

## Principle Physics Developments Evaluated in the ITER Design Review

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**Abstract.** As part of the ITER Design Review, the physics requirements were reviewed and as appropriate updated. The focus of this paper will be on recent work affecting the ITER design with special emphasis on topics affecting near-term procurement arrangements. This paper will describe results on: design sensitivity studies, poloidal field coil requirements, vertical stability, effect of toroidal field ripple on thermal confinement, heat load requirements for plasma-facing components, edge localized modes control, resistive wall mode control, disruptions and disruption mitigation.

### 1. Introduction

The goal of ITER is to demonstrate the scientific and technological feasibility of fusion power for peaceful purposes and a key programmatic goal is that “The device should achieve extended burn in inductively driven plasmas with the ratio of fusion power to auxiliary heating power,  $Q$ , of at least 10 ( $Q \geq 10$ ) ...” [1]. As part of the ITER Design Review [2] and in response to the issues identified by the ITER Science and Technology Advisory Committee (STAC), the physics requirements were reviewed and as appropriate updated. This was a worldwide effort, with major contributions from the ITER Domestic Agencies and the ITPA. A comprehensive review of all of the scientific and technical issues will not be attempted here. Instead, the focus will be on recent work affecting the ITER design with special emphasis on topics affecting near-term procurement arrangements [2]. Areas requiring additional research will be discussed.

### 2. Design sensitivity studies

“Progress in the ITER Physics Basis” [3] provides an extensive review of the empirical scaling projections as well as one-dimensional modelling assessments for ITER. The physics uncertainties in projecting the performance of ITER are addressed assuming that the machine operates at full design parameters. The impact of modest changes in machine parameters,  $B_t$ ,  $I_p$  and  $\kappa$  (+/-10%) on the fusion power and  $Q$  were evaluated with the HELIOS code [4] as well as by spreadsheet analysis. The energy confinement time projections are based on the empirical scaling in [1]. As shown in the figure 1, the operating density, heating power and the power to achieve H-mode confinement define the operating space for the baseline 15MA, 5.3T scenario. To avoid the degradation in confinement at high density, the operating density is assumed to be 0.85 of the Greenwald density limit. The baseline heating power is 73MW and could be further increased if necessary. The back transition from H-mode to L-mode confinement is assumed to occur at the same power as the L to H-mode transition at the same parameters. The analysis by Martin *et al.* [5] taking into account the effective mass of the plasma is used to calculate the power threshold for H-mode confinement. At the nominal  $Q=10$  operating point, the power threshold is ~70MW and the power through the separatrix ~79MW, taking into account an estimate for the radiated power. The uncertainty in the power

threshold for the H-mode transition [3,5] and the possible occurrence of Type III ELMs near the power threshold [6], which degrades energy confinement, motivates further research.

For constant values of the safety factor, a 10% decrease in the toroidal field and plasma current results in  $Q$  being reduced from  $\sim 10$  to  $\sim 6$ . Similarly, if the elongation were reduced by  $\sim 10\%$  and the current decreased to maintain the safety factor at the baseline level, then  $Q$  is also reduced to  $\sim 6$ . Alternatively, when the current is increased from 15 MA to 17MA, the value of  $Q$  is projected to increase to  $\sim 20$ .

These sensitivity studies indicate the benefit of operational regimes with improved energy confinement [7 and references therein], to relax some of the operational constraints and also enable longer duration discharges. For the design review, these sensitivity studies reinforced the importance of reliably operating ITER at full toroidal field, plasma current, and elongation for the baseline scenario to fulfill its scientific and technology mission and of the impact of design or operational decisions, which could reduce the energy confinement time.

### 3. Poloidal field coil requirements

The unique combination of high current, high fusion power, and long pulse operation results in very stringent demands on the poloidal field system to provide adequate flux swing, to control the plasma shape including vertical position, the location of the divertor strike-points and the distance to the first wall, in the presence of disturbances. Analysis of the poloidal field coils for plasma control has focused on the requirements to develop satisfactory current ramp-up and burn phases of the 15 MA scenario, while exploration of possible scenarios for the current ramp-down phase is continuing [8]. This analysis has benefited from additional experimental results [9,10]. A large aperture startup has replaced the variable aperture startup, originally envisioned for ITER, and when combined with an early transition to a divertor configuration and plasma heating, produces a plasma which is more vertically stable, reduces heat loads to the first wall and prevents excessive flux consumption prior to the start of the burn phase. The experimental results have indicated that the rapid development of a high edge pedestal leads to a rapid decrease in  $I_i(3)$  after the onset of the H-mode and a slower decline to values as low as 0.6 towards the end of burn, somewhat outside the original reference range of  $0.7 \leq I_i(3) \leq 1.0$  specified for the original design of the ITER poloidal field control system.

The ITER 15 MA reference scenario has been analyzed with several time-dependent and equilibrium codes [8]. To enable operation of low inductance discharges characteristic of H-mode operation, the design has been modified to: upgrade the current and field capability of the poloidal field conductor; increase the number of turns in PF2 and PF6; increase the limit

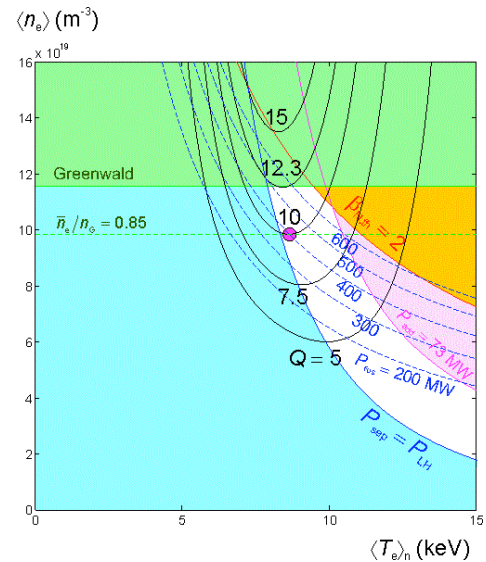


FIG. 1. The operating space is shown for the baseline 15MA, 5.3T ELMy H-mode scenario. At an operating density of .85 of the Greenwald limit, the projected  $Q$  is 10 with 40MW of heating power and  $\tau_E = 3.8s$ .

on the central solenoid vertical separation forces (from 75 MN to 120 MN), based on a detailed stress analysis of the central solenoid; relocate PF6 toward the plasma by 9 cm and radially by 7 cm; sub-cool PF6 to about 3.8; and modify the divertor slots and dome geometry, reducing the dome height by  $\sim 9$  cm and shifting the target plate, as shown in figure 2. Current analysis is focusing on analyzing the effect of plasma disturbances on the operating range and a detailed assessment of rampdown phase of the discharge including the H to L transition.

#### 4. Vertical Stability

Loss of vertical plasma position control in ITER will cause large thermal loads on plasma-facing components and also generate the highest electromagnetic loads. Performance of the ITER vertical stabilization system has been analyzed taking into account results from present devices. These results show that in the current ramp-up and flat-top of the 15 MA,  $Q=10$  reference scenario, a range of internal inductance,  $0.6 \leq l_i(3) \leq 1.2$ , is likely to occur, as shown in C-Mod [11]. In the current decay,  $l_i(3)$  can rise to even higher values and, in this phase, scenario adjustments (e.g., reducing elongation) will be required to maintain acceptable vertical stability. Studies are continuing on this phase.

The parameter,  $\Delta Z_{\max}/a$ , has been identified as a figure of merit for characterizing the effectiveness of the vertical stabilization, where  $\Delta Z_{\max}$  is the maximum “sudden” plasma displacement, which can be stabilized. In present devices,  $\Delta Z_{\max}/a > 0.05$  is required for reliable vertical stabilization with robust stability achieved with  $\Delta Z_{\max}/a > 0.1$  [12-13].

To provide reliable operation at the elongation required, an in-vessel coil system, shown in figure 3, has been proposed for increased vertical stability, the application of resonant magnetic perturbations to stabilize ELMs, and feedback stabilization of resistive wall modes. Analysis performed to date indicates that this system will satisfy the requirement that  $\Delta Z_{\max}/a > 0.05$  at acceptable levels of current and voltage and that it can control the plasma vertical position with minimal overshoot. For comparison, the vertical stabilization system was originally only able to achieve  $\Delta Z_{\max}/a$  of  $\sim 0.02$  [13].

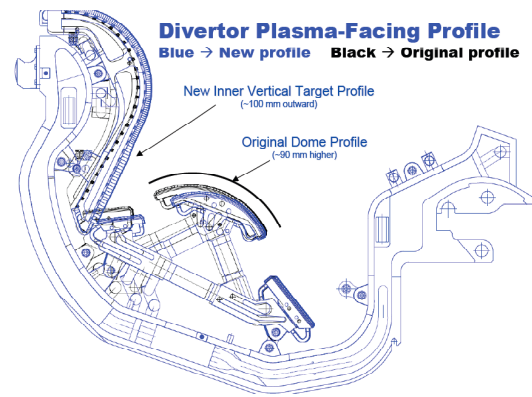


FIG. 2. Proposed changes to the divertor dome and target plate location to improve the location of the strike-points in low inductance discharges.

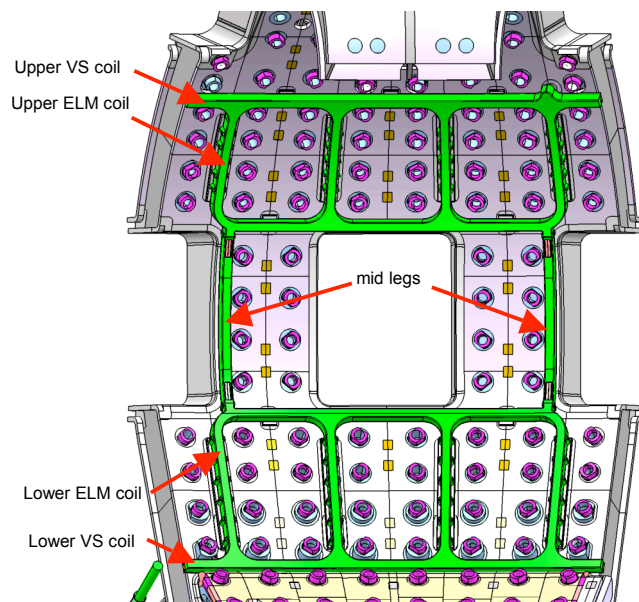


FIG. 3. Proposed design of in-vessel coils for vertical stabilization and ELM control. The ELM control windings can also be used for resistive wall stabilization.

## 5. Effect of toroidal field ripple on thermal confinement

Toroidal field (TF) ripple can affect the confinement of energetic particles (alpha and beam ions) and result in local heat deposition on plasma facing components [14 and references therein]. Recent results from JT-60U [15,16] and JET [17] have shown that toroidal field ripple can also effect the thermal confinement of H-mode plasmas. After the installation of ferritic inserts which reduced the ripple from ~2% to ~1% in the large aperture JT-60U plasmas, the energy confinement time improved by ~15% together with an increase in edge toroidal velocity, increased pedestal temperature, and larger ELMs. The JET experiments, by varying the currents in the toroidal field coils, were able to explore the impact of operating with lower ripple (0.08 to 1%) than in the JT-60U experiments. These experiments also showed degradation in the energy confinement, pedestal height, and plasma rotation with increasing TF ripple. The effect of ripple on JET is strongest in the low plasma edge collisionality regimes in which ITER will operate and indications of degradation in the energy confinement time are observed even at ripple amplitudes of only 0.3%.

The experimental results indicate that the existing toroidal field ripple with the planned ferritic inserts may reduce the projected ITER performance. A study to reduce the ripple specification to “as low as reasonably achievable” was approved. The underlying physics of how ripple affects plasma confinement, MHD stability and ELM behavior is still an active area of research, as will be discussed further in the section on ELM control where resonant magnetic perturbations are used to control ELMs. In addition to the toroidal field ripple, the tritium breeding modules will introduce field perturbations. An acceptable level of perturbations or compensation techniques for the effects associated with the tritium breeding modules remains to be determined.

## 6. Heat load requirements for plasma-facing components

Results from present divertor tokamaks have shown that plasma fluxes to the main wall are dominated by intermittent events leading to fast plasma particle transport, which ends up reaching the plasma facing components along the magnetic field [18 and references therein]. Although the steady-state parallel power fluxes associated with these particle fluxes will be only several  $\text{MWm}^{-2}$  in the ITER 15MA reference scenario, they have the potential to cause local overheating of exposed edges of main wall plasma facing components arising from the achievable misalignment accuracies. Similarly, transient events are expected to cause significant power fluxes to reach first wall panels in ITER along the field line. Such loads from ELMs are likely to cause melting of beryllium up to several 10's of  $\mu\text{m}$  in the exposed edges, which can cause undesirable impurity influxes at every ELM [19].

Loarte *et al.* [19] discusses the physics basis for the revised requirements for the thermal loads to the first wall divertor, while Lowry *et al.* [20] describes how the design of the first wall components has changed in response to requirements on the parallel heat fluxes, effect of ELMs, and halo currents during disruption.

## 7. Edge Localized Modes (ELM) control

ELMs can be driven by the large pressure gradients and current densities associated with the very small energy diffusivity of the H-mode plasma edge. Extrapolations from existing experiments to ITER indicate that unmitigated ELMs on ITER would correspond to ~20MJ

energy loss per ELM. Recent analyses of divertor heat loads due to ELMs and new information about material damage for both carbon fiber composite and tungsten divertor targets [21] has led to the conclusion that an incident energy impulse of more than  $\sim 0.3\%$  of the total thermal plasma energy ( $\sim 1\text{MJ}$ ) can cause tile fatigue and cracking as well as erosion, and larger energy losses can ablate or melt divertor materials potentially degrading the purity of ITER plasmas and greatly reducing the lifetime of the ITER divertor. These results imply a need to reduce the energy loss by a factor of  $\sim 20$  and being able to do so very reliably.

Tools that can either eliminate or greatly reduce ELM energy losses without significantly degrading confinement are therefore critically important for successful operation of ITER. Thomas *et al* [22 and references therein] provides a more comprehensive discussion of this topic regarding the ITER Design Review. An overview of two approaches, pellet pacing and application of helically resonant magnetic perturbations (RMP), which were evaluated as part of the ITER design review, will be given here

Experiments on ASDEX Upgrade, DIII-D and JET have demonstrated that pellets can trigger ELMs, enabling the production of more frequent smaller ELMs. ELMs were reduced by 40% by the application of pellet pacing to control the ELM frequency. The present experiments are accompanied by a small degradation in the energy confinement time, which is associated with increased convective loss [23,24]. Pellet pacing to control ELMs on ITER is a significant extrapolation from current experiments because the ratio of the pellet repetition time to the energy confinement time is much smaller. To decrease the estimated adverse effects of the accompanying convective energy loss at the required frequency and depth of penetration for ITER will require the development of a higher speed pellet injector utilizing smaller pellets. Further experimental results are needed to refine the requirements for the depth of penetration to trigger ELMs and evaluate the impact on energy confinement when using higher frequency pellet injectors. To provide for the capability to incorporate pellet pacing on ITER, the gas throughput requirements were increased.

The application of resonant magnetic field perturbations (RMP) to control ELMs began with early work on JFT-2M [25] and is currently a very active area of research. Different experimental results have been obtained, depending on the applied perturbation mode spectrum [22, 26 and references therein]. The most successful results in controlling ELMs were reported by the DIII-D team, who suppressed ELMs with  $n=3$  RMPs from small aperture, internal, off-mid-plane rows of coils, and obtained an H-factor of  $\sim 1$ , albeit it in a relatively narrow range of the safety factor,  $3.2 < q_{95} < 3.8$  with their existing coils.

Results from DIII-D together with those from the other devices and theoretical considerations provided guidelines for the design of the ELM suppression coils on ITER [22, 26]. In support of the ITER in-vessel coil design, assessments were performed for different coil configurations to evaluate the magnetic spectrum and the coil current requirements. As shown in figure 3, three rows of coils with one row above and one below the mid-plane as well as one on the mid-plane are proposed for each of the nine vessel sectors to satisfy the guidelines and provide the flexibility to avoid unwanted perturbations in the core [28]. These coils would be located behind the blanket shield module and provisions are included to enable remote maintenance of the coils.

A comprehensive understanding of the underlying physics is still emerging, motivating additional experiments and analysis, including an assessment of the range of validity of these guidelines, role of edge pumping and pellet injection. The criteria for field line alignment and

mode spectrum as well as magnitude of the perturbed field remains an active area of research in understanding the results from the different devices. This has also motivated further theoretical work on the role of resonant and non-resonant magnetic perturbations, which is important not only for ELM suppression but, more broadly, for error field correction and avoidance of locked modes [27, 28]. Variation of the location of the strike-point with respect to the DIII-D cyropanels demonstrated that edge pumping is an important consideration. Furthermore, the effective particle confinement time decreased in DIII-D experiments with the application of RMP coils decreasing the plasma density but that should, in principle, be compensated with increased core pellet fueling. Pellet injection into discharges with resonant magnetic perturbations has in some cases, but not always, triggered ELMs. Since pellet fueling is integral to achieving the required densities in ITER, this needs to be studied further. ELM control and mitigation is important for the success of the ITER program and warrants the scientific attention that is being given to it.

## 8. Resistive Wall Modes (RWM)

An important goal of ITER is to demonstrate steady-state-compatible operation at moderate fusion gain ( $Q=5$ ). Operation in “steady-state” regimes in ITER entails operation at high normalized pressure ( $\beta_N > 3$ ), which can destabilize a resistive wall mode [29 and references therein]. Even if plasma conditions allow for RWM stabilization for a given plasma rotation profile and magnitude, the damping will be weak, allowing the RWM to be easily excited to finite amplitude by static error fields [30] or by other MHD events [31]. For this reason, active feedback control of the RWM is necessary in ITER. Experiments on DIII-D and NSTX have shown that feedback control of the unstable RWM at low rotation is possible provided the mode identification and control field response are sufficiently rapid [32].

The in-vessel coils for ELM control can also be used for resistive wall mode (RWM) control. Comparison of the performance of the coil configurations using the VALEN-3D code [33] modeling with a single-mode model shows superior performance when all three toroidal rows of coils are used (stabilized  $\beta_N = 3.74$ ) compared with mid-plane coils (stabilized  $\beta_N = 3.39$ ). The use of just the top and bottom coils enables stabilized  $\beta_N = 3.83$ . The required current for RWM control appears to be modest compared with the ELM control requirements; however, further analysis is ongoing [33,34].

## 9. Disruptions and disruption mitigation

ITER will have the largest plasma current and hence stored plasma and magnetic energy of any magnetic confinement facility. Hence, plasma disruptions will have the greatest potential impact on ITER due to runaway electron

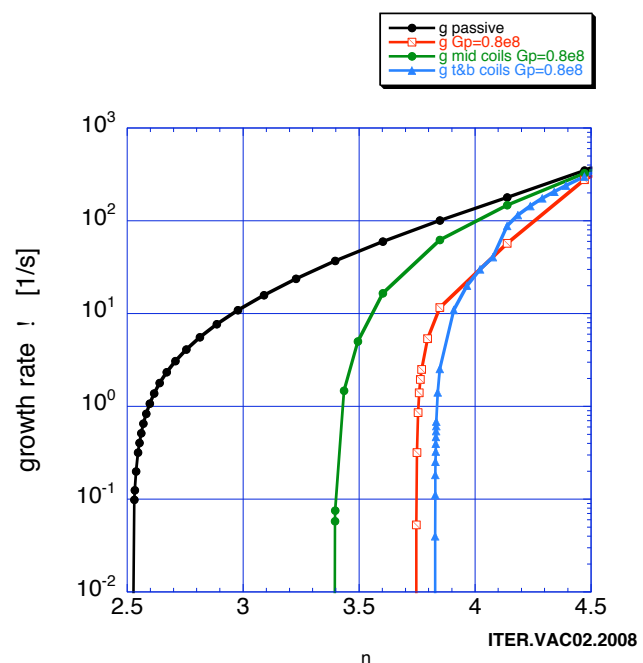


FIG. 4. VALEN calculations [33] of the growth rate of the resistive wall mode for the coils in figure 3.

generation as well as from the release of the stored thermal and magnetic energy [29 and references therein]. Recent application of JET data [35] to the ITER design has identified the need to take into account additional electromagnetic forces due to the presence of a non-rotating kink mode during a vertical displacement event. In addition to the vertical loads due to the vertical displacement event (VDE), which are the dominant loads, the kink mode results in a sideways force on the vessel. Unlike the vertical forces and halo currents, which have been extensively studied on a number of machines, the horizontal sideways force has been documented on JET but not observed to be as large on other machines [29 and references therein]. Further experimental results and theoretical analysis [35,36] would be valuable to improve the extrapolation of forces and tilting moments to ITER. The vacuum vessel load requirements, taking into account a factor of 1.2 for differences between DINA modeling and results on JT-60U, were revised from a peak horizontal force of 25 to 50 while the peak downward vertical force was revised from 75 to 108 MN [74] and a requirement for the tilting moment of  $\sim 215 \text{ MN}\cdot\text{m}$  was incorporated. The most critical area of the vacuum vessel affected by the increased electromagnetic loads on the vacuum vessel structure is the connection of the lower port to the main vacuum vessel shell. To accommodate the change in the design loads, the poloidal gussets supporting the vessel have been reinforced.

To ameliorate the impact of a disruption on ITER operations, massive gas injection is proposed to radiatively terminate the plasma discharge, which alleviates plasma power loading on plasma-facing materials and minimizes halo currents [29 and references therein]. However in ITER, mostly due to its large plasma current, a further requirement is that runaway electrons be suppressed. A large runaway electron current is predicted to damage the beryllium tiles on the blanket shield modules.

Although the details of the runaway confinement are not fully understood, the conservative approach to suppressing runaway electrons is to increase the collisional damping such that the density exceeds the commonly referred to “Connor-Hastie” [37] or “Rosenbluth” density [38] by the injection of large amounts of gas into the plasma volume prior to the current quench. The gas load requirements have been estimated for ITER assuming a fueling efficiency of 20% at up to  $500 \text{ kPa}\cdot\text{m}^{-3}$  [39]. There are significant engineering and operational implications associated with such a large gas load as discussed by Whyte *et al.* [39]. Experimental and theoretical work has shown such gas loads corresponding to the Rosenbluth density are not needed in current experiments; however, current experiments may be operating in a different operating regime due to the lower current. Further research on developing a more quantitative understanding of the gas loads and the mechanism for introducing the gas into ITER for disruption mitigation and runaway suppression is needed but ultimately, this may only be resolved during the hydrogen commissioning phase of ITER.

## Summary

The results presented here were the result of an intense effort by the international community. The review of physics requirements and performance conducted within the Design Review has given rise to design changes in several areas in response to progress in physics understanding in recent years, and has confirmed the capability of the ITER device to satisfy the major aims of its mission. This work has motivated further research, which will enable refinements of the design requirements and support planning for the operational phase. Continued close interaction between the ITER Organization and the international scientific and technical community will be critical to ensure that the most optimal use is made of ITER. The ITER Organization through its focus on the construction of the project identifies

important problems, of interest to the scientific community. The scientific community through its ongoing research program identifies solutions to problems that have not yet been articulated. This synergy is valuable and should be maintained beyond the design phase.

This report was prepared as an account of work by or for the ITER Organization. The Members of the Organization are the People's Republic of China, the European Atomic Energy Community, the Republic of India, Japan, the Republic of Korea, the Russian Federation, and the United States of America. The views and opinions expressed herein do not necessarily reflect those of the Members or any agency thereof. Dissemination of the information in this paper is governed by the applicable terms of the ITER Joint Implementation Agreement. The work was supported in part by U.S. DOE Contract DE-AC02-76CH03073.

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