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Required R&D in Existing Fusion Facilities to Support the ITER Research Plan

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Acronyms

AC	Alternating Current
AE	Alfvén Eigenmodes
CQ	Current Quench
CS	Central Solenoid
DD	Deuterium-Deuterium
DMS	Disruption Mitigation System
D/T	Ratio of Deuterium to Tritium
DT	Deuterium-Tritium
ECCD	Electron Cyclotron Current Drive
ECH	Electron Cyclotron Heating
ECRH	Electron Cyclotron Resonance Heating
ECWC	Electron Cyclotron Wall Conditioning
ELM	Edge Localized Mode
ELMy (H-mode)	High confinement plasma (H-mode) with repetitive ELMs
FILD	Fast Ion Loss Detector
FP	First Plasma
FPO	Fusion Power Operation
GDC	Glow Discharge Cleaning
H&CD	Heating & Current Drive
H/D/T	Hydrogen/Deuterium/Tritium
HFS	High Field Side
ICCD	Ion Cyclotron Current Drive
ICE	Ion Cyclotron Emission
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	Laser Induced Desorption
MHD	Magnetohydrodynamics
NBI	Neutral Beam Injection
NTM	Neoclassical Tearing Modes
PCS	Plasma Control System
PCSSP	Plasma Control System Simulation Platform
PF	Poloidal Field
PFC	Plasma Facing Component
PFPO	Pre-Fusion Power Operation
RE	Runaway Electrons
RF	Radio-Frequency
RWM	Resistive Wall Mode
SPI	Shattered Pellet Injection
TBM	Test Blanket Module
TF	Toroidal Field
TQ	Thermal Quench
VDE	Vertical Displacement Events
VS	Vertical Stabilisation
VUV	Vacuum Ultra Violet

Following the public release of the ITER Research Plan (IRP), 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, 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 both 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, SPI);
- Resolution of diagnostic design issues;
- He H-mode operation and/or H+10% He operation for Pre-Fusion Power Operation (PFPO);
- 3rd harmonic ECH heating validation for 1.8 T operation in PFPO;
- Low $<n_e>$ ECH heated H-modes for operation in PFPO-1;
- Electron Cyclotron Wall Conditioning (ECWC) for use in PFPO-1;
- ECH-assisted and ohmic start-up for First Plasma (FP) and PFPO-1;
- Disruption loads characterization in PFPO;
- Strategy for ELM control;
- n = 1 and n = 2 error fields and correction;

- Divertor lifetime appropriateness to allow operation until well into the Fusion Power Operation (FPO) phase with the first tungsten divertor;
- 3-D field ELM suppressed H-mode as integrated scenario for ITER high Q scenarios;
- Specific issues for Q = 5 steady-state scenarios in ITER with NBI + ECH.

The selected of IRP Category 1 and 2 high priority issues proposed to be addressed in the next 3 years is highlighted in red in Table 1 below that includes all R&D issues identified for the IRP refinement and consolidation (Categories 1 to 3).

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)	Optimization of shard size, velocity, amount, gas vs. shard fraction, composition (D + impurity) to achieve RE avoidance with optimum TQ, CQ (incl. wall loads)	1	Range of tokamaks with different sizes and plasma parameters to allow extrapolation to ITER (including high I_p tokamak) and with appropriate measurement capabilities	More details on R&D work plan for DMS M. Lehnen, et al., 2018 IAEA FEC Conference Paper	PFPO-1
A.2	SPI-single injector demonstration for runaway mitigation	Determination of feasibility to dissipate the energy of formed runaway beams (amount, assimilation) and to improve scheme	1	Range of tokamaks with different sizes and plasma parameters to allow extrapolation to ITER and with appropriate measurement capabilities	More details on R&D work plan for DMS M. Lehnen, et al., 2018 IAEA FEC Conference Paper	PFPO-1
A.3	SPI-multiple injections	Determination of effectiveness of multiple injections 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	More details on R&D work plan for DMS M. Lehnen, et al., 2018 IAEA FEC Conference Paper	PFPO-1
A.4	SPI-multiple injectors	Determination of effectiveness of multiple injection from different spatial locations to achieve RE avoidance with optimum TQ, CQ (incl. wall loads)	1	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	More details on R&D work plan for DMS M. Lehnen, et al., 2018 IAEA FEC Conference Paper	PFPO-1
A.5	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	More details on R&D work plan for DMS M. Lehnen, et al., 2018 IAEA FEC Conference Paper	PFPO-2

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
A.6	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	More details on R&D work plan for DMS M. Lehnen, et al., 2018 IAEA FEC Conference Paper	PFPO-2
A.7	Laser Induced Desorption for in-situ T retention measurement	Demonstrate LIDS as quantitative in-situ diagnostic measurement for T retention in Be co-deposits at divertor	1	Demonstration in tokamak with Be/W environment	Required to provide in-situ measurements of T retained in divertor Be co-deposits (most likely after each operational day)	FPO
A.8	Single crystal mirror testing	Performance of single crystal mirror with/without active cleaning	1	Demonstration in Be/W environment	Required for evaluation of performance of ITER diagnostics using plasma facing mirrors	PFPO-1
A.9	Laser Induced Breakdown Spectroscopy	Demonstrate LIBS as quantitative measurement for T retention in Be co-deposits on main wall	1	Proof of principle demonstration in Be tokamak environment	Can provide an in-situ measurement of T retention in the first wall during shutdown by installation in a robotic arm	FPO
A.10	SPI-single injector. Pellet injection geometry optimization for RE avoidance (incl. TQ and CQ mitigation)	Optimization of injection direction 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 and with appropriate measurement capabilities	More details on R&D work plan for DMS M. Lehnen, et al., 2018 IAEA FEC Conference Paper	PFPO-1
A.11	Develop capabilities to measure fast ion losses	Demonstration of quantitative measurements in a tokamak environment (ICE) and/or of compatibility with ITER operation (FILD)	2	Demonstration in suitable tokamak with fast particles and under ITER-relevant conditions	Required to provide a direct measurement of fast ion losses in ITER	FPO (in present plans)
A.12	Ammonia formation	Determination of ammonia formation during nitrogen seeded plasmas	2	Divertor tokamaks with metallic PFCs	Provides useful input to the fuel processing plant in ITER	FPO
A.13	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	FPO

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
A.14	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	PFPO-1
A.15	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	PFPO-2
A.16	Neutron Detector development	Compact solid state long life radiation tolerant neutron detector development for in port and in-vessel e.g. advanced self powered neutron detectors	2	neutron laboratory	Extend the operating capability and availability of these systems	PFPO-2
A.17	Polarimetric Thomson Scattering	Demonstration of a working Polarimetric Thomson Scattering on a high temperature device.	2	High temperature plasma device >10keV	Extend the dynamic range in temperature of a classic Thomson scattering system	PFPO-2
A.18	X-ray optics	Develop x-ray reflection systems to allow extended spatial coverage and reduce neutron transmission.	2	X-ray optics laboratory and access to a facility to test the components	Extend the spatial coverage and detector lifetime	PFPO-2
A.19	Two Wavelength Thomson scattering	Demonstration of a working 2-wavelength Thomson Scattering system on a high temperature device.	2	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	PFPO-1

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
A.20	Pressure gauges	LaB_6 electron emitter development for ITER relevant conditions	1	Vacuum facility with magnetic field	Extend pressure range and total run time. Evaluate radiation hardness	PFPO-1
B. Implementation of the ITER Research Plan						
B.1. Disruption characterization, prediction and avoidance (for mitigation see Section A)						
B.1.1	Disruption/VDE thermal load characterization	Characterization of thermal loads during TQ and CQ (magnitude, time dependence and distribution)	2	Range of 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	PFPO-1 (it is assumed that DMS will be very effective after PFPO-1)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
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	PFPO-1 (it is assumed that DMS will be very effective after PFPO-1)
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)	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	PFPO-1 (it is assumed that DMS will be very effective after PFPO-1)
B.1.4	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)	PFPO-1 (a disruption trigger is required for the DMS from PFPO-1 onwards)
B.1.5	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	PFPO-1 (this is required to demonstrate disruption mitigation from PFPO-1 onwards)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.1.6	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 - etc.	FPO and PFPO-2 (this is most important for high current/power operation near operational/physics limits)
B.2. Stationary H-mode plasmas, ELMs, ELM control and impact on H-mode and power fluxes						
B.2.1	He H-mode operation with W PFCs and ELM control	Compare He H-mode plasmas with W PFCs pedestal behaviour, ELM control and W operational issues (W accumulation, PMI issues) with D H-modes. Highest priority is with 3-D fields but other ELM control schemes such as hydrogen pellet pacing and vertical movements are also important	1/2	Tokamaks with W divertor PFCs capable to operate with He H-modes and to investigate ELM control	Required to determine how to relate H-mode operational experience including W control and ELM control in He plasmas to D/DT. If experiments show that this relation is difficult to establish or that risks due to PMI are too high (see below), including He H-modes in PFPO may have to be reconsidered	PFPO-2 (also affects PFPO-1 but largest impact is on PFPO-2 in which available heating is expected to provide much wider operational space for He H-modes)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.2.2	Mixed H + ~ 10 % He H-mode operation and ELM control	Establish H H-mode plasmas with a range of He concentrations to determine the requirements for H-mode access and sustainment and for ELM control schemes compared to H plasmas	1	Tokamaks of various sizes and PFCs and H&CD mixes capable to operate with H + 10 % He H-modes and to investigate ELM control	Required to determine if presence of ~ 10% He can widen H H-mode operational space at ≤ 2.65 T and eliminate the need for He H-modes in ITER.	PFPO-2 (also affects PFPO-1 but largest impact is on PFPO-2 in which available heating is expected to provide wider operational space for H+10% He H-modes)
B.2.3	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)	FPO (this affects all ITER phases. The largest implications are on FPO as it impacts fusion power production and plasma self-heating. Specific issues for PFPO are dealt separately below),
B.2.4	H-mode access and confinement of H plasmas with electron heating and low plasma density in PFPO-1	Evaluate H-mode access and confinement in ITER-like H plasmas for PFPO with dominant electron heating	2	Tokamaks with appropriate heating systems to provide the required heating mix and low density operation in H with low core electron/ion thermal coupling	To refine predictions of expected H-mode access power and confinement in the PFPO phase	PFPO (this affects both PFPO phases because plasma density in H-mode will be restricted by the available power. The largest impact is in PFPO-1 as the only additional heating system available is ECRH)
B.2.5	Pedestal parameters in H-mode plasmas with low grad-n/low n*	Determine limits to pedestal plasma parameters including transport and MHD stability for plasmas with low grad-n/low n* 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	FPO/PFPO-2/PFPO-1 (This affects all ITER phases but has largest implications on FPO as it impacts the maximum pedestal pressure and overall confinement that can be achieved. In PFPO-1 the effect may be smaller because neutral penetration in 5MA/1.8T H-modes is larger and correspondingly larger grad-n could occur in the pedestal)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.2.6	Characterization of H-mode of 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	2	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 and whether the assumption of dominant neoclassical transport can be used to predict impurity penetration through the pedestal in ITER or not	FPO/PFPO-2 (this affects both ITER phases but has largest implications on FPO as there will much less freedom to vary n_{sep} by gas puffing due to the need to control divertor power fluxes in FPO when SOL powers and plasma currents will be highest.
B.2.7	Fuelling of He H-mode plasmas	Determine efficiency of gas fuelling of He H-mode plasmas at high edge densities and temperatures	2	Tokamaks of various sizes and PFCs that can study He H-mode plasmas over a range of edge plasma conditions	Required to determine operational density range of He H-modes in ITER (only gas fuelling is possible) and compatibility with H&CD systems	PFPO-2 (also affects PFPO-1 but largest impact is on PFPO-2 where NBI and ICRH are also available to provide much wider operational space for He H-modes)
B.2.8	Fuelling of H-mode plasmas by peripherally deposited pellets	Evaluate fuelling efficiency of pellets with ITER-like peripheral deposition	2	Tokamaks with HFS pellet injection that can provide peripheral pellet deposition	Required to determine efficiency of core pellet fuelling in ITER and thus of the capability to change core and edge fuelling independently	FPO (PFPO-2 is also affected but required fuelling rates are lower and thus there is more fuelling margin because H-modes will only be explored up to ~ 7.5 MA)
B.2.9	ELM control by 3-D fields with no input torque (RF heated plasmas)	H-mode plasmas with no input torque at moderate ratios of P_{input}/P_{LH}	2	Tokamaks with in-vessel ELM control coils that can operate with RF heated plasmas and/or can control input torque	Initial H-mode operation is foreseen to be able to explore ELM control by 3-D fields in RF only heated plasmas and this may be complex due to associated mode locking if the 3-D fields are not optimized	PFPO-1 (PFPO-2 and FPO can also potentially be affected in the H-mode experiments when no NBI is applied but these are not expected to be extensive)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.2.10	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}	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 requires special optimization of the applied 3-D fields to avoid excessive slowing down of plasma rotation and mode locking	FPO (PFPO-2 is 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.2.11	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	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 measurements	Provides basis for impurity exhaust in ELM controlled regimes by 3-D fields in ITER at its possible optimization in PFPO for application in FPO	FPO (PFPO-1 and PFPO-2 are 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)
B.2.12	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 energy and particle confinement (thermal and fast) 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 PFPO for application in FPO	FPO (PFPO-1 and PFPO-2 are also impacted but to a lesser extent because the impact of 3-D fields on plasma confinement has no operational consequences)
B.2.13	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	FPO (PFPO-1 and PFPO-2 are also impacted but to a lesser extent because if ELM power fluxes to the wall are linked to pedestal plasma parameters these will be lower than for FPO plasmas)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.2.14	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.	FPO
B.2.15	Compatibility of plasma fuelling by peripheral pellet deposition and ELM suppression by 3-D fields	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	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	FPO (PFPO-2 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.2.16	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	2	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 PFPO H-modes	FPO (The experiments in H will be performed in PFPO and on this basis the FPO operational strategy will be elaborated; if there are substantial differences in impurity transport/control among H, D and DT the strategy may need to be reconsidered)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
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	FPO/PFPO-2 (this affects both ITER phases but has largest implications on FPO as higher plasma densities and pedestal temperatures will be explored leading to lower neutral fuelling efficiency)
B.2.18	Effects of TF and TBM ripple on H-mode plasmas	Evaluate the effect of TF ripple and TBM ripple on H performance and its mitigation for TBMs (beyond error field correction)	3	Tokamaks that can explore the effects of TF ripple and localized TBM ripple in H-modes over a range of plasma conditions (particular edge collisionality)	Required to determine the effects over the range of TF ripple levels over which H-modes will be explored in ITER (1.3% at 1.8 T to 0.3% at 5.3 T) and the consequences of the localized TBM ripple/mitigation	PFPO-1/PFPO-2 (For TF ripple largest effects are expected at 1.8T, also can potentially affect FPO if effects of TBM ripple are found to vary strongly with B_t)
B.2.19	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	FPO

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.2.20	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 Be wall erosion is excessive)	FPO (PFPO-1 and PFPO-2 are also impacted but to a lesser extent because, due to limitations in variation of n_{sep} to maintain divertor power exhaust, pedestal W impurity profiles are more likely to be hollow)
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 FPO based on the PFPO hydrogenic H-modes to optimize the initial FPO programme	FPO (The experiments in H will be performed in PFPO and on this basis the FPO 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/r*, its dependence on isotope (H/D/T) /plasma specie (He) 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/r*, 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	FPO (Impact on PFPO-1 and PFPO-2 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	FPO (Impact on PFPO-1 and PFPO-2 is lower because power levels are lower and plasma densities are lower due to lower range of I_p in H-modes)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.3.3	Radiative H-modes with Ne and N 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/r*, 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	FPO (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 PFPO-2 phase but are not essential to meet the objectives)
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	FPO (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 PFPO-2 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	FPO (Impact on PFPO-1 and PFPO-2 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 is not likely to be require to meet the objectives in this phase)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted 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) /plasma specie (He)/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 species 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 Be wall erosion. This may eventually determine the minimum clearance between the separatrix and the wall if such fluxes are expected to be excessive	FPO (Impact on PFPO-1 and PFPO-2 is lower because power levels are lower and plasma densities are lower due to lower range of I_p in H-modes)
B.3.7	Radiative divertor operation in He H-modes	Characterize radiative divertor operation in He H-modes and compare to D plasmas including radiation control aspects	3	Tokamaks that can operate H-modes in He and D with impurity seeding and perform the required measurements of the divertor power fluxes	Required to determine whether the experience gained in developing radiative H-modes in He plasmas is relevant/useful for DT plasmas in ITER	PFPO-2

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.4. Plasma-material/component interactions and consequences for ITER operation						
B.4.1	Be wall erosion in He plasmas	He plasma discharges to determine if expected increased of Be wall erosion leads to coating of the divertor	2	ITER-like Be wall and W divertor is required	Required to determine if Be will coat W divertor in He plasma operation in ITER and thus avoid W-He PMI issues	PFPO-2 (also relevant for PFPO-1 but wall fluxes will be lower due to lower power fluxes and plasma current/densities)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.4.2	He plasma modification of W mechanical properties at high fluences	Long-term exposure of W PFCs to He plasmas and characterization of exposed PFCs	2	Tokamaks with W divertor PFCs or linear plasma devices (long pulse capability is favoured to get to high fluences) operating with He plasmas (preferably H-modes to include synergies with ELMs)	Depending on results this could limit length of He plasma campaigns or eliminate them altogether. Also relevant for PFPO-2.	PFPO-2 (also relevant for PFPO-1 but divertor fluences will be lower due to lower power fluxes and plasma current/densities)
B.4.3	Formation of fuzz by He/W interaction and critical fuzz thickness	Exposure of W PFCs to He plasmas, measurement of fuzz and impact on plasma operation	2	Tokamaks with W divertor PFCs operating with He plasmas (preferably H-modes to include synergies with ELMs)	Required to determine if fuzz is expected to grow on ITER divertor target during He H-mode plasmas and to which thickness it can grow. It should also be assessed whether such thickness is expected to affect plasma operation or not.	PFPO-2 (also relevant for PFPO-1 but divertor fuzz growth expected to be lower due to lower power fluxes/divertor temperature and 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	2	Tokamaks that can expose castellated W structures to H-mode plasmas in a range of conditions and provide the necessary	Required to evaluate power fluxes to the ITER divertor and possible melting or W material deterioration due to high surface temperatures near edges	FPO (Impact on PFPO-1 and PFPO-2 is lower because power fluxes to edges of castellations are due to lower stationary and ELM

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
		dominant physics processes		measurements (power fluxes, currents, etc.)		power fluxes due to lower range of I_p in H-modes)
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 macrobushes 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.	FPO or PFPO-2 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	FPO (Impact on PFPO-1 and PFPO-2 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	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	FPO (Impact on PFPO-1 and PFPO-2 is lower because power levels and plasma density are lower, and thus power/particle fluxes, and plasma discharges are shorter)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
				assess consequences for operation		
B.4.8	Splashing of Be and W under transients	Determine detailed physics mechanisms leading to splashing of molten Be and W PFCs under transients in tokamak experiments	2	Tokamaks that can perform controlled experiments of Be or W melting by transients and diagnose dynamics of molten material	Required to determine expected damage to Be and W PFCs in ITER under transients that cause melting and thus contribution to the determination of the required degree of transient mitigation	PFPO-2 or FPO (Impact on PFPO-1 can also be significant as 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 occur in PFPO-1. For FPO 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 Be or W PFC melt damage for reliable ITER operation thus contributing to the determination of the required degree of transient mitigation	FPO and PFPO-2 (Impact on PFPO-1 is expected to be lower because the level of melt damage will be lower as well as the stationary power fluxes to molten PFCs)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.4.10	Dust production	Determine dominant processes for dust production from metallic PFCs by tokamak operation to provide physics basis for evaluation in ITER	2	Tokamak that can perform experiments to determine contribution to Be 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 FPO. This may impose additional transient mitigation requirements beyond those dictated by PFC lifetime considerations.	FPO or PFPO-2 (Impact on PFPO-1 can also be significant as 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 FPO campaigns or in PFPO-2)
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 Be and W PFCs in ITER under transients for which vapour shielding is expected, thus contributing to the determination of the required degree of transient mitigation	FPO or PFPO-2 (Impact on PFPO-1 can also be significant as ITER plasmas can potentially generate transients leading to vapour shield in this phases. Due to the larger plasma energies and transient power fluxes on PFCs vapour shielding, it is expected to be larger for FPO but because the number of uncontrolled transients is lower in FPO, as mitigation schemes should operate routinely, it may be the case that the largest impact is on PFPO-2 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	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,	Required to determine accumulative effects of ELMs on W divertor erosion lifetime	PFPO-1 and PFPO-2 (this impacts all phases but it is more likely to have larger impact in the PFPO phases when operation with controlled ELMs (and

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
		redeposition during the ELMs themselves		as net erosion is not likely to be measurable ELM resolved		not full ELM suppression) may be viable in ITER
B.4.13	Be wall erosion under controlled ELMs	Determine the net erosion of Be 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 Be 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 Be wall erosion	PFPO-1 and PFPO-2 (this impacts all phases but it is more likely to have larger impact in the PFPO phases when operation with controlled ELMs (and not full ELM suppression) may be viable in ITER
B.5. Start-up, Ohmic, L-mode scenario development						
B.5.1	ECRH assisted plasma start-up	Perform experiments to optimize ECRH assisted start-up and to validate models for ITER	2	Tokamaks that can perform ECRH 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	FP and PFPO-1 because these are the first phases in which ECRH assisted start-up will be used
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	2	Tokamaks that can perform ohmic assisted start-up over a range of conditions (e.g. applied electric field, etc.) and PFCs, including Be wall, and can diagnose the	Required for an initial evaluation of the need for ECRH assisted plasma start up for 1.8 T plasma operation	PFPO-1 (also for PFPO-2 as 1.8T operation is foreseen to compare with reference PFPO-1 plasmas without TBM)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
		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		plasma in this initial phase.		
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	3	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 optimum path from low I_p to high I_p L-modes, as central heating levels in ITER are B_t dependent for ECRH and ICRH and central heating is not maintained for all paths	PFPO-2 (FPO will be impacted in a similar way as the optimum path in L-mode is expected to be developed in PFPO-2 for H plasmas and tuned for DD and DT plasmas in FPO)
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	3	Tokamaks that can explore various levels of additional heating and ramp-rates and determine 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 achieve suitable q profile at the end of the ramp (for long pulse Q = 5 scenarios) or to reduce flux consumption in ramp-up (for Q = 10 scenario)	FPO (PFPO-1 and PFPO-2 can also be affected but the degree of optimization required for ITER plasma scenarios is much higher in FPO)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.6. Conditioning, fuel inventory control						
B.6.1	ECWC conditioning	Perform ECWC conditioning and determine requirements for effective cleaning and reliable plasma start-up including recovery from unmitigated and mitigated disruptions	1	Tokamaks that are equipped with ECRH systems and can perform ECWC on a routine basis for general machine conditioning and disruption recovery and diagnose the ECWC plasma	This is one of the two cleaning technique available in ITER with toroidal field on. This is the only cleaning technique for PFPO-1 with the TF field on. Without an effective cleaning technique in this phase the experimental programme could be significantly slowed down (e.g. recovery from disruptions or disruption mitigation)	PFPO-1 because ECWC is the only conditioning technique possible with the TF on in this phase
B.6.2	Operation at low wall temperatures and ITER-like baking cycles	Determine resulting fuel retention by operation at ITER-like temperatures during non-active phases and compare to higher wall temperature operation	2	Tokamaks with ITER-like wall/divertor materials and capable to operate at ITER-like temperatures for wall and baking and to measure outgassed and retain fuel	Required to determine if the wall temperature plays a major role in ITER fuel retention as wall temperature will be different between PFPO and FPO and this could affect the evaluation of expected T retention in FPO on the basis of PFPO measurements in ITER	PFPO-1 and PFPO-2
B.6.3	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	3	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

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.6.4	Efficiency of T removal by plasma operation from Be codeposits in routinely unexposed areas	Evaluate the efficiency of fuel removal in divertor Be codeposits (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.	3	Tokamaks with W/Be PFCs where Be codeposits 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 Be co-deposits) avoiding the need of more complex T removal techniques.	FPO
B.6.5	ICWC fuel removal from Be codeposits	Evaluate the efficiency of fuel removal in divertor Be codeposits by ICWC	3	Tokamaks that can apply ICWC on Be codeposits 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 Be codeposits.	FPO
B.6.6	Start of operation by baking without GDC	Determine if it is possible to start plasma operation following a machine opening to air by only baking in tokamaks with metallic PFCs	3	Tokamaks with metallic PFCs (W/Be preferred) that can perform ITER-like baking cycles	Required to determine to which degree GDC is mandatory to restart plasma operation in ITER given possible issues of GDC electrode lifetime	FPO
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 evaluate the requirements to correct error fields caused by ferromagnetic elements in ITER. To optimize error field correction in ITER by the use of external coils, possibly supported by internal coils.	PFPO-1, PFPO-2 (addition of TBMs) and FPO (larger plasma b_N)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
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.	FPO
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	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	PFPO-1, PFPO-2 (addition of TBMs) and FPO (larger plasma b_N)
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	FPO (this also affects PFPO-1 and PFPO-2 when 3-D fields are applied for ELM control, but for FPO the consequences of a plasma position error are much larger because of the larger plasma energies and power fluxes)
B.7.5	Develop real time ICRH coupling control loop	Develop a real time ICRH 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 such control loop	Required to ensure required ICRH power through time-varying plasma conditions	PFPO-2 and FPO

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
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	FPO (PFPO-1 and PFPO-2 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	PFPO-1 and PFPO-2 as this control loop needs to be developed before FPO
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	PFPO-2 and FPO
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	PFPO-2 and FPO

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
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	PFPO-1 as 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}	FPO (PFPO-1 and PFPO-2 are also affected but consequences of lack of ELM control/W accumulation in these phases are largest for FPO 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	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 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)	FPO (PFPO-1 and PFPO-2 are also affected but consequences of lack of ELM control/W accumulation in H-mode termination are largest for FPO due to the larger

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
		relevant normalized timescales		required parameters (suitable measurements are required)		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	FPO (this also affects PFPO-1 and PFPO-2 but error field effects are expected to be larger with larger b_N so that a larger level of correction is required in FPO)
B.9. Complex scenario control during stationary phases						
B.9.1	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 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	PFPO-2 and FPO

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.9.2	Integrated control of radiative Ne/N H-mode in ITER-like conditions	Demonstrate divertor power load/detachment control while maintaining N/Ne 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/N H-modes and control divertor power loads with ITER-like actuators	Required for routine ITER operation in high Q reference scenarios	FPO (Impact on PFPO-1 and PFPO-2 is lower because power levels are lower and plasma densities are lower due to lower range of I_p in H-modes. Thus radiative divertor operation may not be required to meet the objectives in these phases)
B.9.3	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 but more serious consequences for FPO
B.9.4	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 b_N H-modes plasmas required to achieve ITER's high Q goals	FPO (PFPO-2 will also be affected as NTM control will be commissioned in this phase so that it can be routinely applied in FPO)
B.9.5	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 control of 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	Required to ensure robust operation at high Q in ITER. An initial demonstration in existing experiments is desirable to refine operational control/strategy on ITER	FPO

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.9.6	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	PFPO-2/FPO
B.9.7	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	FPO
B.10. Validation of scenario modelling and analysis tools						

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.10.1	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.10.2	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.10.3	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.10.4	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 ITER-like plasma 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

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.10.5	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 PFPO	Demonstrate foreseen 3-ion scheme for H plasmas in PFPO-2 in plasma conditions relevant to their application in ITER ($H:He^4:He^3$) or alternatives	2	Tokamaks with appropriate levels of ICRH heating and frequency range to demonstrate 3-ion schemes in ITER-relevant H plasmas ($H:He^4:He^3$) or alternatives	Advanced ICRH scheme can potentially widen hydrogen H-mode operational space in PFPO-2. These capabilities need to be assessed experimentally before they can be considered for refinement of the ITER Research Plan	PFPO-2
B.11.2	Validation of shine-through loads with high energy NBI	Perform experiments with high energy NBI (E_{NBI} ~500 keV) to validate models for evaluation of shine-through loads in ITER	2	Tokamaks with high energy NBI and good diagnostics of shine-through power fluxes on PFCs	Required to accurately determine the Hydrogen H-mode operational space which is limited (in the low density side) by shine-through loads	PFPO-2 (FPO is also affected but because shine-through loads of D beams on D or DT plasmas are much lower and thus the consequences of revised shine-through loads are expected to be minor)

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
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	FPO (PFPO-2 is also affected but margin to optimize the fields is larger due to the lower I_p levels)
B.11.4	Advanced ICRH schemes for DT	Demonstrate foreseen 3-ion schemes (D:T:Be) for alternatives or DT plasmas in FPO in plasma conditions relevant to their application in ITER	2	Tokamaks with appropriate levels of ICRH heating and frequency range to demonstrate 3-ion schemes in DT ITER-relevant plasmas (D:T:Be) or alternatives	Advanced ICRH scheme can potentially avoid the use of He^3 for ion heating in DT plasmas. These capabilities need to be assessed experimentally before they can be considered for further optimization of the ITER Research Plan	FPO
B.11.5	Alfven Eigenmode stability in H/He/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 PFPO-2 in preparation/mitigation of possible problems in FPO	From PFPO-2 (ICRH and NBI) but most relevant for FPO because of alpha particles
B.11.6	Validation of models for ECH absorption in 3rd harmonic operation at 5 MA/ 1.8 T plasmas	Provide quantitative experimental evidence on 3rd harmonic absorption in plasma scenario conditions similar to those in 5 MA / 1.8 T plasmas and validate modelling predictions	2	Tokamaks equipped with ECH and with diagnostics for non-absorbed ECH power that can perform ITER-like experiments with 3rd harmonic ECH (namely starting with plasma conditions with low ~ 10-	3rd harmonic ECH is required for H-mode operation at 5 MA / 1.8 T in PFPO-1. Non-absorbed ECH power is expected on the basis of modelling when heating ohmic plasmas but the duration of the low absorbed power phase is short and thus acceptable for the	Key for PFPO-1 but also has impact on PFPO-2

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
				20 % absorption) and study the temporal increase of the absorption for various levels of ECH power	ITER first wall. These modelling predictions are done with models that need to be validated quantitatively in experiment	
B.11.7	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.11.8	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	FPO and PFPO-2
B.11.9	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 (coupling, 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	PFPO-2 and FPO
B.11.10	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	PFPO-2 and FPO

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.11.11	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	PFPO-2 and FPO
B.11.12	ITER hydrogen fast particle MHD stability for H-mode operation with high NBI power and q_{95} = 3 at 7.5 MA	Determine expected fast particle MHD stability with high NBI power and q_{95} = 3 at 7.5 MA by modelling and experiments in present tokamaks	3	Tokamaks with significant NBI fast particle densities and b_{fast} in relevant conditions to the stability of ITER plasmas	Required to determine if fast particle instabilities will occur and have to be controlled in 7.5 MA/2.65T ITER H-modes with NBI heating	PFPO-2
B.12. Specific issues for long pulse/enhanced confinement scenarios						
B.12.1	ELM suppression in high q_{95} / high b_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 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 b_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 b_N scenarios	Required to determine 3-D field structure for ELM control in long pulse Q = 5 scenario and to evaluate consequences for plasma confinement	FPO

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
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 ITER scenarios) and whether this confinement enhancement basis extrapolates to ITER	FPO
B.12.3	Pellet fuelling of high q_{95} / high b_N H-mode plasmas	Determine the effect of high q_{95} / high b_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 b_N scenarios	Required to determine if high q_{95}/high b_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	FPO
B.12.4	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	PFPO-2 and FPO (ramp-up optimization is developed in PFPO-2 for application in FPO)
B.12.5	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.	FPO

Ref.	System/ Issue	Required R&D	Category*	Required experimental facilities	Comment	Phase when system required/ Most impacted Phase
B.12.6	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 b_N operation in ITER Q = 5 steady-state plasmas	FPO

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

Disclaimer

References