ITER Council

Science and Technology Advisory Committee (STAC)
5th Meeting

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ITER Physics Work Programme 2009-2011

Proposal from the Principal Deputy Director-General

Recommendation:

The IC-STAC is invited to:

i. Note this proposal on the ITER Physics Work Programme for the period 2009-2011.

ITER Physics Work Programme 2009-2011

Abstract

A Physics Workprogramme for the period 2009-2011 is proposed which is intended to address key design and R&D issues in ITER Physics in support of the construction of the ITER device and the preparations for operation and exploitation. The Work Programme is analyzed within the framework of the FST departmental structure and provides a detailed definition of the technical activities foreseen for the period 2009 to 2011, an indication of implementation mechanisms and required resources, and estimates of timescales on which key deliverables are required.

Contents

1 Introduction

It is foreseen that the ITER Physics research programme will develop along three principal lines:

- Provision of necessary technical support to the ITER construction project, which will dominate activities during the coming 3-5 years;

- Development of detailed plans for ITER plasma commissioning and exploitation phases, including necessary supporting R&D, which will assume increasing importance during the second half of the ITER construction phase;

- Implementation of an extensive programme of experimental, modelling and theory research to exploit the ITER device, which will be the focus of the operational phase.

As presented at the STAC-4 meeting, the ITER Research Plan (IRP) is intended to define the research activities of the ITER Project during the Construction, Commissioning and Operation phases. The level of definition will gradually increase as the detailed requirements for the provision of support to Project activities and input from the Members' fusion communities are incorporated. Version 1.0 of the IRP [1] analyzed the principal lines of research during the Operation phase and defined priorities for R&D during the Construction phase. The present document builds on the discussion contained in the IRP to define more detailed priorities for Physics research in the period 2009-2011 so as to identify the resources and implementation instruments required and to provide guidance to the Members' fusion communities as to the Project's needs during this period.

In preparing this document, priority has been given to the need to provide physics support to ITER construction in areas such as:

- implementation of the outcome of the Design Review (eg PF system, RMP coils);

- issuing of Procurement Arrangements (eg Vacuum Vessel, Divertor, Disruption Mitigation System);

- the definition of requirements for the ITER plasma control system, in particular as these relate to the development of plasma control strategies and physics algorithms;

- promotion of research addressing key operational issues having an impact on the definition of ITER sub-systems (eg scenarios, tritium retention and removal, dust characterization, TBM-related issues);

- support to the Project's licensing activities.

Activities with a longer term perspective have also been included in preparation for later programmatic priorities such as:

- development of a comprehensive modelling capability, building on activities in the Members' programmes;

- further development of the ITER Research Plan.

In formulating the Wotk Programme, guidance has also been derived from recent detailed discussions with STAC and the fusion community, as reflected in issues such as ELM control and R&D on plasma operation with tungsten PFCs, and from feedback obtained from representatives of the ITER DAs to whom preliminary drafts of the Work Programme were distributed.

There are several instruments for drawing on the expertise of researchers in the Members' fusion communities, defining possible implementation mechanisms for the activities within the Work Programme:

- International Tokamak Physics Activity (ITPA)

- R&D Tasks (initial implementation via DAs)

- Design Tasks (via DAs)

- Visiting Researchers (via DAs)

- Integrated Modelling Programme (under definition)

- External Service Contracts

Where possible in the following detailed technical definitions of the Work Programme, we have indicated the most likely implementation mechanism for specific activities and, where appropriate, the anticipated level of resources required. However, both aspects may evolve through our experiences in the implementation of the programme. In addition, in several areas the most effective instrument for implementation is not entirely clear, and further discussion, in particular with DAs, may be required to finalize the definition of these aspects of the programme.

To facilitate the definition of the Work Programme, the analysis of research needs and activities has been performed within the structure of FST Science Division, since implementation will, to a significant extent, be carried out within the framework. The Work Programme definition is therefore discussed in the following sections within the areas:

- Transport and Confinement Physics

- Plasma Stability

- Divertor and Plasma-Wall Interactions

- Plasma Operations
- Integrated Modelling

The presentation of each area of the Work Programme consists of two sections: the detailed discussions within this document and timeschedules contained in the document IC-STAC-5-4-1-02_ITER-Physics-Workplan-Timeschedule (2EYE3J). The latter document provides estimates of the timescale for the implementation and, where possible, completion of each activity, together with the implementation instrument foreseen and an indication of the level of resources considered necessary (resources are not allocated to voluntary activities such as the ITPA). Higher priority activities, for example those where the timeschedule is constrained by the need to provide input requirements for technical specifications related to Procurement Arrangements, are indicated in red. Analysis of the Integrated Project Schedule to understand the implications for the Physics Work Programme is a continuing process which will ultimately need to be carried out in consultation with the responsible officers for the various WBS where physics input may be required in order to define the input requirements and timescales more precisely. It can be expected, therefore, that these timeschedules will be subject to further revision as the definition of the requirements of ITER sub-systems becomes clearer and as feedback from the fusion community is properly reflected in the programme.

It is anticipated that the Work Programme will be subject to major revisions annually, but it will also be kept under review on a shorter timescale to ensure that the Project's emerging needs are adequately reflected and to build on feedback provided by the Members' DAs and fusion communities.

References

[1] ITER Research Plan v1.0 (2FB8AC_v1_0)

2 Work Programme in Transport and Confinement Physics

The proposed R&D for ITER in this area, detailed below, is foreseen to be implemented as part of ITPA activities and Voluntary Physics R&D by ITER members. The activities are divided into two timescales: short term (final results needed before 2011) and medium term (R&D can be started/continued at this stage but results could be provided after 2011). The timescale of the short term activities is determined by their direct influence on procurement packages or on the foreseen operation of ITER and/or its reference operating scenarios. The ITER systems which could be influenced by this R&D are:

- Plasma magnetic control schemes and requirements (detailed specifications to be developed from 2010).
- Final criteria for optimisation of the ferromagnetic inserts for TF ripple correction needs to be developed (end of 2009-beginning of 2010).
- Pellet injection system and Gas Introduction systems (Procurement Agreements to be signed in May-July 2010).
- First wall and blanket modules (Procurement Agreements to be signed in July 2011).
- In-vessel coils for ELM control and Vertical position control (Procurement Agreements tentatively scheduled for signature in September 2010).

2.1 Short term activities (2008-2010).

2.1.1 Transport and Confinement during transient phases.

Most studies of plasma confinement and transport for ITER have been focused so far on the flat-top burning plasma phases of the scenarios as they determine the fusion performance of the device. Development of a good physics basis for plasma transport and confinements in ITER during transient phases (ramp-up/down, confinement transients during flat top, etc.) is now required as it has an impact on the PF/control specifications for ITER and on the development of ITER scenarios as a whole, either because of the consumption of resistive flux during these phases (ramp-up/down) or because of the margins in coil currents that have to be reserved to cope with plasma movements associated with confinement transients during the flat top of the discharges.

So far the ITER scenario studies have been performed on the basis of empirical scalings (Ejima for resistive flux consumption, scaling with global τ_E of similar confinement transients in present experiments) and/or by the application of transport models/scaling laws for plasma confinement which have been developed for steady-state phases of plasma discharges. Given the degree of optimisation of the ramp-up/down (heated vs. non-heated/L-mode vs. H-mode) phases required for maintaining ITER operational flexibility and ITER specific features (such as alpha heating) which can affect the evolution of the plasma during transients, it is necessary to provide a firm physics basis for the transport models/scaling during these phases in ITER.

- Development of transport models/scalings for ITER ramp-up/down phases (ohmic, L-mode and H-mode). This R&D requires coordinated experiments and analysis/modelling of experimental results by relevant ITPA groups (Transport and Confinement, Pedestal and Integrated Operation Scenarios) and experts at ITER parties. An initial set of experiments to address these issues has already been carried out by the ITPA group on Integrated Operation Scenarios or under Voluntary Physics R&D by ITER members, as part of the ITER design Review activities and further experiments are required in this area in the coming years as the detailed requirements for ITER operation are elaborated. It is therefore required that other relevant ITPA groups (Pedestal and Transport and Confinement) and experts at ITER parties participate in the detailed transport analysis/modelling of these experimental results so that quantitative conclusions can be extracted for the design of ITER operational scenarios.

- Development of transport models/scalings for confinement transients during stationary phases of the discharges. These include among others L-H transitions, H-L transitions, collapses of internal transport barriers, collapses of peaked density profile regimes, etc. Some well diagnosed experimental results have been obtained as part of the Voluntary R&D carried out at the ITER members facilities for the ITER Design Review. It is required that the relevant ITPA groups (Integrated Operation Scenarios, Pedestal and Transport and Confinement) carry out coordinated experiments and detailed transport analysis/modelling of these experimental results so that quantitative conclusions can be extracted for the design of ITER operational scenarios.

2.1.2 Access to high confinement regimes in ITER during steady/state and ramp-up/down H, D and DT phases.

The achievement of high fusion gain regimes in ITER is based on the access to enhanced confinement regimes with an edge transport barrier (H-mode) which provides normalised energy confinement $H_{98} \geq 1$ at the densities required for high fusion gain in ITER. Similarly, control of the current profile shape in ITER during ramp up/down phases may require operation in H-mode in conditions with varying plasma current, shape, etc. Access to these regimes has been so far characterised by maintaining a power flow which exceeds the L-H power threshold by a given margin (typically a factor of ~ 1.3-2 for edge power flow). There are several uncertainties related to the approach above which need to be resolved in the next years as they have direct influence on the design of ITER scenarios and ITER research plan during the non-nuclear phase such as: access to good H-mode confinement in hydrogen would allow ELM control strategies to be developed at this stage, the development of ITER scenarios foresees H-mode operation in DT plasmas during current ramp-up/down phases and before substantial alpha heating, etc.

The proposed R&D activities during the coming years in this area are summarised briefly below:

- Determination of power required for H-mode threshold in ITER-like plasma conditions (high δ/κ, low and high n_e/n_{GW}, low plasma-main wall interaction (for instance between separatrix/wall ~ 2 $\lambda_n^{L\text{-}mode}$), etc.) and development of strategies for its minimisation.

- Determination of power required for steady ($\Delta t \geq 10\ \tau_E$) Type III H-mode operation in ITER-like plasma conditions (high δ/κ, high n_e/n_{GW}, low plasma-main wall interaction (for instance between separatrix/wall ~ 2 $\lambda_n^{L\text{-}mode}$), etc.) and development of strategies for its minimisation.

- Determination of power required for steady ($\Delta t \geq 10\ \tau_E$) $H_{98} = 1$ H-mode (Type I ELMy H-mode, EDA, etc.) in ITER-like plasma conditions (high δ/κ, low and high n_e/n_{GW}, low plasma-main wall interactions (for instance between separatrix/wall ~ 2 $\lambda_n^{L\text{-mode}}$), etc.) and development of strategies for its minimisation.

- Determination of power required for H-mode access during plasma current ramp-up/down phases with similar requirements to ITER (normalised dI_p/dt, plasma shape evolution, etc.)

- Isotope mass and species scaling (H and He plasmas) of required power for H-mode access, Type III ELMy H-mode, steady ($\Delta t \geq 10\ \tau_E$) $H_{98} = 1$ H-mode (Type I ELMy H-mode, EDA, etc.) in ITER-like plasma conditions (high δ/κ, high n_e/n_{GW}, low plasma-main wall interaction (for instance between separatrix/wall ~ 2 $\lambda_n^{L\text{-mode}}$), etc.) and development of strategies for their minimisation.

2.1.3 Characterisation of proposed schemes for active ELM control, compatibility with scenario requirements.

The characterisation of ELMs, of their associated loads to PFCs and the evaluation of their expected values in ITER has advanced considerably in the last 5-10 years. As a consequence of this advance, it has been possible to evaluate in a quantitative way the need for control and mitigation of the ELM loads in ITER. Together with this activity, an extensive R& D programme has been carried out by the ITER parties to develop techniques for ELM control applicable to ITER. This R&D has identified various ELM control schemes which are being considered for implementation in ITER. These are, for example, ELM control/suppression by edge magnetic field perturbation and pellet injection. Despite these significant advances, the understanding of the physics processes that govern the application of the methods for ELM control in ITER remains incomplete. Furthermore, the proposed ELM control schemes have to meet many additional requirements for their practical implementation in ITER, besides sufficient plasma confinement, which remain to be investigated in detail in present experiments. These requirements are, for example: the achievement of the required normalised density needed for the expected fusion performance in ITER, the demonstration of the scheme at low pedestal collisionality, the compatibility of the scheme with the required He pumping and divertor power load control in ITER, etc.

In view of these uncertainties, the strategy for ELM control in ITER has to be based in a portfolio of approaches, whose further development and detailed application to ITER requires an R&D programme to be carried out by the ITER members' in cooperation with the ITER Organisation. This R&D programme is proposed to address the issues below in the coming 2-3 years by: i) exploitation of the various capabilities of the existing experiments (i.e., accessible collisionality range, power to momentum input ratio, internal/external magnetic perturbation coils and spectra, pellet particle content to plasma particle content ratio, etc.), ii) execution of coordinated experiments to determine dimensionless or/and dimensional empirical scaling laws describing the expected performance and effects on plasma of proposed ELM control schemes in ITER, iii) addressing the key compatibility issues of the ELM control techniques with other scenario requirements in ITER such as those associated with core plasma confinement and plasma density, He exhaust, etc., by coordinated assessment of the of the present schemes and joint development of new schemes for ITER and iv) coordinated data analysis and modelling to develop a physics-based evaluation of the expected requirements and performance of the proposed ELM control methods in ITER.

The proposed R&D activities during the coming 2-3 years in this area are summarised briefly below and concern the consolidation of the physics basis behind the schemes for ELM control in ITER in a first phase followed by a second phase in which the emphasis will centre on the issues related to integration with other ITER scenario requirements. The implementation of these activities relies mainly on experimental input and data analysis carried out under the ITPA framework and/or as direct contribution by the ITER members to the ITER physics R&D.

2.1.3.1 Determination of edge magnetic field perturbation characteristics for ELM control and integration with requirements for ITER high fusion gain scenarios

2.1.3.1.1 Consolidation of physics basis for ELM control by edge magnetic field perturbation.

The key issues to be addressed by the R&D programme aim at providing a solid physics basis for the evaluation of the performance of ITER's ELM control coil systems are:

- Dependence of ELM control/suppression on edge plasma collisionality and/or plasma density
- Plasma response to edge magnetic field perturbation: edge magnetic field screening, effect on plasma rotation/rotation shear, influence of pedestal and/or total plasma β
- Influence of edge magnetic field spectrum on degree of ELM control/suppression: dependence on edge q, plasma shape, resonant versus non resonant spectra, etc.
- Effect of edge magnetic perturbation on edge energy and particle transport and edge plasma stability

2.1.3.1.2 Operational issues and integration with overall requirements for ITER high fusion gain scenarios

This concerns the practical application of the ELM control coils in ITER and its compatibility with scenario requirements for ITER:

- Effect control/suppression of ELMs by edge magnetic perturbation on pedestal pressure in H-modes
- Compatibility of ELM-control/suppression with gas/pellet fuelling and (DT, He) exhaust requirements
- Compatibility of ELM-control/suppression with stationary heat flux control
- Interaction of edge magnetic field perturbations with other detrimental MHD activity expected in ITER scenarios (NTMs, RWMs, locked modes, …)
- Effect of edge magnetic field perturbations on plasma rotation/rotational shear and core confinement
- Effect of edge magnetic field perturbations on plasma pedestal and confinement for H-modes with $P_{edge}/P_{L-H} < 2$
- Application of ELM control by edge magnetic field perturbation to transient phases of the discharge such as H-mode current ramp up/down and H-mode phases in the current flat top before and after full performance phase

2.1.3.2 Determination of requirements for ELM control by pellet pacing and integration with ITER high fusion gain scenarios.

2.1.3.2.1 Consolidation of physics basis for ELM control by pellet pacing.

The key issues to be addressed by the R&D programme to provide a solid physics basis for the evaluation of the requirements and the effects on the plasma of the pellet system for ELM control in ITER are:

- Determination of physics processes leading to the triggering of ELMs by pellets in Type I H-mode plasmas
- Plasma energy and particle transport during and following high-frequency pellet-triggered ELMs

2.1.3.2.2 Operational issues and integration with overall requirements for ITER high fusion gain scenarios.

This concerns the practical application of ELM control by pellet injection in ITER and the compatibility of its effects on the plasma with scenario requirements for ITER:

- Determination of the maximum ELM frequency enhancement (with respect to the natural ELM frequency) achievable while maintaining Type I ELMy H-mode or $H_{98} \sim 1$ regime.
- Determination of the minimum pellet size required to trigger an ELM
- Evaluation of the effects on the average pedestal pressure in H-modes of control of ELMs by pellet injection
- Compatibility of ELM control by pellet injection with gas/pellet fuelling and (DT, He) exhaust requirements
- Compatibility of ELM control by pellet injection with stationary heat flux control

2.1.4 Determination of ripple effects on ITER plasma performance and on fast particle confinement.

In the last years the detrimental effects of toroidal field ripple on thermal plasma confinement have been documented by dedicated experiments in JET (variable toroidal ripple studies) and JT-60U (addition of ferromagnetic inserts). While many of the qualitative effects seen in both experiments are similar, the physics processes that lead to the observed plasma behaviour and their extrapolation to ITER remain subject of R&D. Until now, the criteria to define the ripple reduction by ferromagnetic inserts for ITER have been based entirely on the reduction of first wall loads by fast particles. Given the need to define the optimisation criteria for ferromagnetic inserts in ITER and to estimate the loads by fast particles on the wall, it is therefore necessary to carry out an R&D programme to identify the physics mechanisms that lead to the observed pedestal plasma/ELM modifications by ripple and to quantitatively validate the models used for the evaluation of fast particle loads on first wall components for ITER reference scenarios.

The proposed R&D activities during the coming years in this area are summarised briefly below. The implementation of these activities relies mainly on experimental input and data analysis carried out under the ITPA framework (Pedestal, Transport and Confinement, Fast Particles Groups) and/or as direct contribution by the ITER members to the ITER physics R&D:

- Determination of physics mechanisms leading to decrease of thermal energy/plasma density and changes in ELM behaviour with toroidally periodic TF ripple and dependencies on plasma dimensional and dimensionless parameters and fast particle losses/plasma rotation. In particular the possible role of edge ergodisation by TF ripple and similarities with ELM control experiments by edge magnetic perturbations should be explored by experiments and modelling. This requires dedicated experiments with systematic variation of TF ripple, plasma shape shear and plasma β (which affect the edge magnetic field structure). The effects of ripple-induced fast particle losses on plasma pedestal and confinement could be assessed in a more conclusive way for ITER by comparing similar NBI and RF dominated H-mode plasmas at various levels of ripple (by changing current in TF coils or plasma radial displacement). This should be accompanied with the development of models able to reproduce these effects that can then be used to evaluate their magnitude in ITER.

- Validation of models for fast particle loads on plasma facing components with experimental measurements by systematic experiments (variation of fast particle energy, ripple magnitude, distance separatrix-wall, plasma density, plasma shape, etc.) and measurements of the resulting power loads.

2.1.5 Particle transport and fuelling in ITER reference scenario plasmas.

The proposed R&D activities in this area are summarised briefly below and its implementation relies mainly on work carried out under the ITPA framework (Pedestal and Transport and Confinement Groups) and/or as direct voluntary contribution by the ITER members to the ITER physics R&D.

- Demonstration of physics basis for core plasma fuelling in ITER. Fuelling of the core plasma in ITER relies on the effectiveness of plasma fuelling by pellets which are ablated in the plasma periphery (typically 5-10 cm inside the separatrix), i.e. not far inside from the expected position of the pedestal top in ITER. The efficiency of this scheme has specific issues that need to be addressed experimentally such as the additional losses caused by triggering of ELMs, enhanced particle losses following the pellet ablation caused by shallow deposition, etc. Therefore the experimental demonstration of the effectiveness of such a scheme to fuel bulk plasmas (to similar densities as required in ITER) in present devices remains outstanding. It is proposed that experiments in present devices are carried out where the pellet ablation profile is matched as close as possible to that expected in ITER by adjusting pellet size and velocity and adequate edge plasma parameters (by I_p and input power level, for instance) while adjusting the total fuelling rate to achieve a ratio $\langle n_e \rangle / n_{GW}$ = 0.7-0.9. The experiments should address (either within a single device or by coordinated multi-machine experiments) the role of dimensional and dimensionless plasma parameters which are expected to influence pellet transport dynamics and core particle transport (such as pedestal temperature, pressure, core plasma collisionality, etc.).

2.2 Medium term activities (2011 and beyond).

2.2.1 Pedestal width, pedestal energy and uncontrolled ELM energy loss in ITER.

Present understanding of the core plasma transport in ITER indicate a strong link between the core temperatures and those at the pedestal top. Therefore, global plasma confinement and fusion gain in ITER are strongly linked to the magnitude of the pedestal plasma parameters. The determination of the pedestal parameters in ITER has so far been done on the basis of the expected limit for edge stability (peeling-ballooning modelling) and various assumptions concerning the scaling of the pedestal width from present devices to ITER. The scaling of the pedestal width for plasma density and temperature with global plasma conditions within an experimental device and with device size from multi-machine comparisons is presently the subject of R&D and, recently, a scaling for the pressure pedestal width and methodology has been proposed to scale these experimental results to ITER. Further R&D is required to validate this approach as well as to develop an understanding of the physics processes that determine the scaling of the temperature and density pedestal widths and their inter-relation that leads to the pressure pedestal width, which ultimately limits the magnitude of the pedestal pressure in ITER. Similarly, the scaling of ELM energy losses from present devices to ITER has revealed an overall dependence of these losses on pedestal plasma collisionality for regimes with high pedestal pressure, while other dependences on dimensional and dimensionless parameters have been investigated without a firm projection to ITER (scaling with ρ^*, dependence on edge rotation, dependence on plasma shape/q_{95}, etc.). Further R&D is necessary to determine whether any of these "outstanding" dependencies can affect significantly the expected uncontrolled ELM energy loss in ITER.

The proposed R&D activities in this area are summarised briefly below:

- Determination of density, temperature and pressure pedestal width scaling versus dimensional and dimensionless edge parameters. This includes experiments in a single device and across experimental devices where pedestal parameters are matched (in the dimensionless sense) and systematic variations of ρ^*, ν^* (I_p, n_e, P_{input}) and plasma toroidal rotation, for instance. An important point to be addressed in these experiments is the quantification of the role of neutral penetration in the determination of the pedestal width and its interrelation with the pedestal pressure/edge power flux. Such area could be addressed by experiments within a single device comparing edge gas fuelling and pellet fuelling over range of plasma parameters (mainly I_p) and by dedicated multi-machine comparisons aimed at matching neutral penetration parameters and doing some systematic variations.

- Determination of ELM energy loss dependence on dimensional and dimensionless pedestal plasma parameters. Most of this R&D should be carried out in coordination with the above activity and aims at determining additional dependencies of the uncontrolled ELM size in ITER beyond that on ν^*, some of which have been indicated by existing experimental results. These include, among others, dependences on ρ^*, plasma toroidal rotation, q_{95} (for high δ), etc. Their importance for ITER needs to be quantified by systematic experiments within single devices and by appropriate multi-machine comparisons, for instance by comparable scans of normalised plasma rotation at constant input powers, variations of q_{95} at ITER-like δ, etc.

2.2.2 Development of alternative regimes providing high fusion performance in ITER without or with small ELMs compatible with overall scenario requirements.

In the last years, an extensive R& D programme has been carried out by the ITER parties to explore small ELM/ELM-free plasma regimes alternative to the Type I ELMy H-mode, which can meet the plasma confinement requirements for Q_{DT}=10 and Q_{DT}=5 operation in ITER. This R&D has identified various alternative plasma regimes, which are being considered for implementation/exploration in ITER. Despite these significant advances, the understanding of the physics processes that lead to the suppression of Type I ELMs in the small ELM regimes remains incomplete. Furthermore, the proposed small ELM regimes have to meet many additional requirements for their practical implementation in ITER, besides sufficient plasma confinement, which remain to be investigated in detail in present experiments. These requirements are, for example: the achievement of the required normalised density needed for the expected fusion performance in ITER, the demonstration of the regime at low pedestal collisionality, the compatibility of the regime with the required He pumping and divertor power load control in ITER, etc.

In view of these uncertainties, the strategy for development of small ELM regimes for ITER has to be based in a portfolio of approaches, whose further development and detailed application to ITER requires an R&D programme to be carried out by the ITER members' in cooperation with the ITER Organisation. This R&D programme is proposed to address the issues below by: i) exploitation of the various capabilities of the existing experiments (i.e., accessible collisionality range, power to momentum input ratio, etc.), ii) execution of coordinated experiments to determine dimensionless or/and dimensional empirical scaling laws describing the expected performance and effects on plasma of proposed small ELM regimes in ITER, iii) addressing the key compatibility issues of the small ELM regimes with other scenario requirements in ITER such as those associated with core plasma confinement and plasma density, He exhaust, etc., by coordinated assessment of the present regimes and joint development of new regimes for ITER and iv) coordinated data analysis and modelling to develop a physics-based evaluation of the expected requirements for access and performance of the small ELM regimes proposed for ITER.

The proposed R&D activities in this area are summarised briefly below and concern the consolidation of the physics basis behind the regimes with no or small ELMs and their extension towards the plasma conditions expected at the plasma pedestal in ITER:

- Determination of the physics processes leading to the appearance of the edge harmonic oscillation leading to the QH-mode (plasma rotation/rotational shear, fast ion loss, etc.) and implications for low collisionality/high density operation in ITER
- Analysis of the role of plasma edge stability (including plasma shaping and q_{95}) and plasma conditions for grassy ELM regimes and their extrapolability to plasma conditions required for advance regimes in ITER
- Determination of the role of plasma edge stability (including plasma shaping and q_{95}) and edge plasma collisionality for high collisionality regimes with small or no ELMs (Type II, EDA, etc.) and possible extension towards lower q_{95} and lower edge plasma collisionality conditions
- Characterisation of the Type III ELMy H-mode regimes both in conditions of low and high collisionality (high density/high radiative fraction) and evaluation of the pedestal pressure degradation with respect to Type I ELMy H-mode.
- Development of new small ELM regimes and enhanced confinement features. This could, for instance, combined Type III ELMy H-mode with hybrid-like features such as $q(0) > 1$, for conditions of low and high collisionality (high density/high radiative fraction) if the impurity accumulation can be avoided.

2.2.3 Development of alternative methods for ELM control/suppression in ITER and integration with scenario requirements.

This R&D programme is orientated towards the development of alternative methods for ELM control that can be applied to ITER within its baseline design or with minor modifications that could be implemented at a later stage:

- Development of cyclic plasma displacement as a technique for ELM control within the technical limitations and scenario requirements for its application in ITER

- Demonstration of ELM control by stationary modification of the edge current distribution such as: edge plasma heating and current drive, modification of edge plasma distribution function leading to changes of the bootstrap current, etc., which can be achieved by external heating methods in ITER, such as ECRH
- Development of ELM control by pellet pacing with solid low-Z pellets to minimise plasma convective losses associated with ELM control by D pellet pacing

2.2.4 Fast particle instabilities and transport effects.

The proposed R&D activities during in this area are summarised briefly below and its implementation relies mainly on work carried out under the ITPA framework (Fast Particles Group) and/or as direct contribution by the ITER members to the ITER physics R&D.

- Application of the global nonlinear codes (such as M3D, NIMROD, HMGC) for evaluation of fast particle transport in ITER due to TAE and EP modes with emphasis on the multi mode instabilities at medium to high n numbers.
- Application of the modelling codes (OFMC, ASCOT, etc.) for evaluation of the first wall power load in ITER caused by fast particle loss due to collective instabilities (TAE and EP modes). These should include the effects of toroidal magnetic field ripple, magnetic field disturbances due to test blanket modules and increase in $|\nabla P_{\alpha}|$ due to sawteeth.

2.2.5 Particle transport in ITER reference scenarios and density peaking control.

The proposed R&D activities during the years in this area are summarised briefly below and its implementation relies mainly on work carried out under the ITPA framework (Pedestal and Transport and Confinement Groups) and/or as direct contribution by the ITER members to the ITER physics R&D.

- Determination of physics processes leading to collisionality dependence of density peaking in ITER. The emphasis here would be to determine to which level collisionality is the only parameter leading to density peaking and whether or not other parameters such as edge MHD stability and ELM induced outfluxes play a role in explaining present experimental results. This could require, for instance, experiments with similar core collisionalities and different levels of edge fuelling/ELM particle outflux at different plasma currents within a single device and/or multi-machine coordinated experiments with similar core collisionalities but different normalised edge neutral fuelling/ELM particle outflux.
- Determination of requirements for density peaking control in ITER. While density peaking at ITER core plasma collisionalities can be of benefit to increase its fusion performance, it can also be a strong driver for impurity accumulation particularly for high Z elements such as W, which is foreseen for the ITER divertor target. It is therefore necessary to understand whether this peaking can be subject to gradual control in present experiments and which are the central additional heating requirements for such control. It is proposed that experiments in present devices are carried out in plasmas with high confinement (H98 =1) and low collisionality with large density peaking where the level and localisation of central heating (by ICRH and/or ECRH) is varied gradually and its effect on density peaking studied. Device size scaling to ITER could be achieved by appropriate multi-machine experiments.

2.2.6 Momentum transport in ITER reference scenarios and expected plasma rotation in ITER.

The proposed R&D activities in this area are summarised briefly below and its implementation relies mainly on work carried out under the ITPA framework (Transport and Confinement Groups) and/or as direct contribution by the ITER members to the ITER physics R&D:

- Determination of mechanisms leading to plasma rotation and improvement of the physics basis to predict the expected level of plasma rotation in ITER. This represents a continuation of on-going work and would require additional coordinated experiments in present devices in high confinement plasmas (H_{98} =1) dominantly heated by RF (ECRH and ICRH). The purpose of these experiments would be to test the capability of present theoretical models to describe the observed plasma rotation in present devices and to enlarge the experimental database to allow the further development of empirical scalings to ITER.

2.2.7 Evaluation of requirements and performance of hybrid scenario for ITER.

The proposed R&D activities in this area are summarised briefly below and its implementation relies mainly on work carried out under the ITPA framework (Integrated Scenarios, Pedestal and Transport and Confinement Groups) and/or as direct contribution by the ITER members to the ITER physics R&D:

- Determination of mechanisms leading to enhanced plasma confinement in hybrid H-modes such as: enhancement of pedestal pressure, suppression of sawteeth, non-stiffness of core profiles, etc. This represents a continuation of on-going work and would require additional coordinated experiments. The purpose of these experiments would be to test the capability of present theoretical models to describe the observed energy confinement in hybrid scenarios to allow physics-based predictions for ITER operation in this scenario.

3 Work Programme in Plasma Stability

3.1 Plasma Control

Urgent short term activities (about 1 year) are marked in red.

3.1.1 Magnetic configurations

3.1.1 Study of magnetic configurations

1) Analysis of plasma operational space in 15 MA and 17 MA scenarios for specified limits on CS and PF coils currents, field and forces, and limits on the separatrix positioning. Production of plasma equilibrium database.
2) Design of PF scenarios as sequence of plasma equilibrium snapshots.
3) Production of plasma equilibrium data for at request of IO (EQDSK files, magnetic field lines, magnetic fields in given points, etc.).

3.1.2 Generator of Plasma Equilibrium

1) Development of Generator of Plasma Equilibrium on the basis of already existing validated plasma equilibrium code (user friendly code for generation of plasma equilibrium, taking into account engineering limits imposed on the coils and accuracy of the separatrix positioning, and for calculation of magnetic field and field lines in a given regions.)
2) Work on support of Generator of Plasma Equilibrium.

3.1.3 Generator of Poloidal Field Scenarios

1) Development of Generator of Poloidal Field Scenarios on the basis of already existing validated plasma equilibrium code (user friendly code for generation of sequence of plasma equilibria for design of PF scenarios in the approximations of plasma equilibrium snapshots, taking into account engineering limits imposed on the coils an accuracy of the separatrix positioning).
2) Work on support of Generator of Poloidal Field Scenarios.

3.1.2 Plasma initiation

3.1.2.1 2D models

1) Design of scenarios of plasma initiation and self-consistent simulation of early phase of plasma current ramp-up (≈0.05 MA < I_p < 1.5 MA) with feedback control of plasma position and shape. Development of control algorithms for early phase of plasma current ramp-up. (2009)
2) Validation of models of plasma initiation.
3) Study dependence of power required for ECRF assist of plasma initiation on the value of stray magnetic field at breakdown.
4) Design of scenarios of plasma initiation with updated plasma transport and equilibrium models (e.g. inclusion of halo currents) and self-consistent simulation of early phase of plasma current ramp-up (≈0.05 MA < I_p < 1.5 MA) with feedback control of plasma

position and shape using the model of magnetic diagnostics. Development of control algorithms for early phase of plasma current ramp-up. (2010)

3.1.2.2 3D models

1) Preliminary assessment of effect of TBMs, ferromagnetic insets and vacuum vessel ports on plasma initiation and early phase of plasma current ramp-up.
2) Development of 3D electromagnetic model of tokamak, including TBMs, ferromagnetic insets and vacuum vessel ports, for correction of 2D electromagnetic model used in self-consistent simulation of early phase of plasma current ramp-up ($\approx$0.05 MA < I_p < 1.5 MA).

3.1.3 Plasma start-up and termination

1) Development of plasma transport models for transient phases of inductive scenarios (models of plasma confinement, density behaviour after H to L mode transitions, assumptions on impurity content, etc.). Validation of transport model at plasma start-up and termination.
2) Design of scenarios of plasma start-up and termination ($I_p > \approx$1.5 MA) and their self-consistent simulations with feedback control of plasma current, position and shape. Development of control algorithms. Production of database of PF scenarios.
3) Analysis of different variants of plasma start-up and termination in Scenario 4.
4) Analysis of scenarios of plasma commissioning.

3.1.4 Control of plasma current, position and shape

3.1.4.1 2D models

1) Development of specifications on plasma disturbances for simulation of plasma current position and shape control (e.g. H to L mode transition, decay of Internal Transport Barrier, ELMs, Minor Disruption, etc.) The simulations will be used for production of ITER plasma control database (maximum displacements of the plasma –wall gaps, settling time, voltage/current/magnetic field demands, input for AC losses analysis, etc.).
2) Projection on low frequency plasma noise expected in ITER diagnostics used for stabilization of plasma vertical displacements (dZ/dt, $n = 0$).
3) Development of methods of the noise reduction.
4) Analysis of stabilization of plasma vertical displacements by in-vessel coils. Development of requirements to power supply.
5) Analysis of stabilization of plasma vertical displacements by in-vessel coils with the model of power supply.
6) Design of algorithms for feedback control of plasma current, position and shape for operation in normal and abnormal regimes (with saturation of current/field/force in one coil).
7) Validation of the controllers operation during plasma disturbances (H to L mode transition, minor disruptions, ELMs, sawteeth oscillations, noise in diagnostics, etc.) in 15 MA scenarios.

8) Production of the plasma control database (maximum displacements of the plasma – wall gaps, settling time, voltage/current/magnetic field demands, data for the AS losses analysis etc.).
9) Analysis of position control of runaway electrons by in-vessel coils.

3.1.4.2 3D models

1) Development of 3D electromagnetic model of conducting structures for preliminary calculation of transfer function describing diagnostics of d*Z*/d*t*.
2) Development of code with 3D model of conducting structures (including blanket modules) and deformable plasma for simulation of plasma current, position and shape control.

3.1.5 Magnetic reconstruction of plasma boundary

3.1.5.1 2D models

1) Development of reconstruction code with 2D plasma model, without eddy currents in conducting structures. Preliminary optimisation of the set of magnetic probes and loops used for reconstruction. Assessment of the accuracy of magnetic reconstruction taking into account bootstrap current on the plasma edge.
2) Development of reconstruction code with 2D plasma model and 2D model of conducting structures). Optimisation of the set of magnetic probes and loops used for reconstruction. Assessment of the accuracy of magnetic reconstruction during transient phases of scenario (e.g. H to L mode transition).

3.1.5.2 3D models

1) Preliminary assessment of effect of ferromagnetic inserts, Test Blanket Modules, in-vessel coils and 3D eddy currents in the conducting structures on accuracy of magnetic reconstruction.
2) Preliminary development of 3D magnetic reconstruction code taking into account ferromagnetic inserts and vacuum vessel ports.
3) Development of 3D magnetic reconstruction code taking into account ferromagnetic inserts, TBMs and eddy currents induced in vacuum vessel ports and blanket modules.

3.1.6 Error fields

3.1.6.1 Error fields measurement without plasma

1) Preliminary proposals for measurements of error fields before the first plasma. Preliminary development of methodology for identification of the main sources of error fields. Assessment of: the accuracy required for the measurements, the time required for the tools installation and the cost of these measurements. (2009)
2) Preliminary design of the system for measurements of error fields before the first plasma. Preliminary analysis of the time required for the system installation and the cost of its manufacturing. (2010)
3) Detailed design of the system for measurements of error fields before the first plasma. Analysis of the time required for the system installation and the cost of its manufacturing. (2011)

3.1.6.2 Error fields measurement with plasma

1) Preliminary proposal for identification and correction of error fields at start of plasma operation using plasma response to the error fields. Preliminary development of scenarios dedicated for error fields analysis with the goal of identification of the error field structure and development of recommendations on the error field correction. (2009)
2) Detailed proposal for identification and correction of error fields at start of plasma operation using plasma response to the error fields. Development of scenarios dedicated for error fields analysis with the goal of identification of the error field structure and development of recommendations on the error field correction. (2010)

3.1.6.3 3D magnetic studies related to error fields

1) Calculation of currents in the modified error field correction coils ("Baseline 2008"), required for reduction of error fields expected in ITER to the acceptable level.
2) Assessment of deviation of the saturated magnetization of the inserts ferromagnetic material from its averaged value permissible from the point of view of the level of error fields produced by this deviation.
3) Calculation of currents in the ELM control coils, required for reduction of error fields expected in ITER to the acceptable level.
4) Calculation of error fields produced by the coils joints and terminals as they are in the design 2008 and calculation of currents in correction coils required for their correction.
5) Development of methods of defining the coils current centreline misalignments after coils manufacturing.

3.1.6.4 Error fields criterion

1) Development of criterion on tolerable error fields (low n) for operation at different plasma scenarios (in particular avoidance of locked modes at low plasma density and at high βN).

3.1.7 TF ripple

3.1.7.1 TF ripple from TF coils

1) Development of criterion which should be used for optimization of distribution of the ferromagnetic inserts.
2) Optimization of distribution of the ferromagnetic inserts with the goal of minimization of TF ripple on the plasma boundary at the nominal value of toroidal magnetic field.
3) Analysis of TF ripple produced by reduction of the current in each second coil, taking into account ferromagnetic inserts with the optimized distribution.
4) Production of the database of magnetic field from the TF coil and ferromagnetic inserts.

3.1.7.2 TF ripple from TBMs

1) Development of criterion which should be used for design of the TBM correction coils.
2) Preliminary design of the coils dedicated for reduction of magnetic field produced by test blanket modules for updated model of TBM and analysis of the performance.
3) Production of the database of magnetic field from the TBMs.

3.1.7.3 3D magnetic field calculator

1) Development of 3D magnetic filed calculator, taking into account ferromagnetic inserts and simplified magnetic model of TBMs.

3.1.8 Magnetic field from tokamak

1) Production of magnetic field maps of the Tokamak Complex with its updated magnetic model (ferromagnetic rebar).

3.1.9 RWM control

1) Projection on low frequency plasma noise expected in ITER diagnostics used for RWM control (n = 1).
2) Validation of RWM codes in feedback mode.
3) Assessment capability of ELM coils to control RWM. Development requirements to power supply (voltage, current, delay, time constant) and magnetic diagnostics (delay, time constant).
4) Study effect of the model of blanket modules on RWM control by ELM coils (model of “thick” blanket modules vs. “shell” model of the blanket modules).
5) Simulation of RWM control with the model comprising multimode plasma model, 3D model of the coils and conducting structures, and model of magnetic diagnostics.
6) On the basis of the detailed RWM model, production of simplified RWM models (transfer function) used for design of controllers and simulation of the control actions.
7) Design of RWM controllers. Optimization of magnetic diagnostics used for RWM control.

3.1.10 NTM control

1) Validation and benchmarking of NTM codes.
2) Development requirements to diagnostics used in NTM feedback control.

3.2 Disruptions

3.2.1 Urgent activities directly linked to PA (2009-2011).

3.2.1.1 Update of 2D Disruption codes and provide necessary design data and preparation of design database.

Plasma behaviours during disruptions and VDEs are essential to evaluate the electromagnetic (EM) load on the in-vessel components and the vacuum vessel due to the induced eddy and halo currents. Thus, development/update of sophisticated 2D numerical codes, like DINA and TSC, is indispensable to simulate the plasma behaviour correctly. Since EM load is a key issue for the design, it is highly desirable to provide the physics guideline with cross-checking at least two code results. So far, experimental validations and code bench-marking have been extensively done for both codes. However, model development and experimental validation of the halo current are still very limited. Since details of the halo current will have a large impact on the estimation of the EM load on the divertor cassette (PA scheduled in 2009) and blanket module (PA scheduled in 2011), following tasks and R&D must be urgently performed.

- Construction of disruption database (heat loads, data on halo current) and improvement of disruption model (in particular, description of halo width, current profile, magnitude, toroidal peaking factor).

- Update of the halo current model in the DINA or any other codes (e.g., TSC) using individual machine data, e.g., ASDEX-U, and/or ITPA new database. The purpose of this task or R&D is to provide the updated model for the halo current (e.g., halo temperature, width, current profile) to implement the updated model into the sophisticated 2D numerical codes, like DINA and TSC. For this update, halo current database taken from systematic experiments in any individual machine, like ASDEX-U, or ITPA new database should be used.

- Update of the disruption model in the DINA or any other codes (e.g., TSC) to include recent design changes. In the past two years, several design changes have been made, e.g., PF coil configuration, divertor dome geometry, installations of the in-vessel vertical control and ELM control coils. Upgrade of the disruption codes to accommodate these design changes is urgently required and is the purpose of this task.

- Calculations of various representative disruption and VDE scenarios, associated EM load and preparation of design database using the updated DINA code. This task requires calculations of disruption and VDE scenarios with the updated DINA code. The calculation must cover all of the representative scenarios. Wide range of possible scenarios must also be calculated and prepare design database for the engineering assessment.

- Calculations of selected disruption and VDE scenarios with any other codes (e.g., TSC) for cross checking of the calculation results with the DINA code. In this task, same calculations must be performed by other codes, like TSC with the updated models, for the selected scenarios to cross check the results by both codes.

- Simulations of VDE in present tokamak using the data provided (codes validation). In parallel with the simulation calculation for the machine design, code validation must be continued using the available data. Particular emphasis should be on the VDEs, since their physics process is complicated and impact to the machine is large.

- Calculation of sideway forces and their moments expected during VDE.

3.2.1.2 Preparation of physics guidelines for massive material injection for disruption and runaway electron mitigation

Disruption and runaway electron mitigation system is one of the essential machine protection systems in ITER. This mitigation system has been decided to be implemented. Massive material injection seems most promising at present for this purpose. However, detailed physics guidelines have not been specified yet, since physics basis of this technique is still in developing phase and the provisional engineering impact seems very large. Special R&D for the injection valve will be necessary. Pumping and exhaust gas processing system will be influenced substantially to achieve prompt recovery after injection (target value is ≈3 hrs to maintain operation efficiency of the machine). Design modification/upgrades to allow the implementation in ITER will become increasingly difficult as overall design is frozen.
In fact, PA for each system is scheduled in 2010 for Tritium plant, in 2011 for cryopump. In particular, technical R&D for injector needs to be started in 2009, which takes 3 years, to complete PA in 2012. Therefore, the physics basis must be confirmed urgently and the relevant

physics guideline must be specified to allow the identification of any necessary design modification as well as the resultant engineering R&D.

- Preparation of physics basis for the mitigation of runaway electrons by massive material injection. In this R&D, physics basis necessary to prepare the physics guidelines should be investigated and confirmed. Of particular importance are following issues.
    1) Identification of suppression mechanism for REs (primary, avalanche, loss) and required critical density.
    2) Assimilation or fuelling efficiency for candidate gases under various conditions (e.g., gas amount, pressure, location of gas valve).
    3) Assimilation for the case of massive solid pellet injection.
    4) Determination of optimum gas species with respect to RE suppression and higher assimilation.
    5) Localization of radiation energy deposition during radiation collapse.

- Preparation of physics guidelines for massive material injection system for disruption and runaway electron mitigation (e.g., species, amount, response time, dynamic range). This task is to prepare the physics guideline based on the previous R&D (preparation of physics basis). This also includes specifications necessary to start technology R&D for several candidate mitigation schemes based on massive material injection.

- Development of numerical code to simulate the massive material injection process. This R&D is to develop a numerical code to simulate the whole process of RE suppression by massive material injection (i.e., penetration of massive material, MHD trigger, radiation collapse, suppression of seed or avalanche REs). After simulations of RE suppression in present tokamak using the data provided (codes validation), the whole process of RE suppression by massive material injection in ITER case should be simulated. The final target of this R&D is to confirm the physics guideline of optimum material species, amount, response time for RE suppression in ITER.

3.2.1.3. Identification of the requirements for ELM control coils for runaway electron suppression by stochastic magnetic perturbation

Application of stochastic magnetic perturbation has a potential to suppress REs. The effectiveness of this method was first shown in JT-60U previously, and recently in TEXTOR. In-vessel vertical position and ELM control coils are to be installed in the present ITER and possibility to suppress REs with these coils appears. PA of in-vessel coils is scheduled in late 2010, and thus, applicability of this method for ITER must be urgently identified and required coil current and voltage for RE suppression must be specified if it is applicable. In addition, with this control method, (probably milder than massive) material injection system to mitigate other deteriorating impact of disruptions/VDEs, i.e., thermal load and EM load due to halo current.

- Preparation of physics basis for the suppression of REs by stochastic magnetic perturbation. In this R&D, physics basis necessary to prepare the physics guidelines should be investigated and confirmed. Of particular importance are following issues.
    (1) Evaluate the penetration time of stochastic field before thermal quench
    (2) Identify required mode and amplitude of stochastic field near the center

- Engineering assessment of ELM control coils for the suppression of REs by stochastic magnetic perturbation. In this R&D, engineering assessment of the existing ELM control coils should be done to identify the capability of RE suppression by the existing coils or necessary upgrade of the coils should be investigated. Of particular importance are following issues.
    (1) Evaluate the response time of ELM coils to change the perturbation mode from ELM control to RE suppression (high m/n to low m/n)
    (2) Evaluate available coil current during current quench phase before and after RE generation

- Preparation of physics guideline for ELM control coils for runaway electron suppression. In this R&D, physics guidelines to use ELM control coils for RE suppression. Of particular importance are following issues.
    (1) Required mode and amplitude of stochastic field near the center considering the penetration time of stochastic field before thermal quench.
    (2) Required voltage to achieve required response time of ELM coils to change the perturbation mode from ELM control to RE suppression (high m/n to low m/n).
    (3) Preparation of physics guideline for material injection (amount, species, response time) for the mitigation of energy load and halo current to be used in combination with ELM coils for RE suppression.

3.2.1.4 Identification of the requirements for vertical and radial position control to maintain enough time for runaway electron suppression

If position of RE plasmas can be controlled for enough time duration, several other potential control schemes could be applicable to suppress REs, e.g., heavy gas injection, impose of reverse one-turn voltage. The latter has been demonstrated in JT-60U. The former has been investigated in JT-60U and Tore-Supra, but clear conclusion has not been derived yet. For the control method using ELM control coils, the position control will also be essential.

- Requirement of vertical (and horizontal) position control capability for RE discharge for certain time periods (probably > several seconds) without strong interaction of REs with wall should be identified.
- Determination of stopping time of REs with respect to gas amount for various gas species, especially heavy gas.

3.2.2 Urgent activities not-directly linked to PA (2009-2011).

3.2.2.1 Characterization of runaway electron generation and loss

At present, specification of the generation and energy deposition of REs in ITER, e.g., generation conditions, RE current, beam and particle energy, probable locations and area of deposition, time duration, incident angle, has been derived from a very limited database and analyses and remains in a rather tentative state. Thus, improvements are urgently needed.

- Experimental investigations of characteristics of runaway electrons. In this R&D, experimental investigations of RE generation condition, characteristics and loss should be performed. Of particular importance are following issues.
    (1) Dedicated experiments for identification of REs generation conditions expected in ITER. Identify dominant parameter dependence.

(2) Conditions for generation of runaway electrons during VDEs
(3) Identification of physics mechanism behind low B_t and low q runaway threshold and implications for ITER
(4) Effect of plasma facing component material on runaway generation, with particular emphasis on the investigation of JET Be limiter/divertor, in which runaway electron generation is significantly limited.
(5) Effects of existence of supra-thermal electrons by LH on runaway generation in ITER.
(6) Disruption avoidance by ECRH as demonstrated in FTU and ASDEX-U. Derivation of the power scaling.

- Calculations of characteristics of runaway electrons by sophisticated numerical code. In this task/R&D, calculation of characteristics of runaway electrons (energy spectrum, current, beam and particle energy, their time evolutions) for major disruption of ITER by sophisticated numerical code, like ARENA or other equivalent codes, for the following various conditions;
  - Various seed RE generation mechanisms (Dreicer, hot-tail electron, tritium decay, Compton scattering)
  - Various current decay rates
  - With and w/o delay of RE generation due to density increase

Based on these calculation results, updated specification of RE characteristics and generation condition in ITER should be derived.

- Derivation of physics guidelines for energy load due to runaway electrons. In this R&D/task, following issues should be investigated experimentally or numerically. The target of this R&D/task is to derive physics guidelines for energy load in ITER. Following issues are of primary importance.

(1) Dedicated experiments on RE energy deposition and extrapolation of the experimental results to ITER.
(2) Determination of timescale for runaway loss in ITER on experimental basis
(3) Evaluation of runaway magnetic energy deposition at runaway loss.
(4) Development of model of runaway electron behaviour and characterization of their heat load expected in ITER.

3.2.2.2 3D MHD simulation of VDEs for detailed evaluation of toroidal peaking factor (TPF)

Large halo current induced during VDEs generates dominant force on the vacuum vessel, which is the primary confinement boundary of tritium and thus its robustness against these forces is of primary importance. For proper design of the support system of the vessel and in-vessel components, local vertical force must be properly estimated. In this context, the toroidal peaking factor (TPF) is essential for this estimation in addition to the amplitude of the halo current. So far, halo current database on TPF and halo current fraction f_h ($\equiv \langle I_h \rangle / Ip0$) has been used to assess the possible highest local force. Especially, the upper boundaries of f_h and TPF$\times f_h$ are simply taken from the database. For more robust prediction of the possible highest TPF and TPF$\times f_h$ in ITER, understanding of the underlying physics of TPF and 3D numerical code prediction are essential. The purpose of this R&D is to develop a 3D MHD code to simulate VDEs and predict possible highest TPF as well as TPF$\times f_h$ in ITER. The R&D consists of the following several steps.

(1) Develop 3D MHD code for the simulation of VDEs for detailed evaluation of TPF

(2) Simulations of VDE in present tokamak using the data provided (codes validation)
(3) Perform 3D MHD simulation of VDEs for detailed evaluation of TPF and identification of underlying physics
(4) Examine the relation between TPF and halo current fraction under various VDE conditions expected in ITER and derive guideline for ITER case.

4 Work Programme in Divertor and Plasma-Wall Interactions

The area of plasma-wall interactions (PWI) and edge physics is an extremely active field of research, encompassing a wide range of subject areas including both plasma physics and materials aspects and the links between them. Good progress is being made in almost all of these areas, but the extent to which responses to urgent ITER PWI physics issues can be provided on the required timescales depends, to some extent on the availability and flexibility of tokamak and other laboratory facilities. For example, the new ITER-like wall (ILW) experiment at JET can only be expected to make a scientific contribution to the various ITER PWI issues at the earliest by 2011, near the end of the period over which the R&D plan outlined here is concerned

In what follows, a number of priority PWI issues identified for the next 2-3 year period are listed. The work programme has been inspired in part by discussions and consultation between IO representatives and those from a number of facilities within the parties, as well as with members of the SOL and Divertor ITPA Topical Group which met for the 11th time in Nagasaki between 15-18 September 2008. Although some of the R&D detailed here can be implemented through ITPA or, for example, through task forces inside the various parties dedicated to the PWI area, a number of specific ITER R&D contracts in both experiment and modelling will also be required.

4.1 T-retention control

Recently, a strategy for T-retention (and dust – see below) control has been accepted into the ITER baseline [1]. Part of the strategy calls for a "good housekeeping" approach [2] in which a variety of techniques are employed to reduce the accumulated in-vessel tritium. Two of these are the use of isotope exchange and glow discharge conditioning, the latter only possible in ITER on a regular basis (e.g. inter-shot) with ICRF conditioning. Such conditioning may also be required for ease of discharge start-up. There are a number of outstanding issues relating to the use of this technique on ITER and efforts are urgently required to advance the level of understanding and ensure that it will be a viable option. In particular:

- The wetted area and efficiency of the RF conditioning plasma and techniques to increase wetted area (such as application of vertical field to improve poloidal uniformity)
- The detailed exhaust gas composition during RF conditioning
- The optimum glow gas pressure and composition
- The feasibility of using the ITER RF heating antenna for conditioning

Input is required from radio frequency experts and those involved in the application of ICRF conditioning as a routine cleaning technique. Although some of this work can be performed through ITPA, it is likely that a dedicated workshop, organised by ITER, will be required to bring the necessary plasma and RF experts together.

Isotope exchange by running D only phases at the ends of DT discharges has also been suggested as a way of removing tritium retained during burning plasma phases. Tests of this possibility on operating devices are sought, probably using hydrogen and deuterium as a proxy for deuterium and tritium. Cross-machine studies comparing metal wall machines with graphite dominated devices are required in which both spectroscopic (in plasma) and exhaust composition are examined. Comparisons between pellet and gas fuelled discharges would also be useful. A central issue is the extent to which much stronger T-fuelling will be required following an isotopic exchange if an equilibrium with the wall (depleted in T) is necessary to obtain the adequate D-T mix in the core plasma. The overall benefit of the isotope exchange could then be lost. The issue of particle control in the presence of a full beryllium wall (ie strong outgassing from locally saturated areas of the wall when the surface temperature rises) is of considerable importance with respect to the fuelling and pumping requirements during ramp-up and down and remains an as yet unpredictable aspect of ITER operation.

Modelling improvements are also required to extend the capability of the primary existing tool being used to assess ITER divertor performance (B2-Eirene) such that plasma-surface interactions are more realistically included into the code package. Such improvements include the description of mixed materials, co-deposited layers and T trapping and release (recycling).

A third option for de-tritiation is radiation flashing of wall surfaces using mitigated disruptions. Estimates show that this could be an effective technique for removal of significant quantities of wall trapped tritium in ITER, although there remain significant uncertainties as to the application of the mitigation with regard to the consequences for runaway electron generation and elevated toroidal radiation asymmetries (possibly leading to first wall melting) as a result of the gas injection. Under the assumption that disruption mitigation (for example at lower plasma currents and stored energies) does turn out to be a routine method by which transient outgassing can be accomplished, tests are required on beryllium co-deposits to investigate the efficacy of flash heating at various levels of radiation intensity up to conditions at which melt layers are formed.

If CFC were retained in the early stages of ITER operation with tritium, the T-inventory build-up is expected to be sufficiently rapid that only a limited number of full performance discharges (of order several 100, though this number is associated with large uncertainty) could be executed before inventory limits would force cessation of operations [2]. Subsequent removal of tritium accumulated in divertor co-deposits (where the majority is expected to reside) could be accomplished in-situ by oxidation, allowing an extension of operation in CFC. Further research is, however, required to establish the viability of oxygen baking as an efficient removal technique and to properly characterise the collateral effects for in-vessel infrastructure.

Carbon deposition on main chamber surfaces during the first operation phases in H and D is a potential source of T-retention if operations begin with a full W divertor in the DT phase and isotope exchange occurs between T and the H/D trapped in carbon films. The importance of this effect may increase if the T-free operational period with a CFC divertor increases. Experiments are sought to improve understanding in this area. Such tests could involve the introduction of samples bearing C films pre-loaded with deuterium followed by prolonged exposure to hydrogen discharges (short H-campaigns are often executed by tokamaks at the end of operational phases in D). Such samples could also be designed to contain gap structures mimicking ITER first wall PFCs. Coordinated experiments of this nature could be performed in both tokamaks (limiter and divertor machines) and plasma simulators. Many of these devices are already equipped with suitable manipulators for sample introduction.

4.2 Tungsten R&D

The current ITER Baseline calls for installation of a full W divertor before the tritium phase. For this to be possible on the basis of the current ITER construction schedule, the required R&D technology programme would need to be complete around 2013 and the Procurement Arrangement issued in early 2016, coincident with the completion of construction of the first (CFC/W) divertor set [3]. A corresponding physics programme needs to accompany the technology drive to assess the scientific viability of operation with a full W divertor. Such physics R&D encompasses a wide range of issues of which the following refer specifically to the PWI area and require that efforts intensify now in order for sufficient progress to be made before the time at which decisions will be required to proceed with the fabrication of full W divertor plasma-facing components (PFCs).

- In the absence of carbon, a strong natural radiator at divertor plasma temperatures, it is unlikely that the high radiation levels required in ITER to maintain the partially detached state during fusion burn can be obtained without the addition of extrinsic impurities such as neon or argon (~70% of power entering the divertor must be radiated). Demonstrations are required that power loads and erosion can be controlled in the presence of high Z targets under high power operating regimes. In addition to experiments, a strong edge modelling programme including W and W/Be material mixes must start to accelerate, particularly the benchmarking against experimental results as they arrive. The vast majority of plasma-surface interaction modelling to-date has been performed for carbon. The more challenging problem of W requires urgent attention if the situation is to achieve the maturity of that which has been obtained for carbon PFCs. Most ITER modelling to-date has been performed with the SOLPS package. Efforts are underway to include W modelling capability within this package, but the use of alternative modelling tools applied to ITER (e.g. EDGE2D-Eirene) including W would be of considerable interest.
- In addition to flat top conditions, evaluation is also required of the operational range for ohmic and heated X-point ramp-up/ramp-down with high Z divertor targets including impurity seeding for sputtering control.
- Assessment of impurity production by use of ICRH with high-Z PFCs and the role of local versus remote sources (i.e. those due to sputtering on field lines connecting from antenna shields to the high Z divertor targets). The importance of the low Z impurity mix and ionisation stage in front of the RF antenna remains to be assessed, as do the improvements, currently being modelled, to antenna design to reduce radial electric fields thought to be responsible for the generation of sheath rectified voltages which provoke increased sputtering.
- Evaluation of the rate at which W impurities can be flushed from the pedestal under the action of ELMs compared with the influx provoked by the ELM-enhanced erosion as a function of ELM frequency, energy etc. The influence of these ELM-induced impurity influxes on plasma performance, particularly for conditions equivalent to controlled ELMs in ITER (production by physical sputtering alone with high T_{ped}, low ν^*_{ped}) is also an area requiring attention.
- Evaluation of the effect of divertor target damage on device operability and operational space. This could take the form of experiments in devices with small sets of W divertor

tiles that have been pre-damaged in e-beam or plasma-gun facilities with ITER relevant loads or in cases in which accidental damage has already occurred or has been deliberately induced. It could also involve deliberate removal of small regions of PFC surfaces (for example in the divertor in areas where strike points are not normally placed during routine operation but where they could be located for the purposes of short term tests to study the consequences of PFC failure), mimicking failure of some part of the ITER divertor monoblock W targets. Is plasma operation possible under such circumstances and what are the consequences for the neighbouring PFCs?

- The recent observation of growth of W nanostructures under hydrogenic plasma bombardment in the presence of helium fluxes at high surface temperatures [4] is a potential source of increased erosion in the divertor strike point regions. Further experiments are required to understand the process and to investigate the effects of adding seeding impurities such as neon and argon. The problem of blister generation on W surfaces under He and hydrogenic plasma bombardment at high fluences is also a possible source of enhanced erosion and requires further study.
- Evaluation of the permeability of tritium through tungsten at elevated temperatures (characteristic of the divertor strike point regions), and the effect of neutron irradiation (increased trap sites) on T-retention. This has implications both for safety (permeation of T into the cooling circuit) and the levels of T trapped in W.
- Assessment of plasma-wall interaction behaviour and power handling properties of mixed-material surface PFCs under steady-state plasmas and for ITER-like transient loads corresponding to controlled and uncontrolled events. This should include the effects of mixed-species plasma impact (D, T, Be, He, Ar, Ne) on material properties and erosion, a large number of cycles for controlled events, the synergistic effects of steady-state and transient plasma fluxes and relevant surface temperatures.

4.3 Dust

Like T-retention, a strategy to deal with in-vessel dust accumulation has recently been accepted into the ITER baseline. Amongst a number of measures, this strategy calls for the accelerated development of a dust generation rate/transport model such that greater confidence can be assigned to predictions for DT operation in ITER. A significant effort is required in the next few years to improve understanding of the main mechanisms of dust production, both in steady state and during ITER-relevant transients for all materials (C, Be, W).

The assumption in building the safety case for dust [1] has been that the majority will be created by the disintegration of thick co-deposited layers put down in the divertor (C and Be) due to simple growth beyond thicknesses which can remain mechanically intact or under the action of transient loads. In the case of pure W, the majority of dust generation is expected due to melt layer formation under extreme loads or grain release due to surface cracking. For Be, the database is extremely limited, but, in common with W, most is expected to be produced by cracking and droplet formation under transient loads. Once released, the subsequent dust transport/destruction in the plasma is an important aspect of the full dust model. The conversion factor (the fraction of the gross erosion converted into dust) is an important coefficient in which further confidence is required before ITER operation begins. In the next 2-3 years it is important that progress be made in as many as possible of the areas listed below.

- Dust production rates from solid Be and deposited Be layers of thicknesses expected in ITER on both W and CFC substrates under heat loads appropriate to disruptive and ELM driven transients (both for controlled and uncontrolled ELMs and for natural and mitigated disruptions). Characterisation of ejection velocities and sizes of molten droplets and the morphology and size distributions of dust collected following transient heat loads. Benchmark of dust size distributions (and if possible velocities) using numerical simulation of generation process. Link the extent of material damage to the quantity of collected dust to recover a conversion factor for transient processes. Characterise the difference between dust originating from layer disintegration or that evolved from pure material surfaces.
- Exposure of tokamak generated deposits (carbon in the short term before JET operation with the ILW) to ITER relevant transient heat loads and analysis of generated dust.
- Further efforts on tokamaks to attempt systematic dust collection from as many locations as possible in-vessel following operational campaigns. Link gross erosion measurements with dust quantities to establish with the lowest possible uncertainty the conversion factor from erosion to dust production following plasma transport. Comparison of low and high Z devices to establish principal dust creation mechanisms. Cross-machine efforts to establish ability to correlate spectroscopic measurement of impurity influx with divertor deposition.
- Cross-machine studies of deliberate dust injection to investigate dust launch velocities and subsequent transport. Benchmarking against dust transport models.
- Study response of deposited layers to flash heating (as in mitigated disruptions) – such deposits may accumulate, for example, on the W divertor dome and will receive intense radiation loads during mitigated (and even natural) disruptions.

4.4 Heat fluxes to plasma-facing surfaces

Revised heat load specifications, based on extrapolations from results obtained in operating devices, for SOL and divertor parallel power fluxes during steady state and transients have recently been accepted into the ITER Baseline [5]. It is now therefore recognised that in addition to transient pulses due to ELMs, steady heat loads will impinge on the main chamber wall surfaces as a consequence of far scrape-off layer (SOL) tails in the density and temperature profiles driven by turbulent convective transport. The result of this recognition is a decision to incorporate shaping of all first wall elements to maximise power handling.

In the divertor, the issue of transient behaviour not linked to disruptions or ELMs has also been raised during the heat load specification exercise. Longer timescale transients (i.e. durations of seconds and not milliseconds), which exceed the steady state heat exhaust capability of the active cooling by even factors of 3-4 cannot be tolerated for more than a few seconds at most [6]. On the question of the short timescale transients, outstanding issues remain concerning controlled ELM variability and first wall loading during mitigated disruptions. In order to improve confidence in the new heat load specifications as the first wall design is finalised (next 1-2 years), further studies are requested in tokamaks which address the following areas:

- Cross machine comparisons of ELM statistics, preferably but not exclusively for controlled ELMs (waiting times between ELMs, poloidal distribution and sizes of ELM wall footprints), for evaluation of natural variability in wall and divertor heat and particle fluxes. Continuation of efforts to scale ELM radial velocity with ELM size, device size and other dimensional/dimensionless pedestal parameters.

- Further measurements of inter-ELM far SOL cross-field heat and particle fluxes and effective convective velocities – much of the existing database exists only for ohmic or L-mode discharges. Continued pursuit of SOL turbulence modelling (extension to 3D) towards predictive capability for ITER. Efforts are also being sought to include more realistic cross-field transport into the SOLPS5 code package – calculations of wall erosion rates and recycling can be significantly in error if the large amplitude bursts of temperature and particle fluxes arriving at the wall are not accounted for.
- Improved characterisation of divertor power loading during ELMs mitigated or suppressed by resonant magnetic perturbations.
- Evaluation and extrapolation to ITER of the degree of toroidal asymmetry in the intense radiation flash during mitigated disruptions in the case of a single injection point. If the asymmetries are sufficiently strong when the majority of energy stored at the thermal quench is radiated, severe local damage could be incurred at the first wall.
- Begin studies of transient divertor reattachment, particularly in response to loss of gas or extrinsic impurity seeding or due, for example to confinement transients such as H-L transitions, ITB collapses etc. Important considerations are the timescales for reattachment and the distributions/magnitudes of divertor radiation.
- Further investigation of SOL limiter profiles during start up and ramp-down in ITER relevant scenarios to verify the assumptions made in estimating characteristic lengths during limiter phases in ITER. Tests of the model used to make these estimates are of particular interest (which includes scenarios with variable numbers of limiters for inner or outer wall start-up).
- Even though considerable progress has recently been made, consensus is still far from clear as to what physics determines energy transport in the near-SOL and thus what is the correct power e-folding length scaling for extrapolation to ITER. The use of hydrogen plasmas to clarify the isotope scaling of λ_p and coordinated cross-machine studies in high power regimes are possible areas where further efforts could be concentrated.
- It is also important to emphasise that partially detached operation is mandatory for the ITER divertor to survive the steady state power loads that arrive parallel to the field in the separatrix (strike point) vicinity. Obtaining such detachment numerically in ITER is straightforward using the SOLPS code package and indeed detachment is routinely observed in most of today’s tokamaks. However, the code systematically fails to reproduce the detachment seen in experiments when constrained by boundary conditions imposed by the experimental measurements. It is vitally important that this discrepancy be resolved, but this is not reflected by the amplitude of the effort expended in this direction by the research community.

4.5 Erosion

The ITER first wall components will be shaped, to avoid plasma exposure of PFC edges, permitting steady state and transient heat fluxes to be tolerated and to open the possibility of using the first wall PFCs as limiters during the initial and final phases of ITER discharges. The latter is routine in most of the world's operating tokamaks, but until recently has not been considered for ITER. Shaping of components automatically leads to regions of erosion and re-deposition since shadowed regions are created where impurities, once migrated, are no longer exposed to plasma ion fluxes. In the case of carbon dominated machines, chemical erosion by neutral fluxes (which do not follow field lines) adds to neutral physical sputtering such that much of the first wall can be a region of net erosion (with most impurities being convected into the inner divertor by strong SOL parallel flows). For a full Be wall, it is likely that significant net-deposition regions will persist, accompanied by T-retention and the possibility, if layers grow to sufficient thickness, of spalling and associated dust production, especially during transients (notably disruptions). The sheer size of the first wall in comparison to the divertor means that even if local deposition rates are much lower (as expected) than in the divertor, the integrated effect could be significant. It is thus important that efforts begin in operating devices to investigate the effect of first wall shaping on local erosion and redeposition:

- Cross machine comparisons of main wall co-deposition on limiting surfaces
- Development of local models taking into account surface shaping and attempting to predict the observed deposition patterns for both steady (diverted operation) and during limiter start-up/ramp down phases
- Estimates of the quantities of tritium that could be trapped in these first wall deposits
- Efforts should also continue to characterise outer and inner divertor erosion, the movement of impurities from main chamber to divertor and from divertor to divertor, including enhanced efforts to include a full description of drift physics in the modelling packages and benchmark against experiment.

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5 Work Programme in Plasma Operations

The ITER Plasma Operations Physics Workplan includes Plasma Operations, Heating and Current Drive physics, Energetic Particle physics, and Test Blanket Module (TBM) effects on plasma operations. The scope includes defining and developing detailed operational scenarios for ITER plasma commissioning and operations, determining operational performance requirements for ITER plasma scenarios, defining performance specifications for the ITER Plasma Control System, analyzing heating and current drive performance, energetic particle physics, and effects of TBMs on ITER plasma scenarios, collaborating on experiments outside ITER and analyzing their implications for ITER plasma scenarios, and coordinating integrated plasma modelling of ITER operational scenarios.

The results from ongoing R&D, which will be performed mainly by the ITER Parties through the International Tokamak Physics Activity (ITPA), will contribute to specifications in procurement arrangements for many ITER subsystems including PF coils and control systems (to be signed in Nov'08), pellet injection system (May 2010), gas injection system (July 2010), ICRH (April-December 2009), ECRH (March 2010-May 2011), NBI (September 2009-April 2010), ferromagnetic inserts (PA together with VV), ELM + internal vertical stabilization coils (September 2010), and first wall (July 2011).

5.1 Plasma Operations

5.1.1 Develop non-active phase scenarios

The first phase of ITER operation will be in hydrogen and helium to avoid nuclear activation of the surrounding components while bringing the machine up to full technical performance at I_p = 15 MA and B_T = 5.3 T. The operational scenarios for this phase need to be developed in detail including the breakdown, current ramp up, vertical stability, internal inductance and shape control, burnthrough requirements, wall conditioning schemes, heating and current drive, fuelling, MHD control, TF and TBM ripple effects on thermal and fast particle (NBI, ICRF driven) confinement, disruption/runaway mitigation, and current ramp down. This early phase of ITER operation presents particular issues because H-mode is expected to be difficult to achieve and maintain with the power level that will be available at the beginning of operation, based on empirical H-mode threshold scalings. An improved understanding of the physics of the H-mode threshold is critically important now because of its potential impact on the development of the ITER Research Plan. Since the H-mode threshold is expected to be significantly lower in helium than in hydrogen, the non-active phase will attempt to achieve H-mode in helium so that H-mode physics studies can begin in this early phase of ITER operation. Early H-mode physics studies are important for understanding the size scaling of the H-mode confinement and threshold, the impact of ELMs on the first wall and divertor, impurity accumulation, pedestal height and width, and intrinsic plasma rotation.

Experiments on existing machines, mainly performed as joint ITPA experiments, continue to provide a basis for developing ITER operational scenarios in most of these areas. Low voltage startup scenarios with ECH assisted breakdown and/or early heating in the current rise for burnthrough and l_i control have been performed on ASDEX Upgrade, C-Mod, DIII-D, and JET. Other experiments to optimize the current rampdown in an ITER-like scenario maintaining a diverted plasma as long as possible and attempting to reduce flux consumption are needed to further develop ITER scenarios. Developing wall conditioning schemes that can be used during the discharge in real-time so that they can be used on ITER is a challenge that still needs to be addressed in existing experiments. Scaling of disruption/runaway mitigation techniques to ITER still needs to be determined.

Modeling of the existing experimental results together with benchmarking between different codes is essential to predict integrated scenarios in ITER. Different transport and plasma control schemes in the different modeling codes such as DINA and TSC need to be compared for a range of ITER operational scenarios including the effects of transient changes such as L-H transitions, ELMs, monster sawteeth, ITB collapses, and H-L transitions. Flux consumption, shape control, vertical stability including PF coil limits, and l_i control with heating requirements also need to be compared with the different modeling codes from the start of the discharge through the flattop and the current ramp down to ensure consistent integrated scenarios. Modeling of the ITER fuelling schemes needs to be compared with experiments to ensure the expected density profiles can be obtained with the specified fuelling rates. Disruption/runaway mitigation techniques on present experiments need to be correctly modelled and extrapolated to ITER.

5.1.2 Develop deuterium phase scenarios

The second phase of ITER operation will be with deuterium plasmas and because of the high performance and relatively long pulses, the neutron fluence will soon activate the vessel so that remote handling of in-vessel components will be essential. This first nuclear active phase is expected to have somewhat easier H-mode access so that the initial H-mode physics studies in the non-active phase should be more robust in the deuterium phase. So, there will be some differences in the scenario development between the non-active and deuterium phases. Heating and current drive schemes will need to change taking into account the change in H-mode threshold and confinement. Specific commissioning procedures for the nuclear phase need to be developed for deuterium operation.

Modeling of deuterium scenarios must take into account the improved H-mode access and improved confinement. In this phase, ELMs are expected and the improved confinement could lead to other MHD activity that will need to be included in the modeling for deuterium scenarios. The effects of symmetric TF ripple and asymmetric ripple from the TBMs on thermal confinement in H-mode as well as on fast particle confinement need to be correctly modelled.

5.1.3 Develop DT phase scenarios

Following successful deuterium operation, ITER will then enter the DT phase where full fusion performance can be achieved. The standard 15 MA, Q=10 DT scenarios need to be modelled over a range of parameters with different heating schemes including transients. Extrapolations from existing experiments in the hybrid regime with $q(0) > 1$ need to be modelled over a range of scenarios to ITER. The optimum mix of heating and current drive schemes needs to be determined through modelling and comparison with experiments across a range of DT scenarios. Sawtooth stabilization due to energetic particles needs to be included in the modelling. ELMs with particularly large transient heat fluxes and ELM mitigation techniques must be included in the modelling and compared with ITER demonstration discharges from present experiments. NTM control schemes with ECCD and the requirements for power level, phasing with the mode, and radial localization must be developed. The requirements for real-time diagnostics to measure the size, phase, and location of NTM islands and the q profile need to be determined. Burn control techniques need to be developed through modelling of density and temperature profile changes, fuel dilution, and Alfven eigenmode control. Particularly important for DT scenarios is the modelling of energetic particle losses due to symmetric TF ripple and asymmetric TBM ripple.

5.1.4 Specify kinetic control requirements and functionalities for plasma operation

The detailed requirements for kinetic control of a burning plasma need to be determined through both a comparison with more basic control systems on existing experiments and through modelling of kinetic control systems. Pellet injection and gas fuelling of existing experiments and particle confinement measurements on existing devices can be used to attempt to extrapolate fuelling control systems to control the density profile in ITER. It may also be possible to test Alfven eigenmode control on existing devices through attempts to control the growth rate of AEs excited by NBI or ICRF driven fast ion tails. Modelling is then required to extend the control to a burning plasma regime dominated by α heating where the self-heating is sensitive to the fuel mixture and dilution with other species. Simulations of burning plasmas can only be done through computer modelling and can only be validated with DT plasmas on ITER. Burn control schemes thus depend on accurate and detailed realistic burning plasma models with appropriate thermal and energetic particle transport processes. So, a broad range of burning plasma control scenarios must be developed with different transport models to gain an understanding of the range of control issues that may arise in DT burning plasmas in ITER.

5.1.5 Definition of ITER plasma operation

The range of operational scenarios that will be developed with modelling codes will then form the basis for defining the initial operation on ITER. This will include initial breakdown parameters to be used to startup ITER plasmas and the development of plasma current ramp up and down scenarios for first plasma operation. Detailed assessments of the PF coil performance and vertical stability will be used to program initial plasma operation. These scenarios will also be used in liaison with other ITER departments to identify requirements for ITER operation such as real-time diagnostics, fuelling sources, PF control systems, kinetic control systems, and heating control systems.

5.1.6 Definition of physics requirements for ITER operation

The physics requirements for successful operation of ITER scenarios to meet the goals of the ITER program must be defined as well as possible within the limits of physics understanding available. Much R&D effort is expended to improve the physics understanding of a number of areas that are critical to ITER operation. The physics of the H-mode threshold needs to be better characterized to extrapolate to ITER and reduce the uncertainties in the power required to reach the required H-mode confinement level with H98y2 > 1. What are the PF coil voltage and current limits and additional heating levels required to achieve vertical stability, shape control, and l_i control for each of the ITER operational scenarios? What real-time wall conditioning techniques can be employed in ITER to keep impurities manageable and achieve the required confinement for each of the operational scenarios? What are the fuelling system requirements to control the density profile for each operational scenario? What is required to be able to extrapolate the hybrid scenarios found on existing machines to a hybrid regime with q>1 across the profile on ITER? What is required to be able to extrapolate even higher q steady-state scenarios with > 50% bootstrap current on ITER? A range of 17 MA scenarios also needs to be developed for ITER to provide higher confinement than the standard 15 MA scenarios to enhance Q beyond what can be achieved at the standard plasma current level. Burn control systems must be defined for ITER operation within the limits of available physics understanding based on density profile and species control and through the control of the growth of Alfven eigenmodes. The effects of ELMs on plasma operation must be defined as well as possible and ELM mitigation control systems must be defined. The effects of disruption electromagnetic and thermal loads on plasma operation must be defined and taken into account to setup disruption/runaway mitigation control systems. The assessment of impurities on plasma operation must be defined for all envisaged first wall materials and each operational scenario.

5.1.7 Definition of diagnostic specifications for real-time control systems

An area of common interest with the Diagnostics Division is the definition of diagnostic specifications for real-time control systems that are needed for plasma operations. Some of these specifications include the spatial and time resolution required for measurements of the q profile and of the NTM island width, phase, and location for NTM stabilization. How accurately must the position of various diagnostics such as the poloidal field pick-up coils be known in order to control the plasma shape and position within the required limits for plasma operation? What are the time and spatial resolutions required to measure Alfven eigenmodes accurately enough to use them for burn control systems? While detailed calculations and modelling may be sufficient to specify many of the diagnostics required for plasma operation, others require dedicated experiments to test for example the sensitivity of certain diagnostics to noise or RF pick-up on existing machines

5.1.8 Development of the ITER Research Plan

The first version of the ITER Research Plan has been developed jointly with ITER Organization staff and a group of external experts from fusion laboratories throughout the world. This first version has been useful at focussing attention on a broad range of required R&D in support of the ITER program during the construction phase. It has also made clear that there are still a number of gaps in the ITER Research Plan and some inconsistencies with the ITER Machine Operation Plan that must be addressed in the future versions of the Plan. The R&D required for the TBM program needs to be integrated within the ITER Research Plan. This will include modelling of the electromagnetic effects of the TBMs on stray fields for breakdown and on the interaction of the 3D asymmetric ripple induced by TBMs with the confinement of both thermal and energetic particles. Differences between the Machine Operation Plan and the ITER Research Plan need to be reconciled regarding the expected number of pulses per day and the number and duration of operational phases of ITER. This work will need to be carried out between the various departments within the ITER Organization as well as with the help of a number of outside experts.

5.2 Heating and Current Drive

5.2.1 Neutral beam code benchmarking

This task involves benchmarking neutral beam codes including the latest beam ionization model for a range of expected plasma conditions on ITER. Orbit effects need to be assessed by comparison of a Fokker-Planck code and a Monte-Carlo code. The orbit following scheme must be modelled in a background plasma that varies on a time scale comparable to a fast ion slowing down time for NB ions. This benchmarking will be done as part of the Integrated Operation Scenarios ITPA group effort.

5.2.2 ECH/ECCD performance assessment

The performance of the electron cyclotron heating and current drive systems on ITER must be assessed for breakdown assist, burnthrough, and all current rise, flattop, and current decay scenarios. How much power and what ECH parameters are required to achieve sustained breakdown across a range of expected error fields for low voltage startup scenarios in the range of 0.3 V/m? Similarly, what power level and ECH parameters are required for burnthough to sustain the current rise across a range of impurity levels that might be expected for different first wall materials? At what power level and in what density range will the early ECH generate substantial impurities by burning first wall components or possibly damage in-vessel diagnostics? Once the X-point has formed, how much ECH is required to control l_i within acceptable limits for vertical stability and reduce volt-s consumption sufficiently to obtain 400 s flattop burn phases? During the flattop, how much ECH will be required to obtain and maintain H-mode with high confinement? What are the ECCD spatial localization, phasing, modulation frequency and depth, and power levels required to effectively stabilize NTMs over a range of ITER operational scenarios? How much ECH will be required in the current rampdown to keep the plasma in H-mode long enough to maintain l_i control within an acceptable range for vertical stability and to reduce volt-s consumption sufficiently to control the rampdown after a 400 s burn phase? Some of these assessments can be made with experiments on existing machines while others require detailed modeling of a range of ITER scenarios.

5.2.3 ICRH/ICCD performance assessment

The performance of the ion cyclotron resonance heating and current drive systems on ITER must be assessed primarily for heating in the current rise and flattop. Very early ICRH might be possible, but because the antenna should closely match the plasma boundary to provide good coupling, it may not be possible to maintain such good coupling throughout the very early phase of the discharge before X-point formation. Coupling studies of ICRF antennas need to be modeled and compared with experiment to determine how to extrapolate and improve the coupling for ITER. Given the large uncertainty in the H-mode threshold and the rather high predictions of the latest empirical scalings, it will be important for all of the heating systems on ITER to perform at or very near their expected power levels. So, it is essential that coupling studies be carried out as soon as possible so as to be able to make changes to the ICRF antenna system design before the procurement is finalized.

5.2.4 LH/ LHCD performance assessment

While the baseline ITER design does not include a lower hybrid heating and current drive system, such a system could be included as a power upgrade. So, the potential merits of such an upgrade need to be assessed soon to be able to decide on which heating scheme or schemes should be upgraded to provide more power. Lower hybrid is particularly efficient at off-axis current drive with a broad radial profile, which would be very valuable for hybrid and steady-state scenarios. Lower hybrid heating could also be applied early in the current rise to reduce volt-s consumption. However, close coupling to the antenna in the ITER environment may be problematic. Comparisons need to be made with lower hybrid results on existing machines and with models of lower hybrid heating and current drive to attempt to extrapolate results to ITER.

5.2.5 Basis for choice for heating and current drive upgrades

If the latest H-mode threshold empirical scalings are approximately correct, the power installed on ITER for first plasma will need to be rapidly upgraded to substantially higher power levels in order for ITER to achieve sustained H-modes in the early phases of ITER operation. The basis for the choice of which heating system(s) should be adopted for an upgrade needs to be assessed for a broad range of ITER operational scenarios. The need for off-axis current drive in advanced hybrid and steady-state scenarios may preclude additional NBI for an upgrade. Enhanced NBI power at 1 MeV also increases the risk of driving unstable Alfvén eigenmodes that could eject fast ions in a beam that could damage the first wall and reduce the fusion burn. Central heating from NBI or additional ICRF could also lead to monster sawteeth that could significantly reduce the fusion burn in the core and lead to NTMs. Lower hybrid current drive off-axis may be preferable for increased current drive efficiency and to increase the bootstrap current, but lower hybrid coupling issues need to be assessed across a broad range of ITER operational scenarios. ICRF also has similar antenna coupling issues, particularly for early current rise heating where the plasma shape is changing and may be more difficult to control than in the flattop phase. ECRH appears to be the best choice because coupling should not be an issue, but the current drive efficiency is substantially lower than LHCD. All of these issues and even more technology considerations need to be assessed to determine the choice for future heating and current drive upgrades.

5.3 Energetic Particles

5.3.1 Energetic particle confinement

The effects of energetic particles on plasma operation will be substantial. Even on existing devices, the energetic particles from ICRF and NBI excite unstable Alfvén Eigenmodes (AEs). With α heating in the plasma core as well, the stability of intermediate toroidal mode number AEs may become of critical importance for plasma operation both for controlling the fusion burn as well as because these modes could enhance the losses of fast ions that could severely damage the first wall. The effects of symmetric toroidal field ripple on fast ion losses need to be calculated in detail taking the particles all the way out to the wall to determine accurate heat loads on the first wall and where damage may be expected. Ferromagnetic inserts that can substantially reduce the TF ripple effects must also be accurately modelled to determine how much ferromagnetic material is required and where it needs to be placed to reduce the effects of AE instabilities. In addition, 3D magnetic perturbation effects to the plasma equilibrium due to asymmetric ripple from the TBMs need to be accurately calculated to decide on the final size and shape of ferromagnetic material within the TBMs. The issues associated with TBMs will be covered in the next section. The proposed R&D activities in the energetic particles area are briefly summarized below.

5.3.2 Effect of magnetic field ripple on energetic particle losses

Predictions of fast particle losses caused by magnetic field ripple in the presence of optimized ferritic inserts with and without TBMs and an accurate evaluation of the resulting first wall heat load for a range of ITER scenarios are essential for successful ITER operation. A number of Orbit Following Monte-Carlo (OFMC) codes already exist and some of them are capable of describing 3D vacuum field effects of localised ripple. Benchmarking of the codes and calibration against experimental data are necessary and the inclusion of 3D plasma equilibrium effects may also be important.

5.3.3 Effect of collective instabilities on energetic particle losses

Collective instabilities including those excited by energetic particles such as AEs and Energetic Particle Modes (EPMs) can expel energetic particles from the plasma limiting achievable fast particle pressure and increasing the first wall power load. Since experiments show that such instabilities often have a bursting character in time and rapid chirping in frequency, non-linear codes are necessary to fully describe these modes. For ITER, the emphasis should be kept on multi-mode instabilities with medium to high toroidal mode numbers (n ~ 5-20). These codes should be supplemented by the OFMC calculations to evaluate the first wall power load distribution.

5.3.4 Interaction of fast ions with background MHD

Effects of the interaction of background MHD fluctuations with AEs on fast particle redistribution and losses should be evaluated for ITER. It will require studies of such effects in present day devices and comparison with theory.

5.4 TBM effects on plasma operations

A major field of concern requiring urgent R&D is the use of ferromagnetic material for the structure of the TBMs. Because of this, the presence of TBMs in the 3 allocated ITER ports may have significant effects on plasma performance, as a consequence of an increased level of the magnetic field ripples, especially close to the separatrix.

The main expected effects are the following: i) an increase of ion losses and consequent potential local increase of heat loads on the TBMs, their frames, and the surrounding shield blanket modules; ii) a negative impact on the quality of the plasma confinement during H-mode operation. An additional item to be studied is to check if it is required to have the TBM first wall (FW) protected with Beryllium (that it will require to start specific R&D to manufacture it since Be-protection is not acceptable for DEMO) or if it is acceptable to have row steel FW or W-protected FW (expected to be used in DEMO).

In order to be able to predict the level of these effects and to determine if they can be accommodated in ITER possibly using some specific countermeasures, several studies, both theoretical and experimental, have to be urgently performed.

The overall objective is to guarantee that TBM testing will be possible in ITER and to identify which are the required conditions and limitations for achieving such an objective.

The main R&D items to be addressed are the following:

1) Quantification of the effects as a function of the TBM positions, shape and size (i.e., parametric studies).
2) Effectiveness of addition of correction coils around the TBM and required operating parameters (e.g., number of coils, current).

TBM Systems are not under any ITER Procurement Arrangement since they will be developed, manufactured and delivered to ITER by the Parties in the framework of their domestic fusion programs. However, these activities are very urgent since they can have significant consequences on the TBM designs and on TBM frame designs.

It is necessary to have the final TBM conceptual design in 2010 and the final TBM detailed design in 2012, in order to allow sufficient time for manufacturing and acceptance tests and then to comply with the request of delivering the TBM Systems on the ITER site by the end of 2016 (linked to the assumed beginning of ITER operation in July 2018).

5.4.1 Stray field effects on breakdown and plasma equilibrium

The TBMs are presently being designed with two large rectangular ferromagnetic blocks that are vertically oriented next to one another in each of three horizontal ports all on the same side of the torus toroidally. This naturally leads to a rather large n=1 component to the resulting stray field, which could couple well to m=2, n=1 locked modes that could lead to disruptions.

Since the resulting stray field is largest near the outboard midplane, the effects on stray fields affecting the null near the center of the machine, required to get a sustained breakdown, may not be severe. However, even a few mT of stray field in the core where the null is desired for breakdown could make it much more difficult to obtain reliable startup conditions. So, the effects of stray fields due to TBMs on the breakdown region near the plasma core need to be evaluated.

This Design Task covers the following aspects:

- A - Assessment of the effects of 6 TBMs (using designs and masses furnished by IO) using a detailed and well-representative 3D models on the magnetic field distribution in the plasma region and of the TBM-induced ripples in the region of plasma separatrix as a function of the dimensions of the TBMs (height, width, thickness) and of the position in relation to the Shield Blanket Alignment (recession of 50mm, 100mm, 150mm, 200mm, 250mm)
- B - Assessment of the effectiveness of correction coils in correcting the induced ripples down to a level of 0.3% as a function of the number of coils (one central or three), of the level of current in each coil, and of the position of each coil (aligned with the TBM FW, recessed of 15cm, 30cm and 40 cm).

<u>Expected Outcomes:</u>

a) Confirmation of the induced level of ripples assuming appropriate specifications to be used as input for other assessments;
b) Under the same specifications, estimating the needed current in the correction coils to be used as input for the correction coils design.

<u>Time schedule</u>: Preliminary assessment by June 2009; Detailed Assessment by June 2010.

5.4.2 Asymmetric TF ripple effects on fast ion losses

The TBMs produce a toroidally asymmetric perturbation on the toroidal field which is calculated to substantially increase fast ion losses. These increased losses need to be accurately quantified with a sensitivity study to the size of the TBMs and their proximity to the plasma. This requires a fully 3D magnetic perturbation code including a 3D plasma equilibrium combined with an energetic ion orbit following code. Such codes are still being developed and this work is presently being carried out through the ITPA.

This Design Task will assess the effects of 6 TBMs (using designs and masses furnished by IO) on the fast particle losses and consequent heat load distribution in the surrounding areas as a function of the dimensions of the TBMs (height, width, thickness) and of the position in relation to the Shield Blanket Alignment (recession of 50mm, 100mm, 150mm, 200mm, 250mm).

<u>Expected Outcomes:</u>

Starting from the initial data on induced ripples obtained from item 15 (reference data), it is expected to have an estimation, using appropriate models, of fast particle losses and consequent additional heat load distributions.

<u>Time schedule</u>: Preliminary assessment by the end of 2009, then more detailed analyses in the following years with possible integration of experimental results (if any).

5.4.3 Asymmetric TF ripple effects on thermal confinement and transport barriers

Symmetric ripple experiments on JET have shown that at least at low collisionality, there is a significant reduction in thermal energy confinement with increased TF ripple even as low as 0.4%. This leads to the concern that increased ripple due to the ferromagnetic TBMs could also reduce thermal confinement in H-mode and reduce the pedestal pressure. The effects on thermal transport due to an asymmetric ripple cannot be calculated reliably. The best way of assessing the perturbations created by TBM ferromagnetic material on plasma operations is to perform appropriate experiments in present-day machines. Such an experiment should be able to simulate the local ripple effect due to a local magnetic field having similar characteristics as those induced by the TBMs. The experiments could be performed by installing ferromagnetic inserts within the vessel and verifying the impact of the induced magnetic field on H-mode operation. However, to better understand the involved phenomena, such a magnetic field could be generated by appropriate coils installed in a specific region of the vessel in order to explore various magnetic field perturbation levels (by varying the coil current). This approach would allow a parametric study to be performed and, possibly, to predict the effect of the corresponding TBMs in ITER for various ITER plasma operating conditions. So, experiments should be planned to install a scaled mock-up TBM module or simulating coil into a horizontal port on one or more existing tokamaks that readily achieves H-mode. The mock-up TBM should provide 0.5% to 1% additional local TF ripple near the outboard midplane to provide a range of ripple across that expected to be produced by the ITER TBMs. The experiments could be performed, for example, in DIII-D, EAST, or KSTAR. Preliminary results should be available by the end of 2011 before the completion of the final TBM detailed design. Experiments should continue in the following years.

5.4.4 Need of Be-protection on the TBM First Wall

FW Be-protection is expected to be used in ITER, but is not acceptable in DEMO. Therefore, from the TBM point of view, there is no need of launching R&D for manufacturing Be-protection on the TBM FW material (Martensitic steel).

A Design Task is therefore proposed to assess the acceptability of an unprotected TBM FW steel surface and to establish whether or not a Be-protection (taking into account the recession of the TBMs of 50mm, 100mm, up to 250mm) is needed. The case of a W-protection (acceptable in DEMO) should also be investigated.

This assessment is to be used as a guideline for the TBM conceptual design and for the TBM detailed design. These results are needed to decide if specific R&D has to be started on the fabrication of a Be-layer on Martensitic steels (technology not yet available).

6 Work Programme in Integrated Modelling

A proposal for the organization of ITER Physics Activities, endorsed by the ITER Council, includes the following directive to establish an Integrated Modelling (IM) Programme:

> Establishment of a programme on integrated modelling and control of fusion plasmas, including benchmarking and validation activities, which would be co-ordinated by the ITER Organization, but which would be developed using the relevant expertise within the Members' fusion programmes, making use of existing co-ordination structures where possible. The overall aims of this programme would be to meet the initial needs of the ITER project for more accurate predictions of ITER fusion performance and for efficient control of ITER plasmas, to support the preparation for ITER operation and, in the longer term, to provide the modelling and control tools required for the ITER exploitation phase.

The ITER IM Programme being established under this directive consists of two primary elements: i) an R&D programme that develops and validates physics models and provides scoping assessments of their impact on ITER operations, and ii) development of a suite of in-house integrated modelling capabilities that is further supported by a more extensive suite of models residing within the Members' research programmes. The former element (coordination of the R&D programme) is implemented through the ITPA and covered in the remainder of the Physics Workplan that is broken down into topical areas. The latter element (development of an integrated suite of modelling codes) is described here. It will require much closer collaboration with the ITER Members, and it will therefore be implemented through an Integrated Modelling Advisory Group (IMAG).

The in-house suite of codes will include the interactions between the plasma, plasma facing components, control systems, and diagnostics. The applications that drive this need range from the present emphasis on providing design support and reference scenarios, to planning the experimental campaigns in detail, to analysis of the experimental results and control of the plasma once ITER becomes operational. These in-house capabilities will be drawn from the ITER members' programmes and will be accessible to both internal and external users. The IM Workplan described here will further define and refine this development, then begin the process of its development that will likely continue through the life of the ITER project.

6.1 IM Suite definition and coordination

The definition and coordination of the IM Suite of codes will be a cooperative effort between the ITER Organization (IO) and the ITER Members, with the IMAG providing the means of coordination. A charter for the IMAG is being developed (Fall 2007) and will be submitted to the DAs for endorsement and identification of candidate members with appropriate technical expertise. Brief descriptions of the proposed tasks for the early part of the programme are described here.

6.1.1. Define objectives and organization

This task will seek agreement between the ITER Members and IO on the objectives and organization of the IM Suite. The proposed objectives and technical outline of the organization is being developed (Fall 2007) and will be submitted to the IMAG for review and suggested refinement. The overall level of effort as well as distribution of effort between the Members and IO will also be addressed.

6.1.2. Develop standards and guidelines

The IM Suite is envisioned to be comprised of stand-alone codes as well as modular components for specific elements of the physics, control, and system descriptions. All need to be well-documented and maintained. Standards and guidelines to facilitate the integration of these elements are being written (Fall 2007) and will be submitted to the IMAG for further development. The document describes external documentation (top level scientific documentation, implementation documentation, user guide) as well internal documentation, coding standards, qualification standards, verification guidelines, and validation guidelines. This effort is expected to evolve.

6.1.3. Define computing infrastructure

The IM computing infrastructure must accommodate internal as well as external users. Internal computing resources will be supplemented by external resources, so grid computing is expected to be a dominant feature along with some high performance computing (HPC) capabilities. The infrastructure must be coordinated with other IO departments, particularly in terms of data storage and retrieval.

6.1.4. Define standard component interfaces

The IM suite is envisioned to evolve toward a modular structure with different simulation, analysis and control elements uniting different combinations of these modules. The advanced development programmes within the Members' programmes are already moving in this direction for a number of reasons (flexibility in code construction, maintenance, evolution, verification and validation, etc). The ITER IM Suite will build on this effort, but recognizes that a significant amount of work will be required to establish standard component interfaces for the ITER suite that are reasonably compatible with other standards.

6.1.5. Identify and evaluate candidate components

A major, and continuing, task is to evaluate candidate components for inclusion in the ITER IM Suite. Maturity (documentation, verification and validation), availability, and resources for adaptation to ITER standards and guidelines must all be elements of the evaluation. It is expected that multiple expressions of a given component (e.g. different NBI or MHD equilibrium packages) will meet different needs for accuracy and speed, while standard interfaces will facilitate benchmarking against a set of reference cases.

6.2 IM Suite Implementation

The implementation of the IM Suite will also be a partnership between the IO and ITER Members, with coordination through the IMAG. But development of the hardware and software infrastructure will require much greater integration with other departments within the IO, so it will be implemented through a contract.

6.2.1. Hardware and software infrastructure

Advice on the hardware and supporting software infrastructure specific to the IM suite is envisioned to be gained through a contract with industry, and closely coordinated with CODAC within the IO. This contract will also support version control, regression testing, data management, graphics and workflow management.

6.2.2. Adapt candidate components to ITER IM needs

This element will comprise the bulk of the technical development by both the IO and the ITER Members, with the ITER Members providing the bulk of the effort. Targeted components will be adapted to ITER standards primarily through credited tasks. Installation, review of adherence to standards and guidelines, acceptance testing and development of a set of regression tests may be facilitated by the Visiting Researcher Programme in cases where an extended visit is necessary.