Development of the ITER Baseline Inductive Scenario

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Abstract. Sustainment of Q~10 operation with a fusion power of ~500MW for several hundred seconds is a key mission goal of the ITER Project. Past calculations and simulations predict that these conditions can be produced in high-confinement mode operation (H-mode) at 15MA relying on only inductive current drive. Earlier development of 15MA baseline inductive plasma scenarios provided a focal point for the ITER Design Review conducted in 2007-2008. In the intervening period, detailed predictive simulations, supported by experimental demonstrations in existing tokamaks, allow us to assemble an end-to-end specification of this scenario consistent with the final design of the ITER device. Simulations have encompassed plasma initiation, current ramp-up, plasma burn and current ramp-down, and have included density profiles and thermal transport models producing temperature profiles consistent with edge pedestal conditions present in current fusion experiments. In this paper we present new results of transport simulations fully consistent with the final ITER design that remain within allowed limits for the coil system and power supplies.

1. Introduction

The central solenoid (CS) and poloidal field (PF) coil system on ITER, Fig. 1, provides inductive current drive and feedback control of the plasma current, position and shape, as well as stabilization of the unstable vertical motion of the elongated plasma. To provide sufficient flexibility for exploring the physics of burning plasmas, these coils must have a margin for operation that remains within the engineering limits of forces, currents and magnetic fields in the superconducting coils. Successful ITER operation achieves fusion power near P_{fusion} ~ 500MW with sufficient neutron production to operate at a fusion gain of Q ~ 10 ($Q = P_{fusion}/P_{auxiliary}$) for a duration of 300 to 500s. Based on tokamak performance to date, this baseline operation for ITER [1] will use an inductively driven plasma current of 15MA under the assumption of access to high-confinement mode (H-mode) exhibiting edge-localized modes (ELM) that limit the edge pressure. The resulting high stored energy and large edge current density put significant demands on the PF system for shape and vertical stability control. Many of these issues were explored in earlier simulations [2] where several parameters and characteristics of the plasma control and performance are discussed.

The CS/PF coil layout for ITER shown in Fig 1 identifies the coils available for shape and vertical stability control with the possibility for three different stabilizing circuits as indicated. ITER has recently converged on several final component designs for construction. These have resulted in modifications to both the coil geometry and the plasma divertor shape. In finalizing these designs, we can now concentrate on a more accurate assessment of the operating space expected for the full 15MA plasma current. In light of these recent design modifications, we are also re-assessing the expected scenario performance with time-dependent simulations of the plasma evolution. This scenario re-assessment examines performance sensitivities for baseline operation plus evaluation of potential alternative current ramp rates for both the plasma ramp up to 15MA and ramp down after burn. These scenario simulations include 2D free-boundary equilibrium evolution for validating the feedback control of the plasma shape and position coupled with thermal transport models to assess performance. This evaluation is stimulated, in part, by recent experimental studies exploring scaled ITER-like operation [3,4] to validate the models and parameter choices made in formulating ITER operating scenarios.

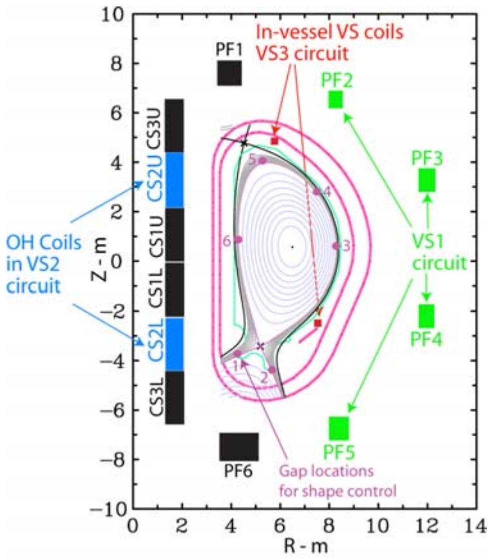

Fig. 1 ITER coils system to sustain inductive plasma current and for shape and vertical stability control. Three vertical stabilization circuits are indicated along with the six controlled plasma-wall gaps.

The studies presented here have been completed with two separate free-boundary equilibrium and thermal transport codes, CORSICA [5] and DINA [6]. These codes were part of the previous benchmark simulation study [2] and are described in this earlier publication. In this effort, CORSICA is used to study performance sensitivities while DINA is exploring the use of fast controlled current ramps. While the TSC code is also participating in the revised operating assessment, this effort is concentrating on developing alternatives to the inductive scenario, the hybrid and steady-state [7] operation on ITER [8]. We have evaluated sensitivity to assumptions concerning the H-mode pedestal parameters using a variation of the edge transport assumptions and have explored limits to the current ramp rate. We have begun exploration of the inside wall-limited start-up and ramp down. Finally, since the early experiments on ITER will include non-nuclear operating scenarios running with hydrogen or helium or with limited neutron flux operation in deuterium (without 50/50 mix of deuterium and tritium), an evaluation of these low activation scenarios has recently been initiated.

2. Operating limits on forces and currents

Improvements to the ITER baseline design as a result of the Design Review have been adopted and these offer significant benefits to the reference scenarios. Enhancements include increases in the current and field limits of both the CS and PF coils, changes in the allowed forces on coils and a re-specification of the coil power supply designs. In addition to these, a small outward movement of the location of the inner vacuum vessel shell can impact plasma control that must be included in the scenario analysis. With the baseline design now essentially frozen, extensive analyses are underway to explore the impact of these system design changes over a range of plasma scenarios, plasma control, and limitations to operations.

Modifications to the central solenoid (CS) coils, Fig 1, along with changes in the plasma wall can alter the operating space available [2] both by modifying the flux consumption needed to reach steady burn conditions and by putting different limits on the allowable forces and currents in the coils. Using static equilibria, we have evaluated changes to the operating space available to ITER for the 15MA inductive scenario during burn. In Fig. 2 we show the calculated change in operating space boundaries resulting from the modification of the coils during final design. These boundaries result from limits to either the maximum allowed current in the coils or from forces on the coils. Scenarios must operate within these limiting boundaries. The relatively small reduction in operating space in the high flux area results mostly from changes in the detailed design of the CS. The differences between the various codes is a result of imposing different shape constraints relative to a target separatrix shape in solving for the free boundary equilibria that remain within the coil current and force limits. This difference is also a result of the method used to smooth the spatial variation of coils currents themselves. While the available operating space is reduced, our re-assessment indicates there is still sufficient margin for successful operation.

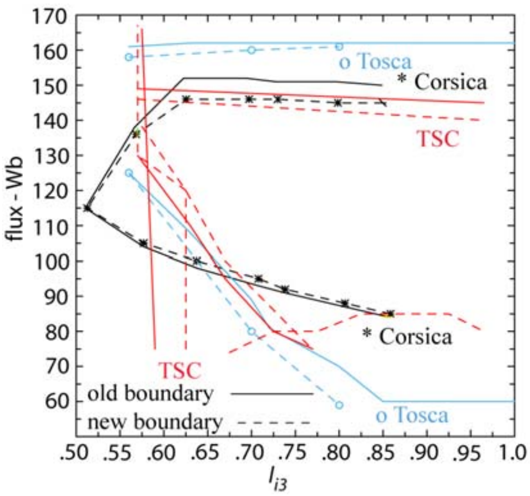

Fig.2 Comparison of operating space changes resulting from final modifications to the CS/PF coils system as calculated with the CORSICA (Casper), TOSCA (Fujeida) and TSC (Kessel) free-boundary equilibrium codes. Differences are due to target shape constraints and smoothing of the coil current variations.

3. Forward free-boundary 15MA controlled scenario

Free-boundary controller simulations represent our most complete evaluation of the scenario development in that they utilize proposed controllers to maintain the plasma current, shape and vertical stability. In simulations presented here, for plasma vertical stabilization we use the VS1 controller that varies differential current in the coils PF2, PF3 and PF4, PF5 shown in green in Fig. 1. The Coppi-Tang transport model [2] is used for simulating the thermal transport since it is fast, robust and defined over the full cross-section of normalized toroidal flux, ρ. This model has been shown to give reasonably good agreement with the profile evolution in DIII-D [9,10] and TFTR [11] and is currently being evaluated in benchmark studies for other experiments [12]. This model allows for scaling of the edge thermal transport so as to generate an H-mode-like edge pedestal inside the separatrix that affects the overall performance of ITER through the energy transport. The (bootstrap) current density peak resulting from the edge pressure gradient alters the dynamics of the plasma controller.

The density profile is prescribed under both L-mode and H-mode conditions with no particle peaking assumed, Fig. 3. The impurity concentration is set to give an effective charge state of Z_{eff}~1.8 during burn. The alpha particle density is simulated from a production rate equation with particle diffusion used to limit the alpha particle build-up and the potential poisoning of the plasma reaction rates. With the application of

52MW of heating power, we assume the plasma transitions into H-mode. We do not apply an H-mode access power scaling law so as to simplify the study and analysis of H-mode performance. We also use analytic approximations to the heating profiles and do not rely on auxiliary current drive other than the bootstrap current that is calculated self-consistently via the NCLASS model [13]. The plasma current ramp for these simulations was fixed at the same ramp rate used in the previous studies [2] with full plasma current reached at 80s. Variations in the current ramp rates were completed with the DINA code and these results are presented later in this paper. We show in Fig. 3 temperature profiles from the Coppi-Tang model under L- and H-mode conditions. In Fig 4, we show a forward feedback control simulation resulting in Q = 10 and P_{fusion} = 480MW determined in part from the edge transport assumption giving a pedestal temperature of T_{ped} = 4.5keV defined as the temperature at ρ~0.95, Fig. 3d. The waveform for auxiliary heating is shown in Fig. 4a along with the resulting evolution of alpha heating power and radiation losses. In this simulation, we use feedback on the stored energy to adjust the auxiliary heating from 100s to 130s to smooth the transition to burn conditions. The resulting coil current waveforms are shown in Fig. 4b and 4c. The control voltages shown in Fig. 4d-4f are the outputs of the VS1 feedback controller with V_{fast} providing vertical stability control and the slower voltages providing the plasma current and shape variations. All coil currents remain within the allowed limits for the new CS/PF coils systems. The voltage waveforms are realizable with the re-designed power supply system. This provides a baseline 15MA inductive scenario that satisfies the ITER design performance while remaining within the coil current and force limitations. The evolution of the resulting current density typical of the L- and H-mode phases is shown in Fig. 5. Sawteeth are present in both L-mode at 80s and H-mode at 300s, and result in the flattened current density profiles near the magnetic axis from a time-averaged sawtooth model. The pedestal bootstrap current peak near the edge is present in the H-mode phase. Also shown in Fig. 5 are the shapes controlled during rampup, burn and rampdown.

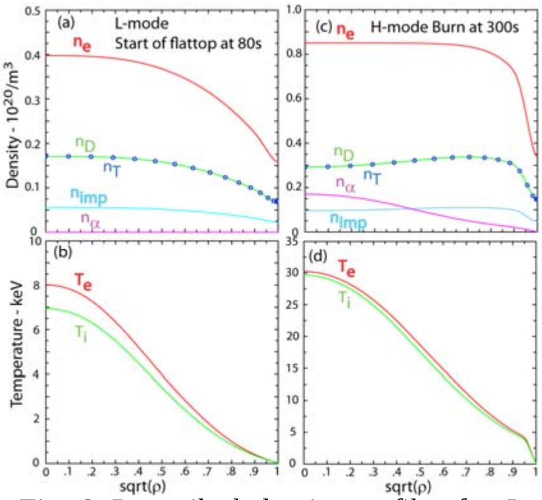

Fig. 3 Prescribed density profiles for L- and H-mode conditions. The alpha-particle density is obtained from a production rate equation with diffusion limiting the density buildup; D = 0.1χ_e. The temperature profiles shown are a result of the thermal transport using the Coppi-Tang model for χ_e, χ_i the electron and ion diffusivities.

4. 15MA inductive scenario performance variations

A series of scenario simulations were completed to explore sensitivity to the assumptions of the H-mode edge pedestal scaling. These simulations are done with the CORSICA backing-out mode [9] that provides for efficient computation of the scenario evolution for plasma parameter variation studies. With the shape control provided by a prescribed set of fiducial boundaries, control of the vertical instability with the feedback controller is not required. This mode of operation allows for systematic variation of parameters while still providing most of the information on plasma performance operational limits such as the coil currents needed to achieve the given shape variation and the forces on the coils. Only power supply voltage demands and controller performance cannot be assessed. Since these simulations include the external circuits, they are used to provide the open loop coil currents and flux gap variations required for forward control using the controller. Control via the gap flux variation and vertical stability control must then be verified in additional simulations such as that shown in the previous section. In previous simulations [9], it has been shown that these results remain close to the forward simulations.

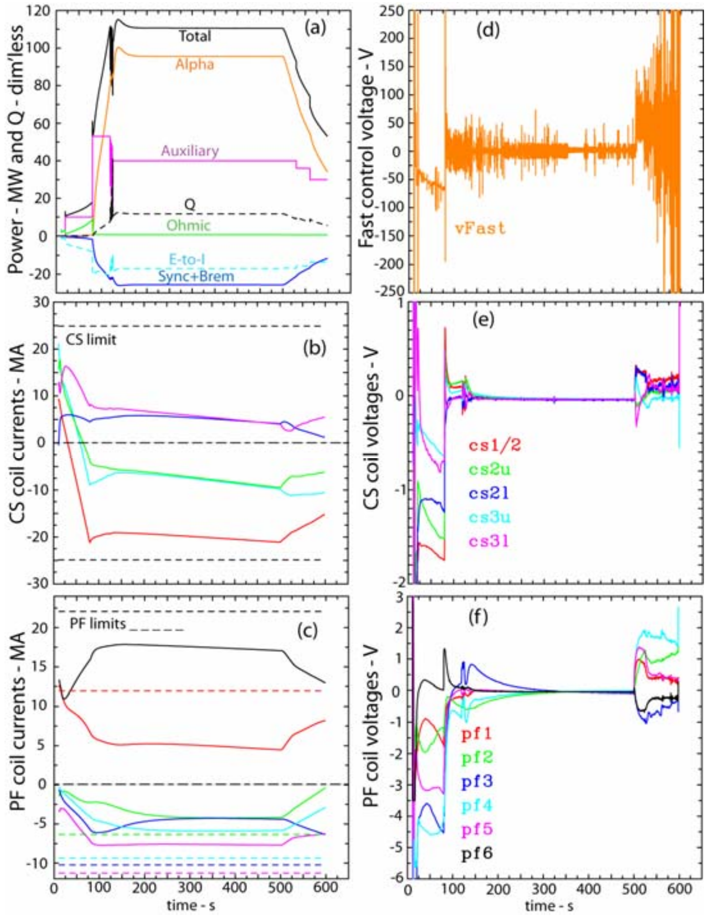

Fig.4 Free-boundary control simulation using the VS1 control circuit showing (a) performance achieved, Q~10 at P_{fusion} ~ 500MW and the resulting coil currents with (b) the CS coil currents and (c) the PF coils (in MA turns). The VS1 converters voltage is shown in (d) with the slower plasma current and shape control voltages on the coils in (e) for the CS coils and (f) for the PF coils (in V per turns).

We have varied these edge transport conditions to scan the pedestal height over a range consistent with expectations of time-average stability to edge localized mode (ELM) phenomena. This was done to span the expected range of performance on ITER to evaluate the likelihood of potential operating space difficulties. We note that, for the range of pedestal temperatures simulated, 3.5keV to 6.5keV, we can achieve full duration burn with, of course, a variation in the P_{fusion} obtained. In Fig. 6, we show results of one of the backing-out simulations with the same waveform design used in the 4.5keV pedestal case with forward feedback control already discussed. We show the plasma current waveform with the resulting evolution of l_{i3} and the vertical instability growth rate. The sensitivity to vertical instability as characterized by l_{i3} is shown if Fig 6a along with the vertical instability growth rate. The higher values of l_{i3} for an elongated plasma

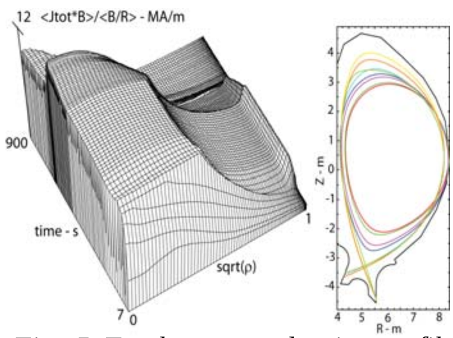

Fig. 5 Total current density profile versus toroidal flux and time showing the evolution due to sawteeth the edge pedestal current density. Shapes controlled during simulated evolution.

are potentially more difficult to control. These simulations were run to a longer time in ramp down than the forward control case as we are currently designing the possibility for a wall-limited ramp down. We show a comparison of the CS and PF coil currents obtained for this case with those obtained in the free-boundary simulation shown in the previous section. We note the consistency of the resulting coil current evolution indicating that these parameter studies well characterize the scenario evolution expected. We are still evaluating the control scenario for late in the current ramp. Also shown in Fig 6d are vertical forces on the CS coils. We show both the coil separation force and the average force and note that both of these remain within the design limits for ITER.

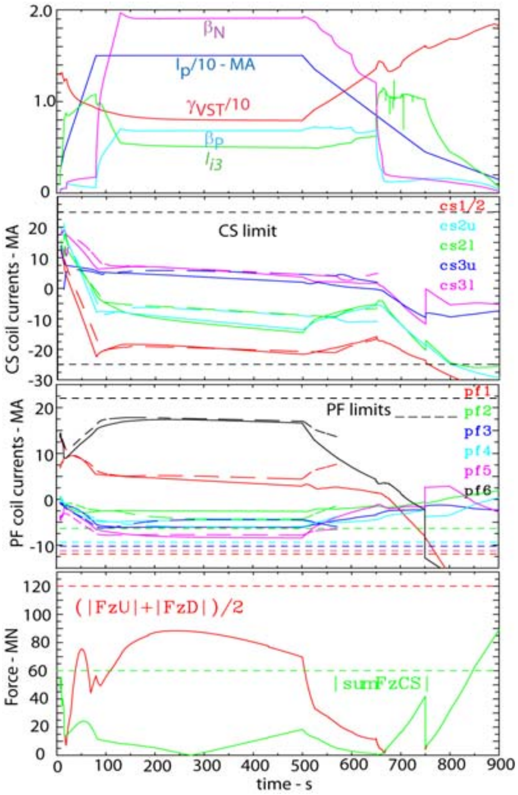

Fig. 6 (a) Stability parameters, CS (b) and PF (c) coil currents where solid is the backing out and long dash is forward control and the resulting forces on the CS (d) for the 4.5keV pedestal case. The difference in coil currents indicates small differences in the shape which are exactly controlled when backing out but controlled by 6 gaps in the forward case.

In Fig. 7, we show the evolution of the scenario studies superimposed on the statically evaluated operating space boundaries from the CORSICA equilibrium studies for burn conditions (black dashed lines in Fig. 2). We note that the entire range of scenario evolution for the pedestal temperature variation remain within the allowed limits for ITER. This indicates we have a fair amount of flexibility in operating scenarios with respect to varying performance levels determined by the pedestal conditions. We have recently initiated studies of the low- and non-activation operation of the ITER experiment and for inside-wall startup. Preliminary CORSICA simulations with deuterium only (no tritium, low activation) and with hydrogen have been completed and are included in the operating space diagram. Without the additional heating power due to the alpha particle production, this operation is limited to shorter pulse durations due to the need for Ohmic heating to sustain the plasma. The times shown in Fig. 7 indicate the maximum pulse duration allow and is limited by the CS1 flux capability. While the pulse durations are significantly shorter than full DT operation, they appear to be adequate for the initial phase of ITER when subsystems are commissioned.

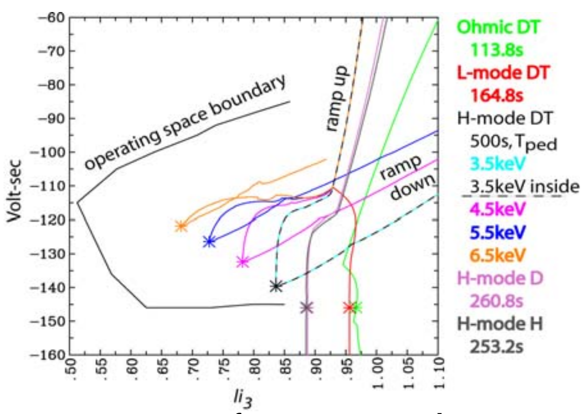

Fig 7. Mapping of scenarios onto the operating space limits for burn conditions indicates a range of successful operation for ITER

5. DINA current ramp exploration

Recent simulations with the DINA code have concentrated on the study of operational ranges of the rates of plasma current ramp up and ramp down in the 15 MA scenario. They are also developing plasma shape variations during the current ramp down in diverted magnetic configuration untill the current is reduce to about one-tenth the flattop value. Simulations were performed using feedback control of plasma current, position and shape, taking into account engineering limits imposed on the coils and their power supplies (maximum currents, voltages, magnetic fields, forces and total power of the converters). Stabilization of plasma vertical displacement, Z, was performed by VS1 controller (6 kV on-load voltages) assuming noise in the dZ/dt diagnostics with a uniform spectrum and RMS value 0.2 m/s in the bandwidth 0-1 kHz. (Stabilization by the in-vessel coils, VS3, was also considered in some DINA simulations.) The plasma current ramp up was performed with early transition to diverted configuration (at ≈3.5 MA). It was shown that the fastest current ramp up can be performed during about 50 s (limited by the voltages available for plasma current and shape control). Feedback control of the internal inductance, l_{i3}, by variation of the plasma current ramp up rate was also demonstrated in DINA simulations [15]. This method was experimentally proved in DIII-D [10].

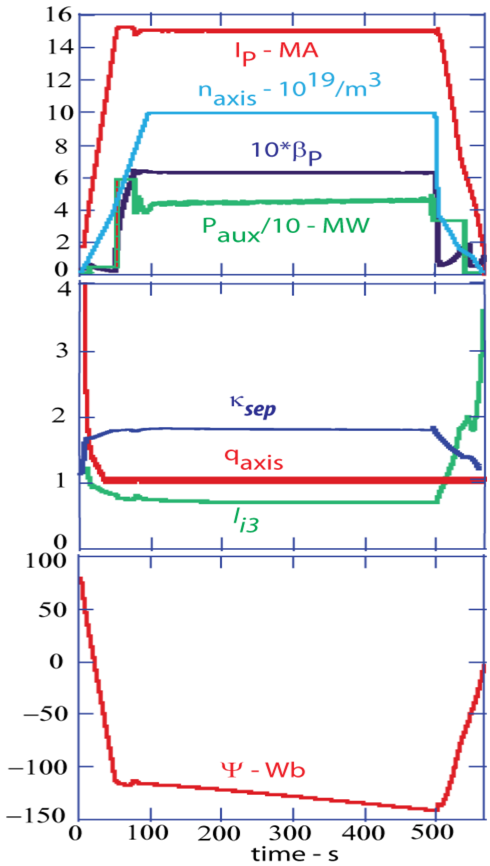

Fig 8 Parameters from the extreme maximum current ramp scenario

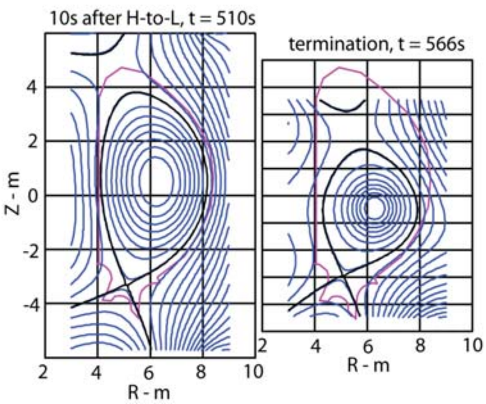

Fig 9. Shape evolution in ramp down of elongation with current (10kA, 1.4kA)

Nominal plasma current ramp down for the 15 MA scenario developed in DINA simulations has the following two phases: 1) plasma current ramp down in H-mode from 15 to 10 MA in diverted magnetic configuration consistent with reduction of plasma elongation keeping $q_{95} \approx 3$, then 2) plasma current ramp down in L-mode in diverted configuration to 1.4 MA. DINA simulations demonstrate the PF system can perform plasma current ramp down during the time between 60 s (voltage limited) and 300 s (current limited). Scenarios of this type have been successfully validated in DIII-D [3]. The DINA simulation shown in Fig 8 is considered to be an extreme case scenario. The current is ramped from 2 to 15MA in 50s, the fastest achieved to date. Rather than being limited by current or forces on the coils or by vertical stabilization demands, this case is limited by the power supply voltage capabilities required to drive the Ohmic plasma current and for shape control. It is characterized by central, inboard startup with formation of the diverted configuration at 3.5MA (9.5s) followed by 4MW of heating during the ramp. Auxiliary heating of 52MW is used to initiate the plasma burn starting at 75s to achieve burn duration of 425s terminated by an H-to-L transition and current ramp-down in 66 s. In Fig 9, we show the programmed shape evolution for this simulation with the elongation ramp starting at 500s and the reduction in density as shown in Fig. 8.

6. Summary

We have completed several simulations using two free-boundary transport simulation codes, CORSICA and DINA, for the re-designed ITER coils. These simulations indicate that ITER should achieve its operational mission. The free-boundary transport simulations with control demonstrate the controllability of the ITER plasma. Several issues must yet be addressed that include different operating modes (hybrid, steady-state) and advanced controllers with improved capabilities. These simulations use a prescribed density profile evolution and the Coppi-Tang thermal transport model to evolve the plasma for the baseline 15MA inductive scenario. We find that the vertical position and shape are controlled with the VS1 circuit that uses only the outer PF coils for control and the CS coils for Ohmic current drive. We have re-evaluated the available operating space for the recent modifications to the ITER coil and first wall geometries and have found small reductions in the available operating space. However, the time-dependent simulations remain within this operating space and indicate that ITER should be able to achieve its 15MA mission with the systems as designed. We have also developed a viable ramp-down scenario for the 15MA inductive case that was validated experimentally. In addition, we have varied current ramp rates and achieved 50s ramp up and 60s ramp down times limited only by the voltage capabilities of the power supply systems and not by vertical stability. These time-dependent simulations are consistent with the operating space boundaries obtained by parameter variations in static equilibria to determine operating limits due to coil currents and forces. These time-dependent simulations represent the current state-of-the-art in producing scenarios representative of experimental operations

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