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ABSTRACT

This paper introduces those plasma control concepts which are likely to change when going from a 500-1000 second tokamak pulse to the discharges in a steady state tokamak, whose design relies on the reduced number of thermal and mechanical stress duty cycles.

1. INTRODUCTION

We choose to define a steady state tokamak discharge as one which lasts during the daily rhythm of the electrical supply industry demand, rather than to any physics and engineering limitations of the device itself. The major technical advantage of steady state operation, beyond the constant availability of fusion power, will be the reduction of the number of thermal and mechanical cycles which the design has to tolerate. The design then becomes cheaper, but less resistant.

As a tokamak extends its operation to longer and longer pulses, the most technically sensitive areas are likely to be:

- continuous and pulsed divertor erosion
- prolonged wall contact and hot-spots
- continuous operation of the superconducting coils.

All are important since they are not in steady state in the longest high performance pulsed tokamaks operating today. A far-reaching consequence of steady state operation is that the total number of pulses simply decreases, leading to

- a reduced number of discharges in the overall steady state program
- a reduced number of learning shots in preparation for steady state operation

By tokamak control we only consider the control of the plasma itself and its related components. The supervisory plant control for auxiliary equipment is relatively conventional, controling ultra-high vacuum systems, high power electrical supplies, equipment heating and cooling systems, auxiliary systems for heating the plasma and all diagnostic systems. Whether this equipment is operating in steady state or pulsed for 5000 seconds makes little difference.

Present experience of steady state tokamak operation is limited to long pulse non-inductive current drive on the TRIAM and TRIAM-1M tokamaks using Lower Hybrid waves for over 2 hours of uninterrupted plasma operation [1]. Apart from this proof of principle experiment, our knowledge of steady state tokamak control is derived from extrapolating from pulsed tokamaks and from computer simulations. Most technical issues associated with steady state tokamak operation appear as control aspects of separate physics or engineering issues.

In this paper, we select some particular issues related to the transition from 500-1000 second pulses towards steady state. In Section 2 we choose a definition of steady state, looking at the different time scales involved in tokamak operation. We present a brief overview of the evolution of plasma control towards longer and longer pulses in Section 3. We examine the parameters which will have to be controlled in Section 4, together with the measurements which will have to be made. The required actuators are discussed in Section 5. In Section 6 we address the question of controllability and the extension of controllability to steady state is mentioned in Section 8. Modelling is discussed in Section 7. Two major issues which relate the technical systems to plasma control are maintaining equilibrium control using superconducting coils, and handling the large first-wall heat fluxes, to which Sections 9 and 10 are devoted. An issue regularly raised concerns intelligent control for tokamak optimisation, presented in Section 11. Technical implementation of steady state plasma control is a minor technical issue, mentioned in Section 12. Section 13 closes the paper with a discussion. Since this paper was written for a workshop which addresses many of these issues in great detail, neither the substance nor the references makes any claim to be a review of the subject. The examples taken are purely illustrative.

2. STEADY STATE

What do we mean by the steady state control of a tokamak? Steady state tokamaks will remain pulsed in nature and will retain an important ability to be shut down on demand. We could simply define steady-state operation as pulsed discharges of which the duration is determined by the electrical supply industry, matching the output to the consumer demand. This will mean pulse lengths in excess of 12 hours. Such a pulse length is considerably longer than the longest characteristic times of the device itself. The characteristic time scales of a tokamak generally increase according to the following list:

- Microinstability period and growth time
- Vertical instability growth time
- Energy confinement time
- Plasma shape and divertor geometry control time
- Particle confinement time
- Current profile modification time
- Current profile relaxation time
- Divertor thermal equilibrium time
- First wall thermal equilibrium and out-gassing time
- Superconducting magnet cooling transit time
- Superconducting magnet cooling characteristic time

Some of the tokamak auxiliary systems will also run into a transition from pulsed to steady state as the pulse length increases beyond 10 minutes. An example is the Neutral Beam Injection cryogenic system. However, most auxiliary systems are already steady state for pulses of 1 minute. The question is often raised about the steady state character of the 500 second ITER design. The ITER-FDR design is about as transient as the modest TCV and C-MOD tokamaks for the plasma parameter time-scales, normalised to the pulse length. The longest pulses are a few times the plasma current L/R resistive decay time during the burn and a few thousand energy confinement times. This implies that from the point of view of control, the nominal ITER pulse length could still benefit from the type of pulse optimisation techniques used on present tokamaks. When increasing the pulse length beyond 500-1000 seconds, issues related to genuine steady state will appear, specifically in the last five characteristic time scales listed.

3. Evolution of Plasma Control Techniques

Early tokamaks were quite primitive. The desired plasma parameters were obtained as a result of pre-programmed power-supply commands or gas-valve demands, designed by trial and error. As the devices developed, the plasma pulses lengthened from milliseconds to 10's of milliseconds and feedback loops were developed to control simple parameters, particularly the plasma position. This required a real time measurement of the plasma position and an algorithm for modifying the pre-programmed voltages applied to the power supplies. As pulse lengths evolved to 100's of milliseconds, there was time to modify and later control the plasma density using fast gas injection valves. These simple feedback loops were all operational by the end of the 1970's. Additional heating systems were added, usually either off or fully on, and their effect on the plasma was systematically studied. Alternative methods of fuelling the plasma were tested, especially pellet injection in a pre-programmed sequence. Around the same time, non-circular plasma cross-sections were proposed on the basis of theory and the first plasmas with vertically elongated cross-sections were created. These plasmas are vertically unstable, the stabilising feedback control of stable plasmas had to be mastered and another step forward was made.
During the 1980's and 1990's, the effort into increasing the temperature and confinement of the tokamak plasmas had revealed still more complex behaviour. The plasma could bifurcate into regimes with better or worse confinement. The question was why? Even today, this question is not fully answered and the control of these enhanced confinement regimes remains an experimental challenge. Initially, enhanced operation appeared as a transient effect on a single experiment, became generalised to several experiments and then effort was put into maintaining the plasma in the desired enhanced state.
The ITER project will bring more challenges to plasma control, even with 500 second pulses. All plasma facing surface components will be in thermal equilibrium, with power flows from the plasma of the order of 5-10MW/m^2. Such power flows are potentially destructive and must be reduced to a tolerable level by radiating away the power before it hits a material surface. The control of the plasma position and shape will have to be extremely precise, due to the energy density of the plasma which cannot be allowed to touch the external walls for more than a half a second. The rate of fusion reactions will have to be controlled.
Extending the pulse length to steady state will be a natural extension to the progress made over the last decades, contuing the transition of plasma control from the original role of parameter adjustment to a more subtle role of plasma performance adjustment and ultimately plasma performance optimisation.

4. Parameters to be Controlled and Measured

All the control parameters for "conventional" pulsed operation will also have to be controlled during steady state operation. These include the plasma current, the plasma cross-sectional shape, the unstable vertical position, the divertor geometry, divertor density, temperatures and radiation, plasma core density, temperatures, impurity radiation and fusion reaction rate. Since steady-state operation will require a maximum bootstrap current fraction due to the inefficiency of non-inductive current drive, the pressure profile will almost certainly have to be actively controlled. The current profile may not be in a steady state or in the optimal state with respect to flux diffusion and may also have to be dynamically controlled. Precise adjustment of the current profile at some critical flux surfaces may be necessary to maintain any internal transport barriers required for steady state operation and to avoid MHD instabilities. Sawteeth may need to be controlled, or may not exist in the chosen scenario; neo-classical tearing modes may have to be actively controlled; resistive wall modes may need active stabilisation; TAE modes may need tailoring of the current profile; resistive tearing modes are sensitive to the precise current profile. Present experiments "control" the current, pressure and impurity profiles using different transient techniques "tuned" by trial and error, an inadequate approach for steady state closed-loop operation. Tore-Supra has shown the difficulty of implementing true parameter control of the current profile [2]. Requiring four to five different properties of the current profile to be controlled simultaneously would be very ambitious and there is no present evidence for its feasibility, except perhaps with Electron Cyclotron Current Drive which is rather inefficient.

To implement a feedback loop, it is essential that the control parameters be measured directly, or measured indirectly using an adequate model. This can be a static or dynamic map between all measurements and a given parameter, including measurement and interpretative uncertainties. When operation is extended from 500-1000 second pulses to steady state, several measurement issues arise. The reliability of magnetic measurements requiring integration during the full plasma pulse has already attracted considerable attention and should be adequate out to 1000 seconds. However, this is a specific example of a general problem of maintaining the precision of all diagnostics during a 12 hour burn. Many effects which affect long term drift of diagnostic calibration will occur not only as the number of discharges accumulates, but also during a single discharge. Neutron and gamma radiation affects the transmission and noise level of electrical and fibre-optic cabling as well as detectors and electronic equipment. Whereas electronic equipment might be positioned far enough away for short pulses, steady state will increase the fluence per pulse by two orders of magnitude. Erosion and deposition in steady state will affect mirrors and windows, however well they are concealed in the first wall. The thermal working of the whole tokamak structure, limited to the first wall surface for short pulses, may affect the alignment and calibration of all line of sight measurements.

Combining diagnostics with different low frequency and high frequency error sources could provide a suitable compromise. An example is the absence of low frequency drift in a reflectometer providing the low frequency component of the separatrix location at some reference points. A development in the interpretation of multiple diagnostics and appreciation of their internal consistency will be required in real-time. The increase in the pulse length is in our favour for additional computational requirements, as is the increase of most characteristic times.

Diagnostics will require redundancy if they are part of a feedback loop or general optimisation system. An example of the effect of losing the fusion power signal without any intelligent warning was presented recently [3], illustrating the necessity of redundancy and consistency checks on all measurements. The Mean Time Between Failures must increase with the pulse length in order to maintain a fixed probability of achieving the designed plasma discharge. Methods of ensuring this will involve improving the reliability of individual components, but will also involve duplication of sensors, diagnostic methods and control algorithms to provide redundancy and cross-checking. This is an issue in many high technology fields, since a regular rhythm of false alarms hiding real alarms is one result of simply increasing the redundancy. Artificial Intelligence techniques are being refined to fulfil the role of fault-finding, alarm checking and alarm validation. A relevant example is the work on fault-appreciation of the mechanical stress sensors on the JET tokamak [4].

Measurements of the control parameters for steady state will have to have calibration and reliability techniques built into them to provide at least two orders of magnitude of reliability compared with a 500 second burn. This will require substantial effort. On the other hand, the precision of the measurements needed for control will not have significantly more severe requirements.

5. ACTUATORS

The actuators required for steady state operation are the same as for long pulse operation. Electrical power supplies are required for the toroidal and poloidal field systems to control the plasma current, shape and position. Gas or pellet fuelling of D-T and impurities in the main chamber and divertor region control the density and radiation. Neutral particle and radio-frequency beams are used for current-drive, current profile and pressure profile control as well as fusion power control. The pressure profile control indirectly controls the current profile, by the bootstrap effect, adding the requirement that the radial deposition profile for the auxiliary systems should be tuneable or at least mixable. The new requirement is that each system should function in a steady state, imposing a requirement on the design of each system, but not creating a new technical issue for overall control. The issues which will appear in steady state are again those of redundancy and reliability. A 12 hour pulse is long enough for backup solutions to be provided in the case of an actuator dropout. Operation might be maintained temporarily, perhaps at reduced power output, while an actuator is recovered. The time scales allow maintenance, replacement and repair of components outside the tokamak installation during a pulse. Such an approach has to be incorporated into the overall plasma control concept but does not require any specific equipment development and is a "soft" development. Maintaining the pulse integrity is particularly important to obtain the major advantage of steady state operation, namely the reduction in the mechanical and thermal cycling. Just allowing the plasma operation to terminate when losing actuators or measurements would not be acceptable if operation could have been maintained. This will require a considerable increase in the complexity of a plasma control management system. Given the long time scales involved, human ratification of proposed corrective actions could be included.

Present plasma control systems already incorporate redundancy in plasma energy feedback loops. A Neutral Beam Injector dropping out does not invalidate the energy or fusion power output control, but causes an increase in the demand to other beams. At the other extreme, losing the toroidal field would automatically lead to a shutdown. Many actuator systems could be built with useful redundancy based on an economical trade off. Other systems may have partial redundancy, such as the Poloidal Field coils, since we might be able to retain reduced operation with the remaining actuators while a power supply recovered normal operation.

A final point concerning the actuators is the circulating power, which is one of the most important aspects of steady state operation. Realistic operation will impose an upper limit on the circulating power, defining the Q of the plant, rather than the popular Q of the plasma. Since the actuators vary from efficient magnets for equilibrium control to inefficient generators for current and pressure profile modifications, we have to be careful not to invoke steady state scenarios which are too dependent on high circulating power. There will be an upper limit on the credible "wall-socket" electrical power which can be used for the more ambitious control techniques, especially if several MHD modes require multi-point current profile tailoring.

6. CONTROLLABILITY

To implement feedback control, we must consider additional issues. The relationship between a given actuator and a particular parameter we wish to control is rarely simple. It is not an overstatement that there is no parameter to be controlled which depends on a single actuator. All parameters and actuators are coupled, represented as an influence matrix, Fig. 1. This diagram does not indicate the strength of the coupling between actuators and parameters, but simply indicates the complexity of the problem. Plasma control has evolved by considering parts of this influence matrix as being decoupled. The cleanest separation has been between the electromagnetic control of the plasma equilibrium and the gas/pellet/beams/RF control of the "kinetic" or internal properties of the plasma. This quasi-decoupling has been useful since it allowed the equilibrium control to be optimised while neglecting the less well understood kinetic control.

Today, this separation is disappearing. The internal parameters of plasma pressure and internal inductance are important in optimising the equilibrium control. Plasma current is no longer an "electromagnetic" control parameter but depends on bootstrap current and non-inductive driven current. Obtaining the enhanced confinement regimes requires a precise knowledge and control of the plasma equilibrium. Coupled to this is the ability/tendency of the plasma to bifurcate its behaviour. Correction for such effects will require actions on both kinetic and electromagnetic actuators. An example is the still badly understood links between the separatrix shape, the boundary flux expansion and the neutral gas pressure on the ELM-free/ELMy H-mode transition. In view of these points, we must consider the control problem to be coupled, even if approximate decoupling is an essential tool for commissioning and a useful tool for establishing basic tokamak performance, avoiding the need to rely on a too complex and perhaps dubious model during early operation.

Different parameters can exhibit stable or unstable behaviour. In the former case, applying no feedback allows the parameter to tend towards a fixed operating point (such as the plasma current), whereas in the latter case, the parameter will continuously drift away from the operating point (such as the vertical position of an elongated plasma). In both cases, the feedback loops serve to obtain an operating point different from the natural operating point. There has been much work carried out on the stability or instability of the fusion power in an ignited plasma which will finally depend on transport mechanisms which are not yet fully determined. The transport of α-particle power, the dependence of any internal transport barriers on α-particle power, the MHD effect of α-particle power are all effects which will determine the ultimate stability of the fusion burn. The divergence of past result is simply due to the assumptions made in the modelling. If the steady state reactor retains a driven element, then the burn stability is guaranteed by definition. The stability of the plasma density depends on the recycling and out-gassing of the walls under very long pulse conditions, which can be modelled, but the answer is a property of the model assumptions. The stability of the current profile with bootstrap current and α-particle power has to be answered, and will be linked to the stability of the internal transport barrier. The stability of the impurity content under long pulse conditions will remain critical, since varying the heat load by varying the radiated power which then changes the out-gassing and impurity fuelling could give rise to stable and unstable behaviour.

The fuelling, fusion power and impurity stable/unstable behaviour will give rise to questions of controllability. For any system, there is a region of state space in which the system remains controllable, given a set of actuator limits. These three examples share a common feature of having zero or almost zero as the lower actuator limit, whereas vertical position stabilisation and current profile stabilisation can call upon positive and negative actuator actions. The stability boundary for these three parameters will have to be examined with particular care in the low frequency limits. Optimisation of control within fixed actuator limits is already being addressed in the case of vertical control, where new methods can provide better control in extreme circumstances than a well-tuned linear controller [5].

7. Modelling

Assuming that the plasma plus its actuators constitutes a system which is normally within its stability limits, we are in a position to design a feedback controller. For this to be effective, a model of the system to be controlled and a model of the actuators are required. The time-variance of the system and perhaps of the actuators will require an adaptive approach to feedback loop design in steady state pulses. Three types of modelling are common for controller design. The first, "White Box" modelling, takes plasma parameters and derives a physical model of the system. The third, "Black Box" modelling, takes data from the inputs and outputs of the system and derives a purely observed, mathematical, model of the system. The intermediate technique, "Grey Box" modelling, takes a white box model and uses measurements to refine some model parameters. The most advanced simulation codes such as TSC provide an example of grey box modelling. The relatively simple case of plasma equilibrium response modelling has examples of all three approaches applied to the TCV tokamak [6,7,8,9]. Figure 2 shows an example of plasma equilibrium response data from two white box models, a grey box model and the measured response. The measurements have a smaller variance than the model difference and the grey box model can provide a more accurate representation of the data. The plasma equilibrium response modelling is a poor example for comparing these techniques, since the white box models are already good. A better example of black box modelling is the identification of the plasma response of kinetic parameter control on JT-60U [10,11]. Modulation of the actuators allowed an Iput/Output model to be generated, including the cross-coupling between the actuators and the plasma parameters. Figure 3 illustrates the separate and simultaneous closed loop control on JT-60U of three kinetic parameters, fusion power, divertor radiated power fraction and core density, using a controller designed on this observed model of the kinetic parameter coupling, in an ELMy H-mode plasma. The key issue is to explore the dependence of the response matrix and the controller on the geometry, since this control technique is very sensitive to the position of the X-point, for example.

The remaining element required for designing a feedback controller is a model of the noise and perturbations to be rejected by the loop. This is difficult due to the lack of white box models of the noise and disturbances and the difficulty of extrapolating existing quantitative models of noise and disturbances to a future device. In steady state, the noise and disturbances will determine the ultimate performance of the control. The feedback loop can always be optimised on the basis of measurements of the noise, but the amplitude and frequency of the largest disturbances can be a significant factor in the design of the actuators. A particular case is the maximum PF coil voltages and system power for rejecting the largest plasma equilibrium disturbances, namely minor disruptions, giant sawteeth or large ELMs. The frequency of these disturbances affects the specification of the superconducting coil AC losses cooling (Section 9). The most blatant example is in the design of the controller for reversed-shear discharges, for which the accumulated knowledge of regular or infrequent disturbances is simply inadequate. Since steady state operation might rely on such a scenario, this issue needs to be addressed.

8. Can transient control always be extrapolated to steady state?

In present experiments, the control of the pressure or current profile frequently involves transient techniques or is successful for a duration less than the relevant characteristic times. Whether control can be achieved in steady state therefore has to be addressed. The demonstrated controllability of a system at finite frequency and finite power does not mean that this system will be controllable at lower frequency and three particular cases arise. In the first, the transfer function tends towards zero gain at zero frequency. Such a system will normally be associated with an infinite actuator requirement as the frequency approaches zero, in either a Single Input Single Output or Multiple Input Multiple Output (MIMO) system. An obvious example is the control of the plasma current by the primary transformer. The second case is more subtle and corresponds to at least two outputs of a MIMO system becoming linearly dependent as the frequency tends towards zero, although independent at higher frequency. A simple physical example is a MIMO electrical circuit with an inductive element somewhere between two of the outputs. These outputs are separately controllable at high frequency, but as the frequency drops, the power required to control the outputs individually explodes. It is tempting to ask under what conditions the bootstrap current and current profile diffusion might constitute such a system, and whether present experiments can be considered as low frequency demonstrations or high frequency demonstrations. Alternatively we can ask what the plasma physics constraints would have to be to avoid an increasing power requirement at very low frequency. The third case is benign, namely one in which the low frequency limit is fully controllable where the high frequency control problem requires increasing power, of which the control of q(0) is an example [2,12].

Maintaining a prescribed current profile is being addressed in detail on Tore Supra, together with identification of the current profile with the required precision using the available measurements [2,13]. Modifying the current profile presently uses a mixture of non-inductive current drive and inductive current drive. Few experiments obtain a stationary loop voltage over the complete plasma radius. In steady state, the current profile will be determined uniquely by the non-inductive driven and bootstrap components. The non-inductive current drive must then control the full radial profile, not just modify the underlying profile, to avoid a simple canonical non-inductive profile. A mixture of current drive profiles or a tuneable driven profile will be required to avoid this, with adequate efficiency. Whether the current profile will be stable or unstable in the presence of non-inductive current drive will also have to be determined, manifested by either returning to a sawtoothing discharge with q(0)=1 or losing the central current with off-axis drive. One issue in modelling current drive, compared with global parameters, is the diffusive nature of the physical process, excluding a simple low-order lumped model for which a simple low-order controller can be designed. The underlying questions are more of a physics issue than a technical issue, but the difficulty of generating a model is a technical issue, certainly requiring more work.

Maintaining a spatially fixed Internal Transport Barrier will be a challenge. It is not certain whether the underlying physics will allow steady state control of the ITB footprint, or whether the ITB will "flutter" in time and in space executing a sort of limit cycle [14], or whether the ITB will simply be uncontrollable at low frequency. The answer will depend on the local parametric dependence of the energy transport and the ITB criteria, as well as the local parametric dependence of the current/pressure profile deposition.

9. SUPERCONDUCTING COIL ISSUES FOR CONTROL

Can the tokamak subsystems themselves operate in steady state, when controlling the plasma? We must be certain that the control actions are compatible with steady state operation. The most important issue here is the accumulation of AC losses in the superconducting coils due to continuous equilibrium control actions. The time constants of the superconducting coil cooling are the largest to be considered from the point of view of control. Steady state operation is the most natural mode for superconducting magnets. The typical loads affecting any superconducting coil include the thermal conduction through the current leads and gravity supports, the radiation load at the surface, the joint power and the Joule-Thompson expansion in case of forced flow windings. In tokamak magnets, two additional sources of thermal load must be taken into account: nuclear heating and AC losses.

Nuclear heating is the major load and a driving design parameter for the toroidal field coils. By adjusting the thickness of the nuclear shield, the thermal load at the toroidal windings can be varied over a broad range. A thin shield is attractive as it reduces the major radius and the cost of the fusion device. However, a large nuclear heat implies a higher coil operating temperature and hence a larger cross-section of superconductor.

In large, force-flow cooled toroidal field coils, the residence time of the coolant in the winding is of the order of 10-30 minutes, given by the hydraulic length of the winding (e.g. about 700m in ITER-FDR) divided by the average helium speed (typically less than 1m/s). Any plasma burn longer than the residence time of the coolant in the toroidal winding can be considered "steady state" for the issue of the accumulation of nuclear heating.

To control the plasma current, the magnetic geometry and the vertical position, continuous field variations are generated by the central solenoid and other poloidal field coils. The feedback system also rejects noise and disturbances. These time varying fields cause AC losses in the superconducting magnets, both as eddy current loss in the normal metallic components and hysteresis and coupling current losses in the active superconductor. The hysteresis loss is due to the irreversible magnetisation of the superconducting filaments. The loss per unit volume is proportional to the filament size. The hysteresis loss is large at the low field sections of the winding, i.e. when the diamagnetism of the superconductor is strong, and becomes negligible at high background field. The hysteresis energy loss only depends on the amplitude and direction of the field change, ΔB, and is independent of the rate of change of the field, dB/dt. The coupling currents loss occurs in multi-filament composites and multistrand cables exposed to transverse field changes. Similar to the eddy current loss in normal metal, the coupling loss energy depends on both the amplitude and the rate of the changing field. The coupling loss in large superconductors can be reduced by twisting the cable components and adjusting the transverse resistance in the cable. However, if the transverse resistance is too high, the current re-distribution among the cable elements is hampered and the stability of the superconductor is affected.

AC losses in the toroidal field coils are not the driving design criterion. The average thermal load is small compared with the nuclear heat and the peak load is small compared with that of the coils which generate the field changes. In the central solenoid, with a large ramp rate for plasma initiation, the coupling losses must be tightly controlled. The coolant residence time is very long compared with plasma initiation and the related AC losses must be absorbed adiabatically in the winding. However, for burn time longer than the residence time, there is no further issue of loss accumulation.

In the poloidal field coils, depending on the frequency of the control actions, both the hysteresis and the coupling loss can be the key issue. The AC losses accumulation, together with the limited temperature margin of the NbTi conductor used in the low field poloidal coils, is a crucial design parameter. The range of amplitude and frequency of the control actions cannot be expanded for a given coil/conductor design.

Both AC losses and nuclear heat load expected in steady state operation can be accommodated in the design of superconducting coils by an adequate temperature margin (in this respect, the ITER-

FDR design is fully steady state). Further design margins must be built into the magnets, mostly the toroidal field coils, to withstand the abrupt field changes due to plasma disruption. The allocation of the correct margin is the crucial engineering issue for a reliable design and an affordable cost.

Last but not least, the most challenging aspect in the superconducting coil technology for steady state operation is the radiation resistance of the electrical insulation of the toroidal field coils. The epoxy resin potting of the wrapped superconducting windings is a well-established technology, which provides excellent mechanical properties of the winding pack. However, the integrated radiation dose over the tokamak lifetime is expected to seriously damage the organic component of the electrical insulation, with degradation of the mechanical properties below the design requirement. Either the development of a non-organic insulation system or the substantial improvement of the radiation resistance of organic systems (together with a thicker radiation shield) may be necessary to extend the lifetime of the toroidal field coils. The actual resistance of a superconducting coil to the combined radiation and electro-mechanical loads can not be investigated in present fusion devices. All the experimental results are obtained by irradiation of small, sandwiched specimens in the core of fission reactors.

To summarise, the AC losses are not a limiting factor, but must be kept low to maintain a low, effective operating temperature at a reasonable heat removal rate. The design assumptions depend on the disturbance model. Since AC losses due to excessive disturbances compared with the design cooling rate could cause a slow build-up of heat, operation could be terminated when the design budget is exceeded, but this would be contradictory to maintaining a steady state.

10. PLASMA SURFACE INTERACTION ISSUES

The greatest challenge of steady state from the point of view of plasma-materials interactions arises from the significantly higher power loading expected during both normal and off-normal events such as disruptions and ELMs and the problem of the trapping, reemission and retention of Tritium [15]. In ITER (FDR), the parallel power flux density flowing in the scrape off layer is expected to exceed 2000 MWm^{-2}, compared with 100-200 MWm^{-2} in JET. This leads to peak power loads at the divertor targets of between 20-30 MWm^{-2}, beyond what can be handled by conventional materials for targets with adequate erosion lifetime. Realistic steady state operation will only be possible if these power flux densities can be reduced to values in the range of 5-10 MWm^{-2}. Current strategy calls for distributed radiation in the core-mantle, scrape-off layer and divertor regions of the plasma using injected noble gas species, and re-circulation of neutrals in the divertor volume to deposit charge-exchange energy on the divertor side walls [16]. Both techniques will have a significant impact on the level of detachment at the divertor targets, which will in turn determine target erosion and re-deposition of tritiated layers. The injection of seed impurities will use pellet injection or gas puff into the edge, probably at many locations simultaneously. This will have to be precisely feedback controlled in order that the power flux across the separatrix is maintained above the H-mode threshold while still providing adequate power exhaust at the edge. Control of the divertor neutral density and additional recycling sources is also critical for control of the plasma density and thus of fusion power output. Such control is achieved through a combination of pumping and fuelling. The former serves to remove helium ash and effect some control of the plasma density, while the latter is required for plasma start-up and to replenish the DT fuel species both consumed during burn and pumped out with the helium. Fuelling rates, again via pellet injection or external gas puffing will also therefore require precise feedback control. Many of the issues related to this control, such as the sensitivity of core density to separatrix density and the effect of long pulses on wall and recycling fuel sources cannot be adequately studied in present machines and will also evolve from a 500 second burn to steady state.

Probably the most serious plasma-materials concern in large devices will be the erosion due to off-normal plasma events. Large ELMs deposit between 3-10% of the plasma stored energy in short time scales (0.1-1ms) with frequencies in the range of several Hz. Up to 80% of this energy loss will eventually strike the divertor targets leading to wide-scale melting and ablation and unacceptable target erosion. Methods must be found to control these events and reduce peak heat flux to plasma facing structures, particularly since sustaining sufficient core confinement in a reactor might require the edge plasma to operate in the very region of parameter space in which these ELMs are most common.

These first-wall issues already appear during a 500 second burn, but the philosophy that a certain budget should be allocated to off-normal events such as the ELMs and minor disruptions and that once this budget has been exceeded, the discharge would be closed down under full control is not acceptable, as for AC losses. Controlling the ELMs by global parameter adjustment is rather primitive and does not always work in practice. Identifying the non-linear ELM dynamics experimentally and looking for methods to reduce the occurrence of large ELMs is just starting and could have an important impact on steady state discharges [17].

11. INTELLIGENT CONTROL – FOR AND AGAINST?

The most demonstrative example of "expert" intelligent control in an existing tokamak has been provided on ASDEX-UG [18,19]. The High Recycling H-mode (HRH) can be lost and return to a High Recycling L-mode (HRL). A corrective action has to be provided to restore the desired operating regime. This cannot be pre-programmed, since the loss of the desired mode is not pre-programmed. The plasma control system has to be given the ability to determine its present operating mode and be given the freedom to provide a suitable corrective reflex action. Figure 4 illustrates this intelligent mode recognition and correction. The loss of the HRH regime is detected from a classification analysis of 14 diagnostic signals and a corrective action is provoked, via a modification to the reference signal specifying the radiated power fraction. The reference signal is allowed to recover its previous value after a pre-determined time.

Several issues have to be raised concerning increasing the level of intelligent control of a steady state tokamak. Firstly, the reduction in the number of discharges in the lifetime of the device mean that optimisation of the performance on a pulse by pulse rhythm would be ineffective. Secondly, the very long pulses mean that the control of the plant will certainly not be constant since important parameters will still drift during the discharge. Thirdly, the reliability of the complete plant must be very high to reduce the occurrence of random real and false alarms which would have a large effect on the total number of successful discharges.

The first point – a reduction in the total number of discharges – is trivial but far-reaching. Present tokamaks develop their control techniques from a large number of attempts, from measurements of the plasma parameters to be controlled, from intuitive reasoning and from trial and error. Although basic tokamak performance could be guaranteed with a minimum of trial and error, based on present experience, this is not obviously the case for advanced control for which the particularities of each device can be important. Restricting the total number of attempts to optimise the performance may lead to non-optimal operation. It is therefore tempting to consider as much on-line or real-time tuning as possible. Techniques to perform closed loop system identification of the plasma equilibrium on the TCV tokamak (Section 7) are perfectly adapted to on-line measurement and the calculations required can be easily executed on the time scale of long tokamak pulses. This opens the door to adaptive feedback control, during which the model of the system is permanently refined and the feedback controller is continually adapted to be at a prescribed optimum which can be guaranteed for stability using existing methods. Allowing adaptive tuning of well-understood parts of the tokamak could therefore be envisaged and the plasma equilibrium control provides a good example of a suitable starting point. Extending this to other areas of plasma control is more delicate, due to the less well-understood input-output relationship. This agues both for and against its implementation, since it would be more useful when successful, but the lack of understanding would make it more difficult to guarantee over a wide tuning range. Identifying the input-output model for kinetic control on the JT-60U tokamak (Section 7) allowed the influence matrix to be estimated, again perfect for real-time optimisation. It seems natural to extend these techniques to steady state plasma control, especially in view of the variation of the input-output relationships for kinetic control as the plasma facing components reach thermal equilibrium. This discussion naturally encompasses the second point of time-variation of the system to be controlled.

Adaptive control will provide a method of implementing more robust control of the plasma parameters, able to react to changing system parameters during the very long discharges. However, the better the control, the more the operation of the tokamak is restricted to the programmed parameters! This irony leads to a reduction in the variety of conditions explored, contrary to all ideas of genetic richness while searching for an optimal operation point. This apparent contradiction will have to be handled, since a reduction in the number of pulses will lead to a reduction in the richness of exploration, unless this richness is deliberately incorporated into the scenario planning. Techniques exist for tuning systems without an underlying algebraic model, exemplified by human tokamak operators up to now. Implementing such optimisation searches would appear to be a huge step forward. On the other hand, implementing them by operators is no less a step forward, and optimisation will certainly have to be carried out.

We have to digress here to debate the questions of automated optimisation and intelligent control, which have given rise to debate in the past. We must differente between the necessary, sufficient and optimal requirements. Tokamaks operate today on rather simple feedback control loops, based on rather simple understanding of the system to be controlled, they work well and they deliver the required performance. It is therefore tempting to define the current techniques as adequate and intelligent control as unnecessary. Beyond that, we could even consider that opening the debate on intelligent control demonstrates a lack of conviction in proven techniques and therefore illustrates weakness in defending a large project. On the other hand, the finesse with which modern tokamaks are now being optimised, to benefit from plasma profile control and its associated performance enhancement suggests that this will be a necessary condition for optimising tokamak performance and that simpler control methods will be inadequate. Finally, since the aim of developing a large steady state device is for operational and economic efficiency, only an optimised control would be acceptable in view of the investment and that whether it is intelligent (programmed) or intelligent (committee) is immaterial. The challenge is ultimately to provide a plasma control concept capable of optimally exploiting the installation. Intelligent control, using developed Artificial Intelligence techniques or conventional control techniques will take a long pulse from simplified feedback control which can guarantee basic performance to the control required to guarantee optimal performance.

On the " against " side, a steady state burning tokamak will have to have nuclear licensing to start operation. The question arises whether Artificial Intelligence techniques could be successfully implemented into the plant description for licensing. Fission reactors are presently in the same position, with considerable research being carried out [20]. The present status is that obtaining licensing for AI techniques is very time consuming for use in nuclear reactor operation. An example is the permission from the Belgian Nuclear Safety Authority for one year to have Fuzzy Control tested on-line at the BR1 reactor at Mol during steady state operation [21]. For our purposes, we can simply note that implementing this type of advanced control is indeed a technical issue, but that there are precedents for satisfying the regulatory authorities.

Following this excursion into Artificial Intelligence in control, we should remember that a steady state tokamak will retain an ability to shut down its operation at any time its integrity is threatened, aborting the programmed operation and allowing time to reflect on "what happened". This feature has to be taken advantage of in the final balance in the choice of advanced control techniques. The integrity of the plant will take precedence.

12. IMPLEMENTATION OF STEADY STATE CONTROL

Almost all the time scales of burning plasmas will be longer than on present experiments, resulting in feedback control loops which have bandwidths much lower than at present. The technical issues of implementing feedback control will become simpler. The handling of the large volume of data in steady state will require attention, as will the choice of the architecture of the data systems for data acquisition, supervisory control and plasma control. One effect of the transition to very long pulses will be the disappearance of the normally structural boundaries between these data systems. On the other side of the balance, performing analysis and diagnostic and control consistency checks in real-time will require more computing power, which can be distributed since the tasks themselves are separable. Implementation of this model for data handling is well advanced on new tokamaks [22,23].

As an exercise, the ITER Plant System Integration Report on the plasma control system [24] was examined to see to what extent the issues of steady state would need to be re-considered to reformulate the control system requirements of a steady state ITER-class device. A brief examination of the PSIR suggests that steady state control would require no major design or philosophy changes to the implementation already proposed and remarkably few changes to detail either. The development of a "flight simulator" capable of emulating the full system is consistent with the requirement that a steady state tokamak should have a limited number of duty cycles [25].

13. DISCUSSION

In this paper we have visited a few issues to see the implications for steady state operation. There are few issues associated with the purely technical control side, apart from the requirement of providing adaptive control to guarantee reliable steady state operation, given the time variation of the plasma. It is considered likely that advanced control techniques already being developed for operating tokamaks will play a significant role in the optimisation of very long plasma discharges. The superconducting coil cooling and the first wall equilibrium present perhaps the major issues. The biggest challenge to steady state operation for these two areas remains the question of disturbance modelling for control design and a knowledge of the disturbances to fix the design limitations to avoid exceeding the disturbance budget during steady state.

Otherwise, most of the control issues are intimately linked to the physics issues which will determine the stability and controllability of the parameters in steady state.

An important point is the extrapolation between demonstrated control of a single parameter and the simultaneous control of several parameters which may all exhibit bifurcations. At what point such control becomes stochastic and impossible to rely on is worthy of more attention [26].
On the positive side, techniques which will be required for adapting the equilibrium and kinetic control during a long pulse are already being developed on existing tokamaks and are showing the required ability to derive adequate models for control purposes. Work on understanding the actuator-parameter coupling has been carried out for ITER and designing the diagnostics for burning ITER plasmas is also well underway [27].
It is a suitable end note for the questions of steady state control to remember that disruptions are inconsistent with a steady state tokamak. Although operational regimes exist where the plasma disruptivity is acceptably low for a 500-1000 second ITER pulse, obtaining a 100-fold decrease in disruptivity in a high performance operation region will be a severe challenge to tokamak control and more work on disruption avoidance will be essential to achieve this goal [28].

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ITER PLASMA CONTROL MATRIX

● = **Major direct** effect

○ = **Secondary** effect

	Control Action (Controllable Parameter or System)	Scenario and Magnetics						Fuelling and Exhaust							Aux. Heating and Current Drive						Shutdw
	Measurable Quantity or Attribute to be Controlled	TF field	PF currents	PF voltage	Prefill pressure	Startup EC	Error field comp. current	DT fuelling (gas into SOL)	DT fuelling (shallow pellet)	DT fuelling (deep pellet)	Impurity fuelling	DT divertor fuelling (gas)	Impurity divertor fuelling	Pumping speed	NBI power	ICH power	ECH power	FWCD (power, location)	ECCD (power location)	LHCD (power, n_par)	Shutdown pellet
1: Scenario	Plasma current, q_edge	●		●																	
2: Magnetics	Plasma shape (R, a, κ, δ)			●																	
	Plasma shape (FW gaps)			●																	
	IC coupling impedance			●				○			○	○								○	
	Plasma current initiation		●	●	●	●															
	Locked Mode susceptibilty		○				●								●						
3a: **Core** Performance	Plasma density							●	●	●	●	●	○	○	○						
	Fusion power							●	●	●	○	●	○	○	○	○	○	○			
	He fraction										○		○	●	○	○	○	○	○	○	
	Core D/T ratio							●	●	●	●										
	Core impurity fraction										●		○								
	Core radiation fraction										○		○	●	○	○	○	○	○	○	
	Core plasma rotation (f_rot)														●						
	W_th or β_N (at given P_fus)	●						○	○	○					○	○	○	○	○	○	
	Axial safety factor q(0)														○			○	○	○	
	Current profile j(r)	●													○			●	●	●	
	Sawtooth period	○													○			○	○	○	
3b: **Edge**	ELM period, magnitude			○				●	○		●										
	n_edge							●	○		○			○						○	
	SOL flow							●	○		○			●							
	SOL radiation fraction										●		○								
3c: **Divertor**	Divertor power input							○	○	○	●	○	●	○	○	○	○	○	○	○	
	In-divertor radiation (x,y)										○		●	○							
	Target plasma (n,T)							●	○	○	●	○	○	○							
	Target power or temperature			○				●	○	○	●	○	●	●	○	○	○	○	○	○	
	Divertor neutral pressure			○				○	○	○	●		●	●							
	Divertor He fraction							○	○	○	●	○	●	○							
4: Shutdown	Fast P_fus and I_p shutdown																				●

Figure 1 Sketch of the influence matrix between actuators and parameters to be controlled

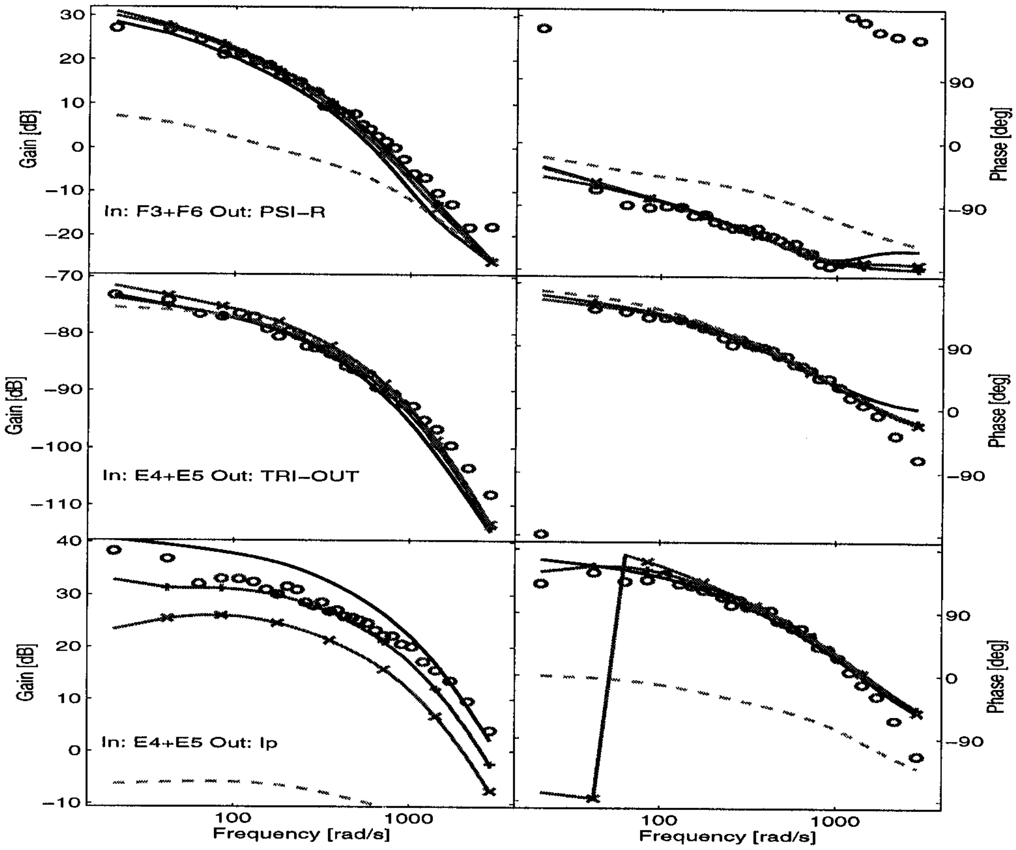

Figure 2 Measured (o) and modelled (lines) transfer functions between the PF coil voltages and plasma parameters

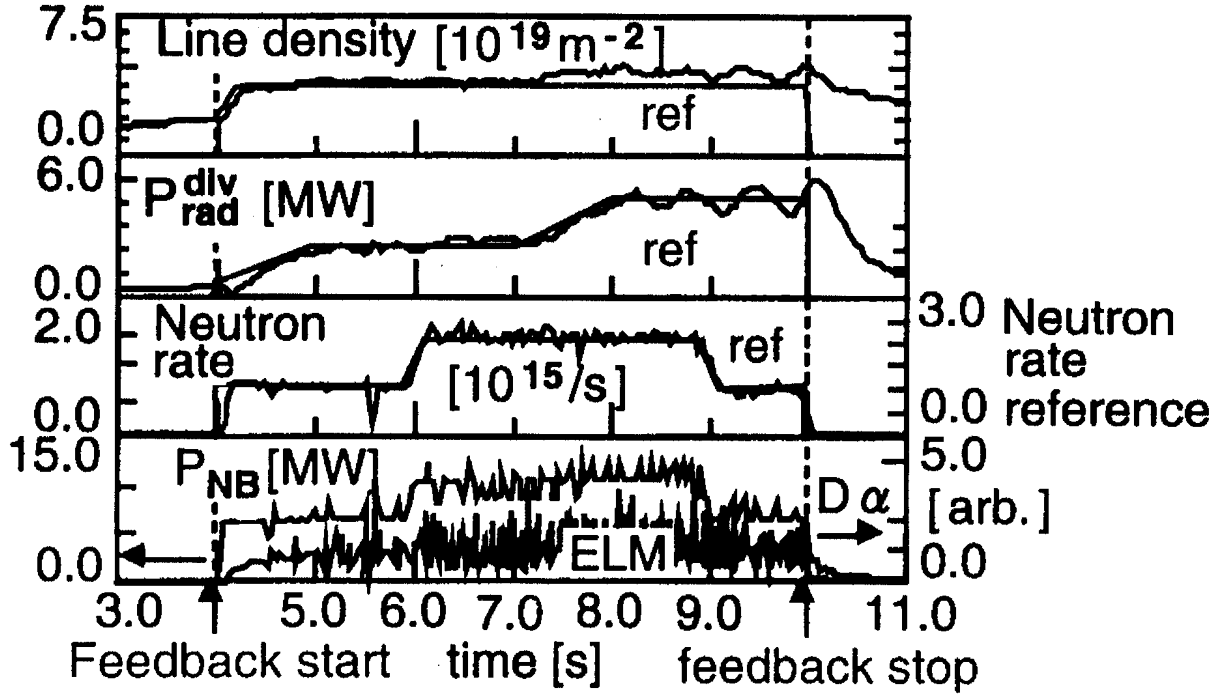

Figure 3 Demonstration of the simultaneous feedback control of 3 kinetic parameters on JT-60U tokamak: line density, radiated power fraction in the divertor region and the neutron production rate. The feedback of the Neutral Beam injection power is shown chopped according to the feedback control signal.

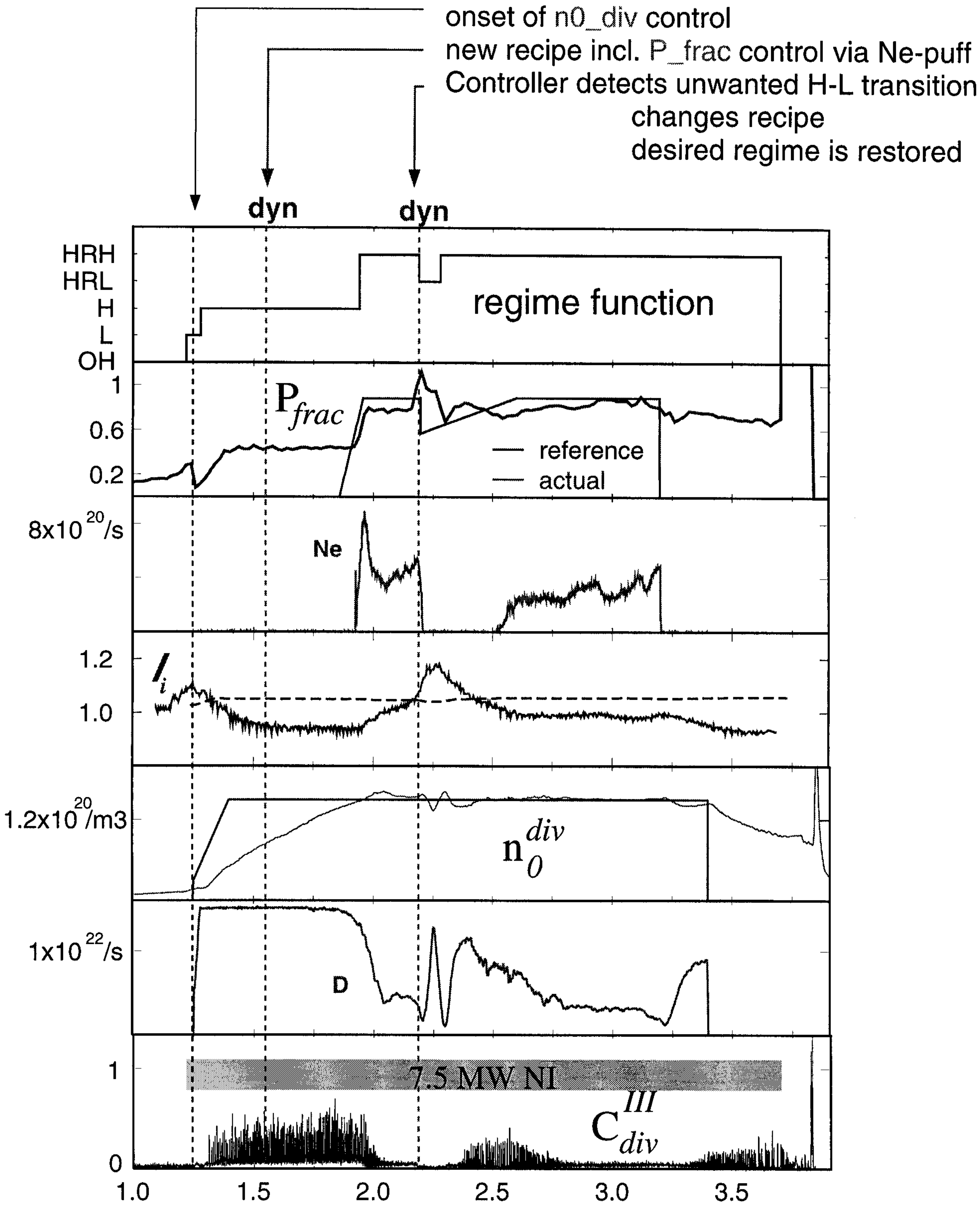

Figure 4 Plasma regime classification as a function of time. The controller switches the Pfrac radiated power fraction demand signal to force the desired mode to be recovered. The reference returns to its original value once the H-mode is recovered