# ASSESSMENT AND MODELING OF INDUCTIVE AND NON-INDUCTIVE SCENARIOS FOR ITER

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**Abstract**

This paper presents recent developments in modeling and simulations of ITER performances and scenarios. The first part presents an improved modeling of coupled divertor/main plasma operation including the simulation of the measurements involved in the control loop. The second part explores the fusion performances predicted under non-inductive operation with internal transport barrier. The final part covers a detailed scenario for non-inductive operation using a reverse shear configuration with lower hybrid and fast wave current drive.

## 1. KINETIC CONTROL OF THE BURN INCLUDING MODELING OF DIAGNOSTIC MEASUREMENTS

Previous simulations using the 1.5D PRETOR code [1], incorporating a 0D model of the divertor, assumed perfect measurements of the relevant input quantities, namely fusion power, volume average density and power reaching the divertor target. It was shown that simultaneous control of the fusion power and divertor power load were possible. As a first step in assessing the effect of including the diagnostic limitations in the control loop, the divertor plate temperature, $T_{target}$ as measured by the IR system, was used as a feedback parameter for impurity (Ar) injection [2]. For $T_{plate}$ 230°C upwards, this system [3], has systematic errors in $T_{plate}$ less than 10%, and the noise at 230°C and 50 ms time interval is ~ 10%. The thermal inertia time-constant of the divertor plate, — a 2 cm thick slab of CFC held at 140°C on the coolant side — is ~ 0.8 s. A simple PD controller working on e = tanh (x + 50 $x^3$), where x = ($T_{target}$ - $T_{set}$ )/$T_{set}$ signal was used to control the incremental gas feed at each time step. This function was found to provide a stable operating point for small perturbations, and to minimize the response time to large perturbations without leading to radiative collapse of the core. Figure 1 shows that $T_{target}$ is constant to within 20% during the burn phase and heating excursion, even though there is a short attached period where the power to the target exceeds 10 MW/m$^2$. The target operating point, $T_{set}$ is 500°C initially, but at t = 400 s, it is reduced to 200°C in anticipation of the H mode termination which leads to a transient increase of the power to the target plates to above 15 MW/m$^2$, briefly increasing the temperature to ~ 1000°C. At low $T_{plate}$ the noise performance of the diagnostic is relatively poor; this can be seen as noise on the Ar fraction. Based on these results, this IR system is adequate for impurity injection control purposes, although an improvement of S/N at low $T_{plate}$ is desirable.

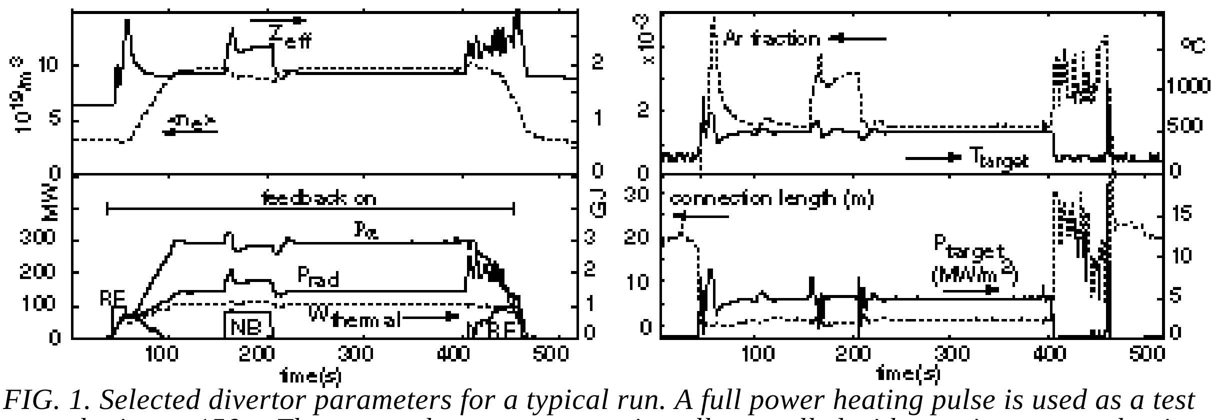


*FIG. 1. Selected divertor parameters for a typical run. A full power heating pulse is used as a test perturbation at 150 s. The target plate temperature is well controlled with zero instrumental noise. The strike width was assumed constant at 0.1 m.*

## 2. PERFORMANCE DIAGRAM FOR NON-INDUCTIVE OPERATION

The 1-1/2D transport code PRETOR implementing the transport modeling assumptions used to predict ITER performances under inductive operation [4] is used to compute the plasma current density profile in absence of inductively driven ohmic current and to predict the confinement enhancement factor as a function of assumed properties of the Internal Transport Barrier (ITB). The ITB is assumed to be located in the zero or negative shear region – which is predicted from the current profile – and to be characterized by the amount by which transport is reduced in this region. A reference operating point was determined under the following assumptions: 20 MW of NBI heating, 80 MW of ECCD off-axis heating with an efficiency of 0.2 $10^{20}$ MA/MW/$m^2$ at 10 keV, a transport reduction of 80% and a location of the off-axis current drive chosen such that the radial location of the ITB is rho = 0.62 (square root normalized toroidal flux). Once this operating point is established, a 0D code is used to extrapolate it to a wider operating space (varying confinement, density and normalized beta) as shown in Fig. 2a. This operating space is useful to translate the confinement enhancement factor HH (defined as the energy confinement time normalized to that given by the ELMy H-mode scaling expression) and operational limits into fusion power under the above assumptions for non-inductive operation.

A separate set of simulations performed with the 1-1/2D transport code is presented in Fig. 2b. In this set, the transport enhancement factor HH is computed consistently from the radial location and magnitude of transport reduction of the ITB. For each point the density is set equal to the Greenwald value, the current drive efficiency and powers are the same as above and no inductive current is present. The imposition of the ITB increases the temperature peaking, and more modestly the density profiles, and therefore increases the plasma stored energy. This increase in stored energy relates to an increase in HH factor which can be computed and is given on the vertical axis. At low confinement (HH of the order one) however the plasma current and therefore the density become very low. Because of the high auxiliary heating (100 MW), the $Z_{eff}$ is high leading to high core radiation and slightly inverted temperature profiles and therefore HH slightly below 1 as visible in Fig. 2b.

This diagram, which is valid for the above current drive efficiency and powers, helps to translate the observed ITB properties into an equivalent HH factor for ITER.

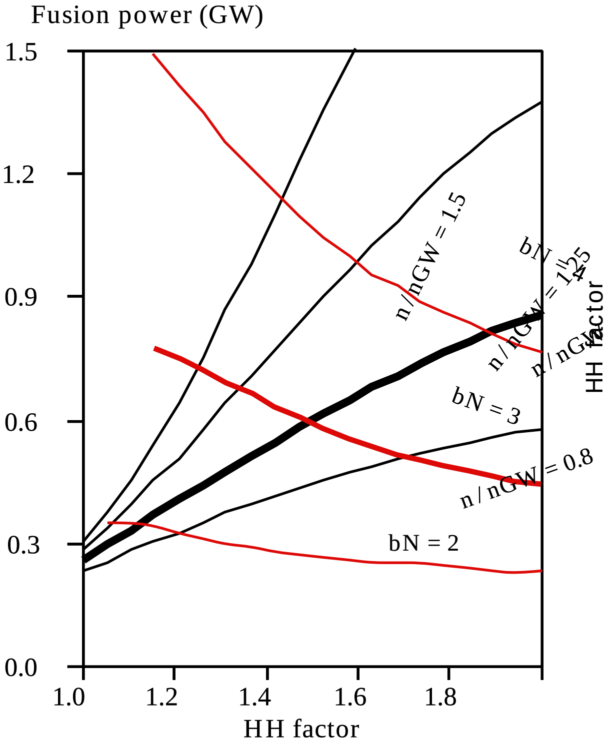


*FIG. 2a: Fusion performance for 80% transport reduction, inside rho(ITB) = 0.62 and $\gamma_{CD}$ = 0.2.*

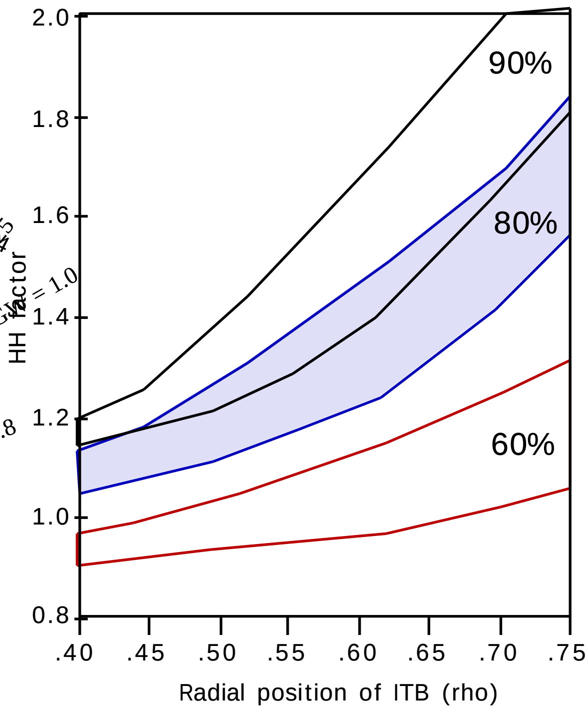


*FIG. 2b: HH versus rho(ITB) for various transport reductions (60, 80 and 90%). The spread corresponds to various off-axis CD profiles.*

Although the physics of the ITB are not yet well determined, it is possible to derive the necessary conditions to achieve a given fusion objective for ITER under non-inductive operation. Achieving better than for instance Qth = 5 at the Greenwald density with an off-axis current efficiency of 0.2 and 100 MW of power would required HH > 1.35 and betaN > 2.8. This in turn requires at least 80% transport reduction with rho(ITB) larger than about 0.5. This provides a valuable goal against which present experiments in optimized shear configuration can be compared.

## 3. NON-INDUCTIVE SCENARIO USING COMBINED LOWER HYBRID AND FAST WAVE CURRENT DRIVE

The previous section studied the static performances of non-inductive operation by reaching a steady-state regime (i.e., all plasma parameters reaching their equilibrium value) without inductively driven current for each variation of the characteristics of the Internal Transport Barrier. This section addresses the dynamic aspects of non-inductive scenarios in order to determine the requirements for plasma formation, establishment of the ITB and its dynamic control over time. At least three aspects need to be studied:

• The replacement of the ohmic current required during the plasma formation by non-inductive current (a combination of bootstrap and current drive);

• The establishment of a desired current density profile and its sustainment as the ohmic contribution is being replaced;

• The control algorithm needed to maintain the ITB and the plasma fusion power over time.

Such a dynamic scenario has been developed using the 1-1/2D code ASTRA [5] and a combination of Fast Wave and Lower Hybrid current drive systems. The characteristics of the ITB used in the simulations are given by a local heat transport model which is described in a companion paper [6]. The model relates the radial location of the ITB and the magnitude of transport reduction as defined in section 2 above to an analytic dependence on the local magnetic shear. The analytic expression is chosen to produce a reasonable agreement with present reversed shear experiments from JET, TORE SUPRA and TFTR.

The time traces of key global parameters for this scenario are shown in Fig. 3. The plasma profiles reached during the final steady-state phase where the fully non-inductive regime is established are shown in Fig. 4. This scenario is divided into four phases:

*1. Initial current ramp up and formation of the RS configuration:*
A combination of ohmic current ramp up and Fast Wave heating is used to ramp the plasma current while establishing a reversed magnetic shear in the central region.

*2. Low-current reversed shear steady state equilibrium*
The next phase of the scenario is the replacement of this off-axis OH current by a Lower Hybrid (LH) driven current while holding at the same time the RS configuration and controlling the central current with FWCD.

*3. Second LH-driven current ramp up and plasma shaping*
A controlled increase in plasma volume, shaping, and FW/LH power is applied to evolve to a full size plasma with the required q profile.

*4. Feedback-controlled density rise and burn phase*
The average density is increased to ramp up the fusion power to the required value of 300 MW. A RS configuration with $q_0$ around 3.5 and a minimum q-value, $q_{min}$ = 1.5 is produced. A moderate amount of FWCD (~ 20 MW) is sufficient to maintain the central current. The ITB is held at mid-radius by the control of the LH power deposition profile through the $N_{//}$-spectrum of the LH launcher. Preliminary BANDIT-3D [7] calculations predict LH power deposition and current drive profiles centered at r/a ≈ 0.6-0.7 with a launched $N_{//}$ of 2.2, but with a low edge density. The bootstrap current is peaked slightly inside the ITB with a bootstrap fraction around 64%.

The simulations indicate that with a careful choice of plasma shaping, average plasma density, surface plasma voltage (to control the ohmic current component), and of two independent — one central, one off-axis — heating and current drive systems, a fully non-inductive regime can be established and controlled over time (see also [8]). Additional work is needed to produce these scenarios with other choices of heating and current systems (Neutral Beam and/or Electron Cyclotron Resonance Heating) and study the sensitivity of these scenarios to the assumed transport model for the ITB and to the other implicit assumptions used in the modeling such as density profile or coupling with the divertor to determine the plasma purity.

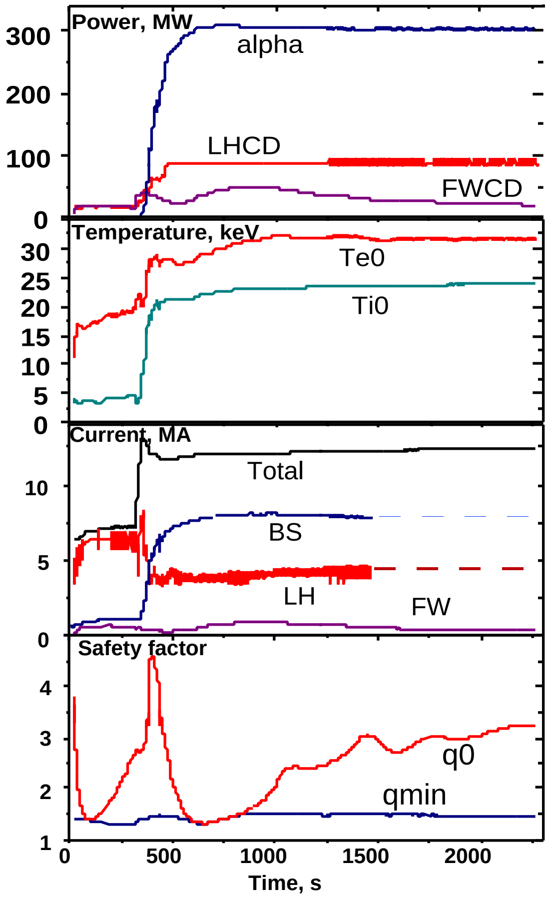


*FIG. 3. Time traces for (a) power; (b) central temperatures; (c) plasma current and (d) q(0) and $q_{min}$.*

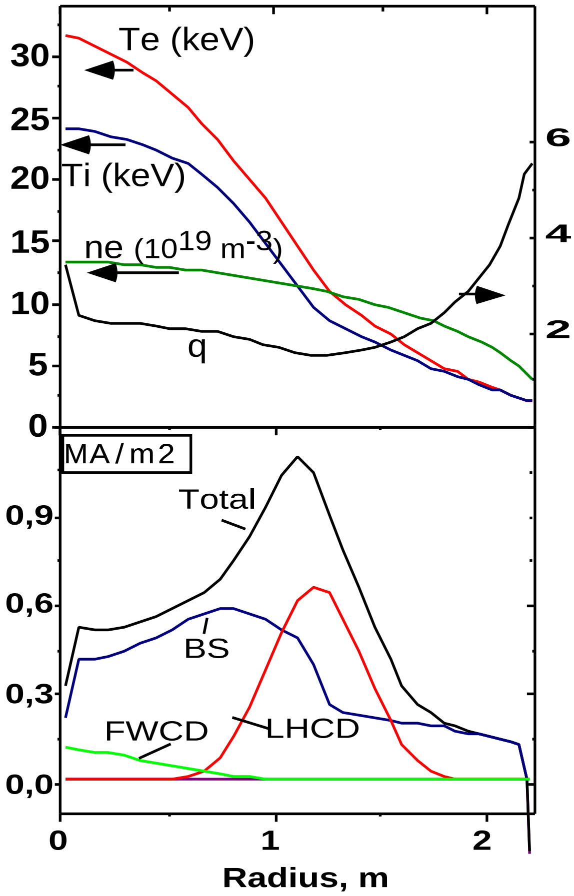


*FIG. 4. Radial profiles for (a) temperatures density and q; (b) current densities.*

### *Acknowledgment*

*This report is an account of work undertaken within the framework of the ITER EDA Agreement. The views and opinions expressed herein do not necessarily reflect those of the Parties to the ITER EDA Agreement, the IAEA or any agency thereof. Dissemination of the information in these papers is governed by the applicable terms of the ITER EDA Agreement.*

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