallic PFCs (W preferred) that can perform GDC boronization	Required to refine the T housekeeping strategy in ITER	DT-1
B.6.7	T inventory in boron dust	Evaluate hydrogen content and size distribution of B-rich dust produced in tokamaks	2	Tokamaks with metallic PFCs (W preferred) that can perform GDC boronization	Required to refine the T housekeeping strategy in ITER	DT-1
B.6.8	Deposit stability and T release from B during vessel venting	Experimentally determine the processes occurring when boron layers, co-deposits, and dust are exposed to air/humidity during vessel venting	2	Tokamaks with metallic PFCs (W preferred) that can perform GDC boronization	Boron layers are reported to undergo modifications upon venting, flaking is also possible. The processes need to be understood and the possible release of T from boron layers, deposits and dust, quantified.	DT-1
B.7. Basic scenario control and commissioning of control systems						
B.7.1	Development of criteria for allowable n = 1 and n =2 error fields in ITER (locked mode threshold, increased disruptivity in H-L transitions, etc.)	Development of scaling of critical n =1 and n = 2 error fields for ITER operation	2	Tokamaks that can apply error fields in a controlled way (i.e. by external error field coils) from various locations in the plasma cross section (LFS, HFS, Top/Bottom)	To provide criteria on acceptable deviations of TF, CS. PF coils from their ideal shape and position due to manufacturing and positioning tolerances. To provide criteria for error field correction in ITER by the use of external coils, possibly supported by internal coils.	SRO and DT-1
B.7.2	Noise in dZ/dt and development of means to reduce it	Determine the source of noise in dZ/dt (plasma/hardware) signal used for VS control and develop schemes for its minimization that can be ported to ITER	2	A range of tokamaks to compare source of noise in dZ/dt and determine its origin as well as to demonstrate that noise reduction techniques are robust across devices	Required to minimize AC losses in the superconducting coils driven by reaction of PF coil system on noise in dZ/dt to keep VS stability. Required to reduce heating of the VS in-vessel coils/busbars and, possibly, to reduce thermal fatigue of the divertor due to strike point oscillations.	SRO
B.7.3	Error field identification and correction	Determination of schemes to optimize error field identification and correction on the basis of the as-built tokamak component specifications and for a range of plasma scenarios (with emphasis on simultaneous correction of resonant and non-resonant error fields).	2	Tokamak with ITER-like systems to correct (external coils) and to identify error fields (in-vessel coils) that can explore error field identification and correction for a similar range of plasma conditions to those in ITER	To optimize the strategy for error field identification and correction in the various phases of the research pan	SRO and DT-1

B.7.4	Impact of application of 3-D fields for ELM control on plasma position control	Determine consequence of the application of 3-D fields for ELM control on plasma position control with ITER-like sensors and develop schemes for optimum position control	3	Tokamaks equipped with in-vessel coils for ELM control and magnetic sensors with similar distribution to that of ITER	Required to optimize plasma position control with applied 3-D fields for ELM control	DT-1 (this also affects SRO when 3-D fields are applied for ELM control, but for DT-1 the consequences of a plasma position error are much larger because of the larger plasma energies and power fluxes)
B.7.5	Optimize ICH coupling control in ITER relevant scenarios	Optimize ICH coupling control avoiding excessive W sputtering, developing real time ICH coupling control loop with ITER-like actuators (e.g. separatrix position control) and (normalized) timescales for actuators and plasma portable across tokamaks	3	Tokamaks with ICRH heating and suitable ITER-like actuators that can implement control loop	Required to ensure required ICRH power through time-varying plasma conditions while avoiding excessive W influx	SRO and DT-1 (note link to B.4.14)
B.7.6	Develop real time divertor power flux measurement and control loop	Develop a real time divertor power flux measurement and control loop with ITER actuators (power, gas fuelling, impurity seeding), sensors, and (normalized) timescales for actuators and plasma portable across tokamaks	3	Tokamaks with good divertor power flux diagnostics and other ITER-like sensors that can be used to provide a real time measurement and can implement such control loop	Required to ensure a given level or upper value) of divertor power flux through time-varying plasma conditions	DT-1 (SRO will also be affected but power fluxes are expected to be much lower so that control loop can be tested but is not required in these phases)
B.7.7	Develop density control loop based on gas and pellet fuelling	Develop a real time plasma density control scheme based on gas and pellet fuelling with ITER-like (normalized) timescales for actuators and plasma portable across tokamaks	3	Tokamaks with gas and pellet fuelling systems that can implement such control loop	Required to ensure good density control in stationary and transient phases	SRO as this control loop needs to be developed before DT-1
B.7.8	Optimize NTM control commissioning	Define an experimental strategy to commission NTM control that can be ported across tokamaks and minimizes the number of plasma pulses/conditions required	3	Tokamaks capable to stabilize NTMs with ECRH/ECCD over a range of experimental conditions	Required to minimize experimental time dedicated to NTM control commissioning	SRO and DT-1
B.7.9	Optimize Sawtooth control commissioning	Define an experimental strategy to commission sawtooth control that can be ported across tokamaks by minimizing the number of plasma pulses/conditions required	3	Tokamaks capable to stabilize sawteeth by ECRH/ECCD and ICRH/ICCD over a range of experimental conditions	Required to minimize experimental time dedicated to sawtooth control commissioning	SRO and DT-1

B.7.10	Installation and demonstration of ITER Plasma Control System (PCS) in a tokamak with ITER-like actuators/timescales	Install and demonstrate plasma operation with the ITER PCS in a tokamak by suitable tuning of the actuators to mimic ITER-like operation	3	Tokamaks with ITER-like actuators that can replace their control system by the ITER PCS	Required to optimize testing and refinement of PCS for ITER operation by application to real tokamak operation	SRO since PCS needs to be ready for commission and operation for initial plasma operation
B.8. Transient phases of scenarios and control						
B.8.1	ELM control and W accumulation control in L-H and H-L phases at constant and varying I_p	Demonstrate ITER-like ELM control schemes (3-D fields, pellet pacing) during H-mode access and exit phases with constant and varying I_p	2	Tokamaks with ITER-like ELM control schemes (3-D fields from in-vessel coils and pellet pacing) that can apply them to a range of H-mode scenario access/exit phases. For assessment of W accumulation control, central heating capabilities and W PFCs are required	Required to ensure robust entry to and exit from stationary H-modes while maintaining ELM control and avoiding W accumulation both when entry/exit takes place during the current flat top as well as with evolving I_p/q_{95}	DT-1 (SRO is also affected but consequences of lack of ELM control/W accumulation in these phases are largest for DT due to the larger plasma energies/power/current levels)
B.8.2	Development of integrated H-mode termination scenarios	Demonstrate integrated H-mode scenarios with controlled density evolution, plasma radiation, divertor power fluxes, etc. by actuators available in ITER and with relevant normalized timescales	2	Tokamaks with ITER-like actuators (H&CD, fuelling, impurity seeding, etc.) that can apply them to a range of H-mode scenario termination phases and demonstrate control of required parameters (suitable measurements are required)	Required to ensure robust exit from stationary H-modes while avoiding plasma physics limits (e.g. density limits) and operational limits (e.g. excessive divertor power fluxes)	Both SRO and DT-1 however consequences of lack of ELM control/W accumulation in H-mode termination are largest for DT-1 due to the larger plasma energies/power/current levels)
B.8.3	Dynamic error field correction for transient confinement phases	Develop dynamic error field correction schemes for transient confinement phases (L-H, H-L) portable across tokamaks to mitigate error field effects on confinement and minimize risks of disruptions in these phases	3	Tokamaks equipped with in-vessel coils that can apply time varying error field correction within the timescale of confinement transient phases	Error field correction depends on plasma conditions. Transient confinement phases can be sensitive to error fields particularly in ITER in which plasma rotation is expected to be low	SRO and DT-1 but error field effects are expected to be larger with larger β_N so that a larger level of correction is required in DT-1
B.9. Complex scenario control during stationary phases						
B.9.1	Develop fully integrated real-time ELM control	Demonstrate Type I ELM prevention and avoidance by fully integrated real-time control schemes	2	Tokamaks with capability to control ELMs using RMPs, gas or pellet pacing	This is to demonstrate not the actuator capability but the full real-time control that prevents and avoids ELMs that can be detrimental to the divertor life-time or the tokamak discharge itself	SRO and DT-1
B.9.2	W real-time detection and control schemes for all phases of the ITER tokamak discharge	Determine realistic synthetic diagnostics that can be used to assess the diagnostic capabilities to determine the W concentration in the plasma, and how to use such detection schemes as part of W control schemes	2	Tokamaks with W components and various dedicated diagnostics that detect W.	This can be an analysis task, including on developing the diagnosis side, but should also include the diagnostic as part of a realistic W control loop including actuators, plasma response, etc.	SRO and DT-1

B.9.3	Develop and demonstrate AE control strategies for ITER	Demonstrate schemes for AE stabilization with ITER-like actuators (e.g. ECRH/ECCD) over a range of plasma conditions	3	Tokamaks and stellarators that can produce unstable AE and control their instability with ITER-like actuators (e.g. ECRH/ECCD)	Required to ensure control of AE if these lead to unacceptable fast particle losses and/or redistribution	DT-1
B.9.4	Integrated control of radiative Ne/Ar H-mode in ITER-like conditions	Demonstrate divertor power load/detachment control while maintaining Ne/Ar radiative H-mode operation (with controlled ELMs) at low margin of P_{sep}/P_{LH} with ITER-like actuators and normalized timescales	3	Tokamaks that can operate in radiative Ne/Ar H-modes and control divertor power loads with ITER-like actuators	Required for routine ITER operation in high Q reference scenarios	Both SRO and DT-1 but impact is lower for SRO because power levels are lower and plasma densities are lower due to lower range of I_p in H-modes. Thus, highly radiative divertor operation may not be required to meet the objectives in these phases if W influxes are appropriate
B.9.5	W accumulation control by central H&CD and central MHD control in H-modes	Demonstrate schemes for W accumulation control in H-mode plasmas by central H&CD or by control of central MHD by H&CD with ITER-like actuators and normalized timescales	3	Tokamaks with W PFCs that can explore a range of H-mode conditions and can explore W accumulation control with ITER-like actuators and normalized timescales acting on core transport (heating) or core MHD	Required to demonstrate control capabilities of core W accumulation for fixed edge W exhaust provided expected to be provided by ELM control	All phases (note link with B.9.2)
B.9.6	NTM control algorithms for routine use	Develop tokamak-portable NTM control algorithms and apply them routinely for NTM control in tokamaks	3	Tokamaks with ECRH/ECCD system for NTM control that can run ECRH/ECCD system in feedback mode for NTM control	Required for routine operation of high β_N H-modes plasmas required to achieve ITER's high Q goals	DT-1, however, also SRO when NTM control will be commissioned in this phase so that it can be routinely applied later
B.9.7	Access/exit to/from burn and burn control	Perform experiments in which a significant fraction of heating power is used to simulate alpha heating and demonstrate and assess the control requirements for correct access/exit from burn as well as of stationary burn with ITER-like actuators	3	Tokamaks that can emulate fusion power evolution by additional heating and have ITER-like actuators for control of the plasma. Simulations that assess the stable burn operation space.	Required to ensure robust operation at high Q in ITER. An initial demonstration in existing experiments and control assessments is desirable to refine operational control/strategy on ITER	DT-1
B.9.8	Combined tests of ITER control algorithms (e.g. pellet fuelling + gas fuelling and impurity seeding for fuelling and power flux, etc.)	Perform experiments combining a set of control algorithms and ITER-like actuators and optimize the control loops to provide robust ITER operation near operational limits	3	Tokamaks with appropriate set of ITER-like actuators and control loops that can be tested in a combined way.	ITER operation requires simultaneous operation of many control loops that can require sharing of actuators and can lead to conflicts among loops. Demonstration of the proposed control loops in an integrated way in a tokamak experiment can be used to refine the integration of the various loops in ITER and to identify possible conflicts associated with plasma behaviour that may not be obvious otherwise	SRO and DT-1

B.9.9	Demonstration of low disruptivity operation with ITER-like plasmas and actuators	Demonstrate routine disruption-free tokamak operation with ITER-like actuators and ITER-like plasmas (near operational boundaries) for ITER high Q scenarios	3	Tokamaks with ITER-like actuators and control schemes (with appropriate normalized times) that can operate in H-mode plasma scenarios as required for ITER high Q goals	Robust disruption free operation is required to achieve ITER's high Q goals. Experience needs to be gained in the optimization of ITER actuators to achieve disruption free operation for each high Q scenario as they operate close to different limits depending on the scenario	SRO and DT-1
B.10. Validation of scenario modelling and analysis tools						
B.10.1	Assess ITER operation scenarios using the proposed new auxiliary heating mix (simulation and validation)	Run plasma simulator using new H&CD configuration in each ITER phase and validate with suitable plasma discharges from present experiments	2	Joint modelling activity supported by experimental plasma discharges	Apply new heating and current drive mix to simulate new baseline plasma scenarios SRO: 40 MW EC (1 EL, 3 UL), 10 MW IC DT-1: 60 MW EC (2 EL, 3 UL), 10-20 MW IC, 33 MW NBI	SRO, DT-1
B.10.2	Develop IMAS/PCSSP diagnostic and actuator models for control and for validation of scenario modelling tools	Develop IMAS/PCSSP diagnostic and actuator models that are portable across tokamaks and demonstrate use for control/model validation purposes	3	Tokamaks to provide required diagnostic and actuator design information and to test synthetic diagnostic/actuator in control loop	Synthetic diagnostics/actuators are required for implementation of plasma control in ITER and for validation of modelling predictions	All phases
B.10.3	Develop IMAS/PCSSP workflows to provide parameter measurements on multiple diagnostic input	Develop IMAS/PCSSP workflows to derive measurement for plasma parameters from multiple diagnostic input	3	Tokamaks that provide required multiple diagnostic input for workflow to produce measurement and comparison with evaluation based on single independent diagnostic measurement information	Most key ITER parameters are determined by various simultaneous diagnostic systems. Workflows are required to provide measurements for plasma parameters based on multiple diagnostic input and limitations/ inaccuracies of each individual diagnostic contributing to the measurement	All phases
B.10.4	Develop IMAS/PCSSP plasma reconstruction chains using measurements and uncertainties	Develop tokamak independent IMAS/PCSSP plasma reconstruction chains (e.g. plasma equilibrium), demonstrate routine use in fusion experiment and compare performance with existing reconstruction chains at the existing facility	3	Tokamaks providing required tokamak-specific data for reconstruction chain and demonstration in real-time and post pulse plasma analysis	Plasma reconstruction chains are required to analyse ITER data. It is important to ensure that these tokamak -independent chains are developed and well tested before they are required for ITER	All phases
B.10.5	Improve ITER IMAS scenario modelling capabilities by experimental validation	Apply ITER IMAS plasma scenario simulators to design plasma pulses in existing tokamaks, validate with experiments and refine models in simulators for higher fidelity	3	Tokamaks that can produce operating scenarios over a range of conditions and with ITER-like actuators to compare with ITER simulator predictions	Reliable ITER pulse design requires validated plasma scenario simulators. The models in the simulator will be refined as result of ITER operation but an initial validation against experiment is required before their first application to ITER	All phases
B.10.6	Improve faster than real-time plasma IMAS predictor to predict pulse trajectories and required control actions by experimental benchmark	Apply ITER faster than real time plasma IMAS predictor to plasma pulses and compare predicted pulse trajectory with experimental one and improve predictors for higher fidelity	3	Tokamaks that can produce ITER-like plasma scenarios over a range of conditions to compare predicted pulse trajectories with experimental ones	Anticipation of control actions requires reliable faster than real-time predictor in ITER. The models in the predictor will be refined as result of ITER operation but an initial benchmark against experiment is required before their first application to ITER	All phases

B.11. Heating and Current Drive and fast particle physics						
B.11.1	Advanced ICRH schemes for SRO and DT-1	Demonstrate schemes for ion heating in SRO and DT-1 relevant to their application in ITER in these phases	2	Tokamaks with appropriate levels of ICRH heating and frequency range to demonstrate ion heating schemes in ITER-relevant SRO plasmas and DT-1 plasmas	Dominant ECH heating leads to relatively low ion temperatures in SRO and it also impacts DT-1. Use of ICRH can potentially provide significant ion heating allowing the assessment of high Ti plasmas (note that for D plasmas in SRO this may be restricted to short phases due to neutron fluence restrictions)	SRO and DT-1
B.11.2	Validation of shine-through loads with high energy NBI	Perform experiments with high energy hydrogen NBI (E_{NBI} ~500 keV) to validate models for evaluation of shine-through loads in ITER	2	Tokamaks and stellarators with high energy hydrogen NBI and good diagnostics of shine-through power fluxes on PFCs	Required to accurately determine Hydrogen-NBI H-mode operational space which is limited (in the low density side) by shine-through loads	DT-1 (when NBI is first planned)
B.11.3	Evaluate fast particle losses with 3-D fields and their correlation with plasma response and ELM control	Determine fast particle losses in H-mode plasmas when 3-D fields are applied and identify their correlation with plasma response and ELM suppression. Explore means to minimize fast particle losses while sustaining ELM suppression	2	Tokamaks with significant fast particle densities that can apply 3-D fields from in-vessel coils for ELM control and with diagnostics to measure fast particle losses	Required to identify fast particle losses caused by the application of ELM control by 3-D fields and their possible optimization to reduce losses while sustaining ELM control	DT-1
B.11.4	Alfven Eigenmode stability in H/D and DT plasmas	Determine the stability of AE modes in wide range of plasma conditions and species and measure damping rates	2	Tokamaks with capability to produce fast particle distributions for a range of plasma species and conditions and equipped with means to excite AE and determine their damping and their effect on fast particle confinement	The results of these experiments will be used to validate models for the prediction of AE stability in ITER and their effect on fast particle confinements. Exploring various plasma species allows the identification of possible experiments to be done in SRO in preparation/mitigation of possible problems in DT-1	From SRO (ICRH) but most relevant for DT-1 because of alpha particles
B.11.5	Validation of ITER IMAS H&CD models with self-consistently evaluated plasma parameters	Perform experiments targeted to the validation of ITER Heating and Current Drive IMAS models for NBI, ECRH/ECCD and ICRH with self-consistently evaluated plasma parameters and improve models if required.	3	Tokamaks with ITER-like H&CD that can perform experiments over a large range of parameters to validate Heating and Current Drive deposition profiles with appropriate measurements to validate modelling	ITER IMAS plasma scenario modelling is based on models to describe H&CD profiles with self-consistently evaluated plasma parameters. It is important to ensure that the models applied to ITER plasma scenario modelling can describe accurately present experiments.	All Phases
B.11.6	Control of ICRH minority density in controlled ELM H-modes with peripheral pellet fuelling	Evaluate the controllability of ICRH minority density in H-mode plasmas with ITER-like features regarding fuelling (peripheral fuelling and ELM control)	3	Tokamaks that can perform peripheral HFS pellet fuelling and ELM control in H-mode plasmas while applying ICRH minority heating with variable minority concentrations	Required to determine optimum minority control strategy for ICRH heating in realistic ITER-like conditions regarding H-mode fuelling and ELM control	SRO
B.11.7	Effect of peripheral pellet fuelling and ELM control by 3-D fields on ICRH heating of H-mode plasmas	Determine the effect on ICRH heating for H-mode plasmas (matching, etc.) of peripheral pellet fuelling and ELM control by 3-D fields.	3	Tokamaks that can apply ICRH heating to H-mode plasmas with 3-D fields for ELM control and with peripheral pellet fuelling	Required to assess effects of ELM control by 3-D fields and peripheral pellet fuelling on ICRH heating of H-mode plasmas and to evaluate strategies for its optimization	SRO and DT-1

B.11.8	Validate reconstructions of fast ion distributions for ITER stability analysis/predictions	Perform plasma experiments with significant densities of fast particles and reconstruct their distributions from experimental measurements/modelling	3	Tokamaks that can generate significant fast particle densities and have diagnostics to determine their distribution	Reconstructions of fast ion distributions are required to analyse fast particle MHD stability in TER data. It is important to ensure that these tokamak-independent reconstructions are developed and well tested before they are required for ITER	SRO and DT-1
B.11.9	Develop and benchmark fast particle stability analysis tools including 3-D and kinetic effects	Develop and benchmark stability analysis tools (incl. 3-D and kinetic effects) with specially design experiments	3	Tokamaks that can generate significant fast particle populations, can apply 3-D fields for ELM control and have appropriate diagnostics to determine MHD instabilities driven by them	Fast particle stability analysis tools are required for ITER plasma analysis. It is important to ensure that these tokamak-independent analysis tools are developed and well tested before they are required for ITER	SRO and ST-1
B.12. Specific issues for long pulse/enhanced confinement scenarios						
B.12.1	ELM suppression in high q_{95} / high β_N scenarios and consequences for plasma confinement	Determine externally applied 3-D field structure to achieve ELM suppression in plasma conditions suitable for Q ≥ 5 long pulse operation or Q ≥ 10 at high q_{95} and evaluate consequences for plasma confinement (3-D field structure and confinement effects are expected to be significantly different to those for conventional H-modes both due to the different q_{95} and β_N that affect plasma response)	2	Tokamaks that can apply 3-D fields for ELM with in-vessel coils and explore high q_{95} high β_N scenarios	Required to determine 3-D field structure for ELM control in long pulse Q ≥ 5 scenario or Q ≥ 10 at high q_{95} and to evaluate consequences for plasma confinement	DT-1
B.12.2	Role of fast particles, rotation and pedestal stability on core-edge feedback for beyond-H_{98} confinement	Perform experiments to identify the physics mechanisms leading to enhanced overall energy confinement in hybrid/advanced H-mode plasmas and, in particular, determine: - role of central NBI particle deposition - role of fast particle density - role of plasma rotation - role of MHD pedestal limiting stability (ballooning, peeling ballooning, peeling)	2	Tokamaks which can explore wide operational range of advanced/hybrid H-mode plasmas and vary fast particle content, core fuelling profile, plasma rotation and pedestal collisionality	Required to identify the physics processes leading to enhanced H-mode confinement (required for long pulse Q ≥ 5 or Q ≥ 10 at high q_{95}) and whether this confinement enhancement basis extrapolates to ITER. Namely the purpose is to identify the role of each possible physics mechanism (fast particle effect on plasma beta or on transport, plasma rotation, central fuelling and ion heating, etc.) leads to H > 1 confinement since this impacts the extrapolability of these scenarios to ITER (e.g. low normalized rotation. low core fuelling and low central ion heating)	DT-1
B.12.3	Develop alternative small ELM/no-ELM regimes ITER operation scenarios	Develop no-ELM free or small ELM scenarios with capability for Q > 5 / Q ≥10 operation at ITER	2	Experimental development scenarios with no/small ELMs which can meet all requirements for high Q in ITER	Focus on the realization of such scenarios under realistic ITER high Q conditions for all scenario phases (not just stationary flat-top values which is the topic of B.2.8)	SRO and DT-1. Note link to B.2.8

B.12.4	Optimization of the IRP strategy to stepwise increase performance towards Q≥10, with a focus to achieve this at I_p < 15MA	Use of existing experience and integrated simulations to optimize the ITER DT-1 path towards Q ≥ 10	2	D (and DT when possible) plasma scenarios integrated simulations of high-performance plasmas	Focus on the development path for fully integrated operation scenarios, with improved performance to provide Q ≥ 10 with I_p < 15 MA based on physics understanding that can be extrapolated to ITER (see B.12.2)	DT-1 Note link to B2.12
B.12.5	Design of new scenarios for DT-2	Perform plasma simulations to study feasibility of hybrid and steady-state scenarios with the new H&CD upgrade options and validate with existing experiments	2	Joint modelling and validation activity supported by experimental data	Should consider DT-2 upgrade scenarios and the target to achieve Q ≥ 5 in long pulse/steady state conditions: ▪ From 60 to 67 MW EC ▪ From 10 to 20 MW IC ▪ From 33 to 50 MW NBI	DT-2
B.12.6	Pellet fuelling of high q_{95} / high β_N H-mode plasmas	Determine the effect of high q_{95} / high β_N on the fuelling efficiency of pellet with peripheral deposition for a range of H-mode plasma conditions	3	Tokamaks with HFS pellet injection that can provide peripheral pellet deposition and explore high q_{95} high β_N scenarios	Required to determine if high q_{95}/high β_N can affect pellet fuelling efficiency or ITER pellets due to effects on HFS pellet drift which can depend on q_{95}/rational surfaces crossed by the pellet	DT-1
B.12.7	Optimization of current ramp-up to achieve target q profile for long pulse scenarios by feedback control	Demonstrate the achievement of a range of q profiles at the start of flat top by feedback control with ITER-like actuators in the ramp-up (ramp-rate, density, H&CD, etc.)	3	Tokamaks with sufficiently long ramp-up phase to allow feedback control and with ITER-like actuators that can be applied in the ramp-up phase	Required to minimize experimental development to identify and demonstrate feedback control schemes to achieve a target q profile at the end of the ramp-up in ITER	SRO and DT-1
B.12.8	q profile control feedback in medium/long timescales	Demonstrate schemes to sustain the target q profile by ITER-like actuators over timescales relevant to the ITER scenarios	3	Tokamaks with ITER-like actuators and sufficiently long flat top phase to allow feedback control over relevant timescales	Required to minimize experimental development to demonstrate feedback control schemes for the q profile during Q ≥ 5 long pulse scenarios in ITER.	DT-2
B.12.9	Demonstrate RWM control in combination with ELM control	Demonstrate schemes for simultaneous control of RWMs and ELMs by sharing of in-vessel coil capabilities	3	Tokamaks with in-vessel coils for ELM control and RWM control that can access Type I ELMy H-mode conditions unstable to RWMs to demonstrate scheme	Required for high β_N operation in ITER Q ≥ 5 steady-state plasmas	DT-1 and DT-2

Table 1. List of issues to refine and/or consolidate the IRP requiring R&D in experimental facilities.

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References

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