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ABSTRACT

Recent experiments in the DIII-D tokamak [J.L. Luxon, Nucl. Fusion **42**, 614 (2002)] have demonstrated high β with good confinement quality under stationary conditions. Two classes of stationary discharges are observed — low q_{95} discharges with sawteeth and higher q_{95} without sawteeth. The discharges are deemed stationary when the plasma conditions are maintained for times greater than the current profile relaxation time. In both cases the normalized fusion performance ($\beta_N H_{89P} / q_{95}^2$) reaches or exceeds the value of this parameter projected for $Q_{\text{fus}} = 10$ in the International Thermonuclear Experimental Reactor (ITER) design [R. Aymar, *et al.*, Plasma Phys. Control. Fusion **44**, 519 (2002)]. The presence of sawteeth reduces the maximum achievable normalized β , while confinement quality (confinement time relative to scalings) is largely independent of q_{95} . Even with the reduced β limit, the normalized fusion performance maximizes at the lowest q_{95} . Projections to burning plasma conditions are discussed, including the methodology of the projection and the key physics issues which still require investigation.

I. INTRODUCTION

Discharges with normalized fusion performance well in excess of the International Thermonuclear Experimental Reactor (ITER) [1] and Fusion Ignition Research Experiment (FIRE) [2] baseline designs have been obtained under stationary conditions in the DIII-D tokamak [DIII-D]. The criteria by which this assessment is made are discussed later in this section. Several physics issues have motivated these experiments. Design of proposed future experiments is typically based on analysis of multi-machine databases. It is important to validate the design choices made on this basis on present-day machines. Further, it is important to validate the performance under stationary conditions with respect to relaxation time scales of the plasma. For example, recent experiments in the Tore Supra tokamak [3] show spontaneous oscillations in the plasma temperature in discharges operated under feedback control [4]. It is important to examine whether this is a generic feature of the underlying physics of the tokamak system. Finally, it is an interesting scientific question to determine what is the maximum possible fusion performance of a tokamak with the minimal required feedback control, e.g., without any active stabilization of plasma instabilities. The experiments reported here do not represent a definitive answer to these questions but in many respects determine the state of the art of such investigations.

One additional practical motivation for these experiments is the desire in the ITER program to carry out limited nuclear testing of some components. The objective would be to maximize the neutron fluence. Obviously, the maximum fluence would be obtained if ITER could be operated continuously, even at significantly reduced fusion power and gain. Practical issues in the design and operation, such as the electrical and fuel costs of continuous operation and the high capital cost of a heat sink capable of continuous low gain operation, have led to the concept of "hybrid" operation. This pulsed operational scheme would maximize the fluence per pulse and would be limited by the installed heat sink and the available inductive flux. Discharges discussed here project to fluence per pulse approaching $10^{-4} \text{ MW} \cdot \text{s} / \text{m}^2$, assuming heat sink limits of 3000 s.

The performance claims for these experiments are based, in part, on dimensionless figures of merit for fusion performance and duration. With respect to duration, a necessary condition for steady-state operation is that no inductive flux is required to sustain the plasma current. The key figure of merit to assess this is the fraction of noninductive current $f_{\text{NI}} \equiv I_{\text{NI}}/I$ where I is the total plasma current and I_{NI} is current sustained noninductively. The discharges discussed here typically have $f_{\text{NI}} \sim 0.5$ and therefore are not consistent with steady state operation. However, they become stationary in the sense that the plasma parameters such as density, pressure, and current density are

not changing in time. The longest time scale in these plasmas is set by the equilibration of the current density profile. The figure of merit for reaching stationary conditions is taken to be the diffusive time for the lowest radial moment of the current profile at constant current $\tau_R \equiv \mu_0 \langle \sigma \rangle A / \pi k_{11}^2$, where $\langle \sigma \rangle$ is the average conductivity over the cross-section of area A and k_{11} is the first zero of the Bessel function J_1 [5]. For the discharges discussed here, $\tau_R \sim 1-2$ s. The validity of this figure of merit has been verified by monitoring the current profile evolution directly using motional Stark effect (MSE) [6] spectroscopy [7].

For evaluating fusion performance, the dimensionless parameter $G \equiv \beta_N H_{89P} / q_{95}^2$ is a convenient quantity. This quantity is related to fusion gain as follows. For deuterium plasmas, the fusion gain $Q_{\text{fus}} \equiv P_{\text{fus}} / P_{\text{IN}}$ is proportional to $nT\tau$ [8]. In a burning plasma, the situation is more complicated due to the self-heating by the α particles: $Q_{\text{fus}} \equiv P_{\text{fus}} / (P_{\text{Loss}} - P_{\alpha})$ where P_{Loss} is the power transported out of the plasma. Rewriting $P_{\text{fus}} = 5 P_{\alpha}$, the formula for Q_{fus} can be rearranged to give $P_{\alpha} / P_{\text{Loss}} = Q_{\text{fus}} / (Q_{\text{fus}} + 5)$ which is also proportional to $nT\tau$. Increases in the quantity $nT\tau$ are limited by the pressure limit and by the confinement quality. The pressure limit is described by the quantity $\beta_N \equiv \beta / (I/aB) \propto nT / (IB/a)$. The confinement quality can be described by various scaling laws [9,10,11]. All of the scalings are nearly linear in I , so $\tau = HI(\tau_{\text{scal}}/I)$. Combining these,

$$nT\tau = (\beta_N) (H I) f(P, B, R, a, \dots) \quad (1)$$

The limits on B are technological, but limits on I for a given B are described by the safety factor $q_{95} \propto B/I$. So finally a formula related to Q_{fus} is obtained:

$$\frac{P_{\alpha}}{P_{\text{Loss}}} = \frac{Q_{\text{fus}}}{Q_{\text{fus}} + 5} \propto \frac{\beta_N H}{q_{95}^2} \equiv G \quad (2)$$

The power scaling of the confinement is not explicitly factored out in this formula. The loss power is implicit in setting a target value for G based on a specific design, and, like B , it has technological limits based on wall loading rather than a physics constraint. For comparison purposes, the ITER-89P scaling will be used here. Two reasons motivate this choice. It is a global energy confinement scaling and can be evaluated directly from the stored energy without correction for fast particle content (which is <20% in all cases discussed here). It also has a more realistic β scaling [23,24] than the IPB98y2 scaling [10]. As will be evident, the high performance discharges discussed here push toward the β limits. Using a scaling which has pessimistic β scaling would inflate the confinement multipliers obtained in the evaluation.

The remainder of the paper is organized along the following lines. A description of the various stationary high performance discharges encountered on DIII-D along with the key operational details are presented in Section II. Section III discusses the impact of the

magnetohydrodynamic (MHD) instabilities on limiting the current profile peaking, the stability limits and the confinement quality. Projections of specific discharges to ITER and FIRE design parameters are presented in Section IV, including the methodology of the projection. Conclusions based on the data presented are given in Section V.

II. TYPES OF STATIONARY DISCHARGES

The peaking of the current density profile in tokamak discharges is limited, in most cases, by some MHD instability before resistive equilibrium is reached. Given the strong electron temperature dependence of the plasma conductivity ($\propto T_e^{-3/2}$), the resultant current profile for a constant potential across the entire plasma would have $q(0) \ll 1$ except for cases with very high q_{95} or very large off-axis noninductive currents. The type of MHD instability which limits the current peaking provides a convenient way to classify the various stationary discharges obtained in DIII-D. The most common limiting case is that with a repetitive sawtooth instability [14,15] in the plasma center. Other instabilities which are observed to limit the peaking of the current profile are fishbones [16,17] and tearing modes [18,19]. In this section, high performance discharges from the DIII-D tokamak limited by each of these instabilities will be presented, followed by discussion of the operational recipe to establish these stationary discharges. Comparison with ongoing research in other tokamaks will also be mentioned.

A. Stationary discharges limited by tearing modes

Discharges with stationary pressure profiles for durations up to $3 \tau_R$ in the DIII-D tokamak have been reported previously [7,20,21]. The current profiles reach stationary conditions with $q_{95} > 4$ and $q_{\min} \geq 1$ without sawteeth. The key element appears to be the presence of an $m = 3/n = 2$ tearing mode located near the half minor radius. Recently, techniques have been developed to raise β in the discharges. As shown in Fig. 1, a discharge with $q_{95} = 4.4$ can be operated for $\sim 1 \tau_R$ with $\beta_N = 3.2$, which coincides with $4 \ell_i$, which is a simple estimate of the no-wall $n = 1$ β limit [22]. Attempts to raise β further eventually lead to an $m = 2/n = 1$ tearing mode which severely degrades confinement and which can lock, usually leading to disruption. This β limit will be discussed more extensively in Sec. III. In the example shown in Fig. 1, the presence of an $n = 3$ tearing mode (likely $m = 4$ based on the rotation frequency) is sufficient to avoid sawteeth at lower β . As β is raised starting at 2400 ms, the $m = 3/n = 2$ mode appears. There are magnetic fluctuations with < 1 G amplitude at the wall which is resolved as $n = 1$, but they have none of the typical time-dependent characteristics of fishbones or sawteeth. The rotation frequency indicates that the source of the fluctuations must be located near the plasma center; therefore, it may be a helical deformation of the axis or a small saturated $m = 1/n = 1$ island. Reconstructions of the magnetic equilibrium using

Fig. 1. Various plasma parameters versus time for a DIII-D discharge with $q_{95} = 4.4$, $B = 1.7$ T. (a) Plasma current $\times 10$ (MA) (red), neutral beam power (MW) (gray), neutral beam power time-averaged over 200 ms (magenta). (b) Internal inductance $\times 4$ (green), normalized β (red), ratio of global energy confinement to the ITER-89P scaling (blue). The dashed lines indicate the levels anticipated in the ITER baseline scenario. (c) Normalized fusion performance (see text for definition). (d) Amplitude of magnetic fluctuations at the vacuum vessel (G) for $n = 1$ (blue), $n = 2$ (red), and $n = 3$ (green). (e) Intensity of D_{α} emission in the upper outer divertor leg (10^{17} photons/cm²/s). (f) Line-averaged density (10^{21} /m³) (red), Z_{eff} at the mid-radius from active charge exchange spectroscopy of carbon (green), gas flow of deuterium/100 (torr $\cdot$ ℓ /s).

MSE measurements [23] routinely give q_{min} in the range 1.0–1.15 with the long-time average around 1.07 for the example shown in Fig. 1. This value of q_{min} would not be consistent with a saturated $m = 1/n = 1$ island.

The fusion performance of this discharge is quite remarkable, considering the high value of q_{95} . As seen in Fig. 1(c), the parameter G is within 10% of the value estimated for the ITER baseline scenario ($q_{95} = 3.0$, $Q_{\text{fus}} = 10$). As will be seen in Sec. IV, translation of this discharge to ITER parameters could satisfy the basic mission goals at current substantially lower than the design value. The ability to run at high β_N and high confinement quality at this current are somewhat surprising, especially in the presence of tearing modes. The discharge duration is limited in this particular case by the limit of energy which can be supplied by one of the neutral beams, and not by loss of stability or confinement.

B. Discharges limited by fishbones

Discharges with lower q_{95} and without sawteeth for $>0.5 \tau_R$ were reported previously [24,25]. These discharges are characterized by fishbone instabilities which limit the current penetration. This may be due to transport of the fast ions from the center [26] which in DIII-D also carry current, or due to reconnection associated directly with the fishbones [27]. The discharge illustrated in Fig. 2 from DIII-D has $q_{95} = 3.6$. This type of behavior has also been observed in this regime on the ASDEX-Upgrade tokamak [28]. While interesting for present-day devices, it is not obvious how such a regime could be reproduced under burning plasma conditions, due to substantial differences in the relative fast ion population and in its velocity distribution. Therefore, this type of discharge will not be discussed further here.

Fig. 2. Various plasma parameters versus time for a DIII-D discharge with $q_{95} = 3.7$, $B = 2.1$ T. (a) Plasma current $\times 10$ (MA) (red), neutral beam power (MW) (gray), neutral beam power time-averaged over 200 ms (magenta). (b) Internal inductance $\times 4$ (green), normalized β (red), ratio of global energy confinement to the ITER-89P scaling (blue). The dashed lines indicate the levels anticipated in the ITER baseline scenario. (c) Normalized fusion performance (see text for definition). (d) Amplitude of dB/dt (T/s) from analog combinations magnetic signals for n odd (blue) and n even (red). (e) Intensity of D_α emission in the upper outer divertor leg (10^{14} photons/cm²/s). (f) Line-averaged density (10^{20} /m³) (red), gas flow of deuterium/100 (torr $\cdot$ l/s) (magenta).

C. Higher performance discharges limited by sawteeth

Using similar techniques to those developed to make discharges limited by tearing modes or fishbones, discharges at lower q_{95} have been obtained which operate for $>3 \tau_R$ at high β_N with high confinement, even in the presence of sawteeth. The discharge shown in Fig.3 operates at $q_{95} = 3.2$ with $\beta_N = 2.7$ and $H_{89P} = 2.3$ for >4 s. Attempts to operate with higher β_N result in destabilization of an $m = 2/n = 1$ tearing mode, as in the discharges at higher q_{95} where the current peaking is limited by an $m = 3/n = 2$ tearing mode. The limit, however, does not coincide with the estimate of the $n = 1$ no-wall limit ($4\ell_i$) as previously discussed. The fusion performance in this discharge [Fig. 3(c)] is 40%-50% higher than that required for $Q_{fus} = 10$ operation at ITER parameters. This level of performance is maintained in the presence of an $m = 3/n = 2$ tearing mode throughout the duration of the discharge. The high performance phase of the discharge ends only because of scheduled termination of the neutral beam power and plasma current.

Fig. 3. Various plasma parameters versus time for a DIII-D discharge with $q_{95} = 3.2$, $B = 1.24$ T. (a) Plasma current $\times 10$ (MA) (red), neutral beam power (MW) (gray), neutral beam power time-averaged over 200 ms (magenta). (b) Internal inductance $\times 4$ (green), normalized β (red), ratio of global energy confinement to the ITER-89P scaling (blue). The dashed lines indicate the levels anticipated in the ITER baseline scenario. (c) Normalized fusion performance (see text for definition). (d) Amplitude of magnetic fluctuations at the vacuum vessel (G) for $n = 1$ (blue), $n = 2$ (red), and $n = 3$ (green). (e) Intensity of D_α emission in the upper outer divertor leg (10^{17} photons/cm²/s). (f) Line-averaged density (10^{21} /m³) (red), Z_{eff} at the mid-radius from active charge exchange spectroscopy of carbon (green), gas flow of deuterium/100 (torr $\cdot$ ℓ /s).

Fig. 4. Time history of (a) dB/dt (T/s) from a magnetic loop on the vessel, (b) D_α intensity (10^{15} photons/cm²/s) from the upper outer divertor strike point for a discharge similar to that shown in Fig. 1.

D. Key operational elements

Since the first observation of stationary discharges without sawteeth in 2000, this type of discharge has been reproduced and extended through three subsequent run campaigns and many individual operational days. Given normal variability in equipment and wall conditions, successful reproduction of such discharges allows a determination of the essential operational elements of this regime.

There are several key elements in the formation phase of the discharge. The plasma is maintained in L mode throughout the current ramp. Neutral beam heating is applied early to heat the plasma and slow the penetration of current into center. The level of heating is moderate to prevent either substantial reversal of the q profile or formation of an internal ion energy transport barrier. It is crucial to minimize the particle inventory of the walls at this time to prevent a large increase in density at the L-H transition [29]. In the DIII-D tokamak, this is done by biasing the double-null plasma upward magnetically. This ensures that the density scrape-off layer (SOL) is attached to the upper divertor strike points which are maintained by feedback control at pumping apertures located in the top of the vacuum vessel. At the end of the current ramp, a hot, low-density L-mode edge plasma is obtained with $q_{\min} > 1$.

At the onset of the plasma current flat-top, a L-H transition is induced by transiently shifting the plasma to a true double-null. Since the ∇B drift is downward, this reduces the L-H transition threshold power and results in a prompt transition to H mode. The plasma shape is returned quickly to an upward biased state in order to gain density control as quickly as possible. As the plasma stored energy rises, the feedback system is activated to control the neutral beam power to maintain a constant stored energy. The initial regulation level is critical to the scenario. The β must rise high enough to trigger a tearing mode with $n > 1$ (preferably $n = 2$, but in some cases like Fig. 1 an $n = 3$ is sufficient), but not trigger an $n = 1$ mode. Typical values for DIII-D are in the range

$\beta_N = 2.5\text{--}3.0$. It is also essential that this $n > 1$ tearing mode establish itself before the onset of sawteeth [Fig. 3(d)]. For cases which never develop sawteeth, it is important to wait for ~ 1 s before attempting to raise β above this initial level. After time for the current profile to relax, β_N can be raised reproducibly to near the $n = 1$ no-wall limit. For discharges with sawteeth, a relaxation period does not affect the obtainable β . Feedback control of the density is helpful in all cases to maintain a stationary discharge. Notice that in Figs. 1 and 3, gas is required to maintain the line-averaged density. Particle balance calculations indicate that the wall is not playing a role in the long-time evolution of the discharge [20,21]. Note that the discharge in Fig. 2 has neither stored energy nor density feedback control. As the pump reduces the density [Fig. 2(f)], the discharge makes a transition from fishbone limited to $n = 2$ tearing mode limited [Fig. 2(d)].

F. Comparison with other tokamaks

Similar performance discharges have been obtained in the ASDEX-Upgrade tokamak [28]. Regimes with the current peaking limited by either tearing modes or fishbones reach high β_N ($\beta_N > 3$) with good confinement ($H_{89P} > 2$). Discharges without sawteeth and with an $m = 3/n = 2$ tearing mode at $\beta_N = 2.7$ have been reported from the JT-60U tokamak [30]. The International Tokamak Physics Activity (ITPA) topical groups on steady-state operation and on transport and internal transport barriers have requested coordinated activities on the four largest divertor machines (ASDEX-Upgrade, DIII-D, JET, and JT-60U) to map the domains where these discharges exist, particularly in q_{95} and density. Recently the JET tokamak successfully replicated a discharge from ASDEX-Upgrade in which the current peaking was limited by an $m = 3/n = 2$ tearing mode [31]. The discharge ran for ~ 4 s at $\beta_N = 2.8$ and $H_{89P} = 2.1$ with $q_{95} = 3.9$. It is beyond the scope of this paper to compare results among the four machines, but it is evident that the discharges reported here from DIII-D represent a generic class of operations for tokamaks.

III. IMPACT OF THE TEARING MODES AND SAWTEETH

As discussed above, the maintenance of high fusion performance in the presence of MHD instabilities is somewhat surprising. In addition, the possibility of a stationary current profile in the absence of sawteeth was not anticipated. In this section, the influence of the tearing modes and sawteeth on the current profile, global stability, and confinement will be discussed.

A. Impact on the current profile

The effect of sawteeth on the current profile is well known. The instability periodically redistributes the central current density outward. Typically, the approximation of conservation of helical flux is applied [15]. The central current peaks until the instability is triggered by a yet undetermined mechanism. (See [32] for a model which has been used to describe the instability onset.) The net effect is an oscillation of the central q in the neighborhood of $q = 1$, with corresponding oscillations in the central pressure. At low q_{95} , the mixing radius over which the current and pressure are affected can be quite large, perhaps greater than half of the minor radius. Since the instability transforms poloidal flux into toroidal flux or thermal energy, the effective resistance of the plasma is increased somewhat [33]. The sawteeth observed in the discharge shown in Fig. 3 are not known to have an unique behavior beyond the standard effects described above.

More surprising is the discharge in a stationary state with $q_{\min} > 1$ with a modest $m = 3/n = 2$ tearing mode. Analysis of the poloidal flux evolution [34] indicates the electric potential is fully relaxed on either side of the tearing mode, but at slightly different values [20]. This would imply the tearing mode itself adds a voltage source. It has been noted that a truly stationary island should not generate a voltage. However, as shown in Figs. 1(d) and 3(d), the claim of stationarity does not preclude fluctuations in the plasma parameters; it only implies the long-time average behavior shows no evolution.

As a first step to understanding the effect of the tearing mode on the current profile, the change in the conductivity and the Ohmic current density due to the island can be estimated. Starting from the unperturbed electron temperature profile, the conductivity is calculated without an island and then with successively larger islands at the $q = 3/2$ location. The conductivity is assumed to be perfectly flat across the island. Applying the measured surface voltage to these conductivity profiles, the change in the Ohmic current density profile is computed, assuming constant potential across the plasma. If the

broadening of the conductivity profile due to the island is the cause of the current profile broadening, the change in the Ohmic current must reduce the total central current density by $\sim 10\%$, i.e., $q(0)$ must rise from 0.95 to 1.05. Taking the central current density from a magnetic reconstruction including the MSE data, it is found that the conductivity must be flattened over ~ 10 cm for a typical case. Since conductivity is not flattened axisymmetrically, a better estimate would be to increase the required island width by $\sqrt{2}$ to ~ 14 cm. This is about twice the measured island width (typically 6–7 cm). Therefore, the modification of the conductivity profile has some broadening effect on the current profile, but alone it appears insufficient to explain the absence of sawteeth. A more precise modeling of the total current density profile is not possible, due to uncertainties in the bootstrap current, neutral beam current, and the effect of edge localized modes (ELMs) on the edge current. None of the models for these phenomena have been experimentally validated at the level of the radial dependence. In the case of ELMs, no model exists at present.

Assuming the poloidal flux evolution measurements which find an axisymmetric voltage source due to the tearing mode are correct, this voltage could be essential to the stationary state without sawteeth. To generate such a voltage, the island amplitude would need to fluctuate asymmetrically in time to yield a time-averaged voltage. As noted previously, the magnetic fluctuation amplitude [Fig. 1(d)] is strongly modulated throughout the discharge. Careful examination of the magnetic signals indicates the fluctuation amplitude (and therefore the island width) is strongly modulated by each ELM (Fig. 4). Typically, the amplitude of the $n = 2$ fluctuation drops by a factor of 2 in < 1 ms, then recovers and saturates in 5–10 ms. This modulation is not consistent with a gross motion of the plasma at the ELM. The modulation size is consistent with a 30% reduction in the island width, assuming the width is proportional to the square root of amplitude of the magnetic fluctuation. The recovery time scale is consistent with measurements of the edge pressure fluctuation due to the ELM [35]. Since the edge pressure gradient dominates the current generation at the edge (through the bootstrap current), it is not clear whether the time scale is governed by the pressure or current equilibration time scale.

The saturated amplitude of the tearing mode at high β is governed by a balance between the classical tearing stability parameter Δ' and the local pressure gradient. Since there is no evidence that the ELMs modulate the local pressure gradient at the island location, the present working hypothesis is that loss of edge pressure or current at the ELM modulates Δ' and therefore the island width. Calculations with model equilibria based on experimental reconstructions will be done to test this hypothesis.

B. Impact on stability

The presence of sawteeth is correlated with a reduction in the achievable β_N , as shown in Fig. 5. Discharges with $q_{95} \leq 4.0$ have sawteeth and are limited to $\beta_N \approx 2.8$. Attempts to operate at slightly higher β_N lead to a large $m = 2/n = 1$ tearing mode which severely degrades confinement. In principle, higher β_N can be achieved even with the $m = 2/n = 1$ tearing mode, but this requires a substantial increase in power. The mode frequently locks to the wall and disrupts the plasma under such conditions. Therefore, the onset of the $m = 2/n = 1$ is considered the effective β_N limit. Higher β_N also can be achieved for durations shorter than τ_R (Fig. 6). The highest β_N discharge in Fig. 6 operates with above $\beta_N \geq 3$ for >1.2 s before the $m = 2/n = 1$ tearing mode appears. This underscores the need to qualify operating scenarios for durations longer than τ_R . In the absence of sawteeth, discharges have been maintained at $\beta_N = 3.2$ for about $1 \tau_R$. These should be extended to longer duration for more confidence the ability to operate at this high β_N .

Fig. 5. Time history of (a) normalized β and (b) $n = 1$ magnetic fluctuation amplitude (G) for three DIII-D discharges — $q_{95} = 4.4$ without sawteeth (green), $q_{95} = 4.0$ with sawteeth (blue), and $q_{95} = 4.0$ with sawteeth and slightly higher β_N (red).

Operation at high β_N in the absence of sawteeth has been shown previously in the DIII-D tokamak in transient discharges [36]. Surprisingly, the discharges with sawteeth are well above the expected $m = 2/n = 1$ limit in the presence of sawteeth. As shown in Fig. 7, the difference is correlated with the presence of the $m = 3/n = 2$ tearing mode before sawteeth begin. In Fig. 7(a), a pair of shots with $q_{95} \sim 3$ is shown. The discharge in black is pushed to high β early in the discharge as described in Sec. II. The discharge in red is operated in the conventional manner, delaying heating well into the current flattop after sawteeth have begun. Both discharges have a $m = 3/n = 2$ tearing mode of ~ 5 G at the wall, but the discharge with later heating develops an $m = 2/n = 1$ tearing mode.

Fig. 6. Time history of (a) normalized β and (b) $n = 1$ magnetic fluctuation amplitude (G) for three DIII-D discharges — stable (green), slightly higher β , but unstable (red), transiently stable to high β (blue).

Fig. 7. (a) Time histories of a pair of discharges with $q_{95} = 3.2$. The upper box shows the normalized β , the middle box shows the $n = 1$ magnetic fluctuations (G) and the lower box shows the $n = 2$ magnetic fluctuations. (b) Time histories of a pair of discharges with $q_{95} > 4$. The traces are the same quantities shown in (a).

Clearly, the presence of the $m = 3/n = 2$ mode is not the essential element. Evidently, the broadening of the current profile, due to the early heating and the bootstrap current associated with the β_N level necessary to trigger an $n > 1$ tearing mode, results in a stationary current profile which is much more stable to the $m = 2/n = 1$ tearing mode. A similar situation is found at $q_{95} > 4$. The discharge shown in black is the same shown in Fig. 1. The discharge shown in red has an $m = 3/n = 2$ mode which is triggered by one of the first few sawteeth. Again, even though the $m = 3/n = 2$ modes are of equal magnitude, the discharge with sawteeth gets an $m = 2/n = 1$ mode at $\beta_N = 2.1$ while the discharge with the early heating runs at $\beta_N = 3.2$ without sawteeth or an $m = 2/n = 1$ tearing mode. The $n > 2$ modes do not affect directly the stability in the same manner in which they affect the current profile, but the correlation of their appearance with

favorable profiles for $m = 2/n = 1$ tearing mode stability provides a benchmark for successfully timing the discharge for stable operation.

C. Impact on confinement

The presence of sawteeth and tearing modes do not appear to affect strongly the confinement quality in these discharges. Figure 8 shows the confinement relative to the ITER89P L-mode scaling and relative to the IPB98y2 H-mode scaling for a scan of q_{95} at fixed density and plasma current. The value of H_{89P} remains around 2.4 or 20% above the typical H-mode enhancement ($H_{89P} = 2$). The enhancement relative to the IPB98y2 scaling is fairly constant around 1.4 with a slight drop at the lowest q_{95} . The fact that H_{89P} is relatively constant while H_{98y2} drops at low q_{95} indicates that the power degradation in IPB98y2 (or perhaps more correctly the β scaling) is too severe for this dataset. This observation is consistent with other experiments on DIII-D [12,13]. As discussed previously [21], the relatively small tearing modes are expected to have only a modest effect on confinement.

Fig. 8. Variation of confinement time and confinement quality with q_{95} . The blue symbols are the thermal confinement time in s (multiplied by 10 to be on the same scale). The red symbols are the ratio of the thermal energy confinement to the IPB98y2 scaling. The green symbols are the ratio of the global energy confinement to the ITER89P scaling. The symbols indicate the range of the quantity shown within an individual discharge, not the uncertainty.

D. Impact on fusion performance

Using G as a measure of the relative fusion performance, the performance maximizes at low q_{95} (Fig. 9). While β_N is lower in the sawtooth cases [Fig. 9(a)] and confinement is independent of q_{95} (Fig. 8), the strong q_{95} dependence of G is still the dominant dependence. Figure 9 also shows two more discharges with duration $> 2 \tau_R$ but not at the β_N limit in the transition region ($q_{95} \sim 4$). It may be possible to operate in this region somewhat above the ITER baseline value of $G = 0.42$.

Fig. 9. Variation of (a) normalized β and (b) normalized fusion performance (G) for discharges with different q_{95} . Discharges with $q_{95} \leq 4$ have sawteeth while those at larger q_{95} do not.

IV. PROJECTIONS TO BURNING PLASMAS

The high β_N and confinement parameters achieved in DIII-D and other tokamaks under stationary conditions warrant projections to burning plasma conditions. As concrete examples of such conditions, the basic parameters of the ITER [1] and FIRE [2] designs are used for illustration. In making such projections many choices must be made. The methodology employed here assumes conditions similar to those achieved in DIII-D can be obtained in future devices and should not be construed as a prediction of an integrated operational scenario for either device. The focus is on the core plasma fusion performance and hopefully will serve to motivate a more complete assessment of this type of operation.

The basic philosophy is to rely as much as possible on the DIII-D experimental data. In this light, the projection maintains the DIII-D poloidal cross-section shape and keeps q_{95} fixed when moving from the DIII-D aspect ratio to ITER or FIRE. The projection assumes the device's nominal size and toroidal field which then determines the plasma current. It is assumed that the DIII-D β_N can be achieved under these conditions. The loss power is determined by applying the achieved DIII-D confinement multiplier for three typical scaling laws — ITER-89P [9], IPB98y2 [10], and a pure gyroBohm, electrostatic scaling derived from the ITER confinement database [11]. The bremsstrahlung losses are also computed self-consistently. The pressure is divided into density and temperature by choosing a density relative to the Greenwald limit [37], assuming equal electron and ion temperatures. Each design's rule for Z_{eff} is adopted. The ITER design assumes 2% beryllium and 1.2% carbon [1] while FIRE assumes 3% beryllium [2]. The helium ash from the fusion reactions is calculated self-consistently, assuming $\tau_{\text{He}}^*/\tau_E = 5$. The hydrogenic ions are assumed to be equal parts deuterium and tritium. It is important to note that the DIII-D discharges are not on a ρ_* scaling path to either design. The experimental collisionality would yield densities in excess of the assumed density limits. Therefore, the maximum density relative to these limits from the designs are used. The calculation is fully 1D using the measured electron density and temperature profiles from the DIII-D discharges. The electron profiles are used rather than the ion or averaged profiles since burning plasmas will have dominant electron heating and electron heat transport is expected to dominate the heat flux. Finally, auxiliary power is input thermally, so that no enhancement of the fusion power above thermonuclear is assumed. Multiplication on the blanket is also neglected.

A. Projection of the $q_{95} > 4$ case

One motivation of the higher q_{95} cases, as discussed above is to maximize the neutron fluence. (FIRE does not have a nuclear testing mission so only ITER is discussed in this section.) To estimate the duration possible in ITER, the inductive and resistive flux consumed on the ramp-up phases is calculated assuming an Ejima coefficient of 0.6 and is then subtracted from the total available (275 Wb). The flux consumption during the flat-top phase is estimated by calculating the plasma resistance using neoclassical conductivity [38]. Table I lists some of the 0-D parameters including the fusion performance for each of the three scalings.

TABLE I
Projection of higher q_{95} discharge to ITER

$q_{95} = 4.4$, $\beta_N = 3.2$, $n/n_G = 0.85$, $\min v_{*,I} = 0.017$, $B = 5.3$ T, $I = 10.3$ MA, $\alpha_n = 1.38$, $\alpha_T = 1.93$, flat-top time = 5500 s = 1.5 hours

	H	P_{fus} (MW)	P_{aux} (MW)	Q_{fus}
ITER89P	2.2	550	183.	3.0
IPB98y2	1.58	530	114.	4.7
Pure gB	1.61	470	23.	20.

For the standard ITER H-mode scaling, the projection indicates that 530 MW of fusion power could be generated using for pulses in excess of 1.5 hours. This provides a fluence of $\sim 1.0 \times 10^{-4}$ MW $\cdot$ a/m² for each pulse. If the power is limited to the initial installed auxiliary power of 73 MW, then $\beta_N = 2.8$ can be achieved (assuming the IPB98y2 scaling) for 4700 s pulses yielding 460 MW of fusion power, or about 75% of the fluence of the full power case.

Remarkably, if the pure gyroBohm scaling projection were realized, ITER could achieve its baseline mission of 500 MW fusion power with $Q_{fus} > 10$ at 70 % of the design current for the baseline scenario. This would substantially reduce the risk of damage in the event of disruption. However, the trade-off is in a corresponding increase in the difficulties faced in handling the heat load in the divertor due to the reduction in the density. A serious accounting of the risks and benefits is warranted by the robustness of this scenario in large tokamaks worldwide.

B. Projection of the $q_{95} < 4$ case

As shown in Fig. 9, the fusion gain parameter maximizes at the lowest q_{95} or highest current for a given scenario. At present, the DIII-D data only extends down to $q_{95} = 3.2$. The baseline of both ITER and FIRE call for $q_{95} = 3.0$. However, as shown in Table II the DIII-D stationary high performance scenario with sawteeth projects to performance

well above the present design for ITER, even for the ITER89P scaling. FIRE, shown in Table III, depends more on the particulars of the confinement scaling, but the additional current capability and shaping of the FIRE design are adequate to cover the uncertainty.

TABLE II
Projection of low q_{95} discharge to ITER

$q_{95} = 3.2$, $\beta_N = 2.8$, $n/n_G = 0.85$, $\min v_{*,I} = 0.032$, $B = 5.3$ T,
 $I = 13.9$ MA, $\alpha_n = 1.25$, $\alpha_T = 1.92$, flat-top time = 2300 s > 30 min

	H	P_{fus} (MW)	P_{aux} (MW)	Q_{fus}
ITER89P	2.4	780	60.	12.9
IPB98y2	1.47	740	18.5	39.
Pure gB	1.25	700	0	∞
	(1.63)*			

*Number in parentheses is the value of H for DIII-D case. The confinement multiples must be reduced in the ignition case to obtain energy balance

TABLE III
Projection of low q_{95} discharge to FIRE

$q_{95}=3.2$, $\beta_N=2.8$, $n/n_G=0.7$, $\min v_{*,i}=0.071$, $B=10$ T, $I=6.3$ MA,
 $\alpha_n=1.25$, $\alpha_T=1.92$, 20 s flat-top needs 30 V-s

	H	P_{fus} (MW)	P_{aux} (MW)	Q_{fus}
ITER89P	2.4	270	53.	5.0
IPB98y2	1.47	260	40.	6.4
Pure gB	1.63	240	3.9	57.

The projection to ITER indicates that fusion power ≥ 700 MW could be realized for >30 minutes. This illustrates that it is primarily the reduction in the ramp-up flux consumption which increases the fluence in the higher q_{95} case (Table I). The additional fusion power from the high current operation is offset somewhat by the lower β_N limit in the low q_{95} case, but the main effect on fluence is the availability in the flat-top of ~ 60 Wb due to the reduced plasma current in the higher q_{95} case. The enhancement of the flux consumption due to the sawteeth has been neglected in this calculation.

For ITER's baseline mission of >500 MW fusion power with $Q_{fus} > 10$, this scenario robustly satisfies these objectives. Even with the pessimistic ITER89P confinement scaling, Q_{fus} projects to >10. The possibility of ignition is strongly supported by these projections. Using the pure gyroBohm scaling, the confinement multiplier must be reduced by >20% to obtain energy balance. Applying the standard IPB98y2 confinement

scaling, $Q_{\text{fus}} \sim 40$, and using full design current of 15 MA, ignition is marginally obtained.

For FIRE the projections are not as optimistic, but still favorable. The difference can be attributed to the fact that the change of ρ_* from DIII-D to FIRE is smaller than that to ITER, so there is a smaller relative increase in confinement. In addition, the projections are at constant β_N while fusion power increases as β^2 . FIRE has a significantly smaller normalized current at fixed q_{95} because of its higher aspect ratio, leading to a smaller β compared to ITER of DIII-D. The FIRE design compensates for this by using more poloidal cross-section shaping than the DIII-D discharges discussed here. Projecting with the full current (7.7 MA) and shaping of the FIRE design leads to 500 MW fusion power and $Q_{\text{fus}} \sim 25$, using the IPB98y2 scaling and ignition with 20% margin using the pure gyroBohm scaling. These projections underscore the importance of current to fusion power and the nonlinearity of Q_{fus} as the dominant α -heating regime is approached. Another key feature of the FIRE design from Table III is that high field tokamak can achieve the high densities favorable for divertor power handling without challenging the density limit.

C. Critical issues for extrapolation

Three principal concerns must be addressed in order to gain confidence in the extrapolation of the DIII-D discharges here to future devices — the physics of the approach to a stationary state current profile, the confinement projections, and the divertor heat load issues. In each of these areas, there are preliminary indications that the extrapolations will hold for future devices, but more work is needed.

As discussed above, the physics of the current evolution is not well understood. The cases with a stationary state due to fishbones do not appear to have an analogue in a burning plasma environment. The extrapolation of the sawtooth-limited plasmas depends strongly on understanding the role of fast ions in the sawtooth dynamics. If the sawtooth effects on triggering an $m = 2/n = 1$ tearing mode are modified strongly in DIII-D by the presence of the neutral beam ions then it is difficult to see how this lower q_{95} scenario could be maintained in ITER or FIRE. However, the ratio of fast ions stored energy to thermal stored energy in ITER is higher than DIII-D, with obvious differences in the velocity space distribution. The fast ion effects need to be modeled in both cases with close comparison to experiment. If the reduction in the triggering effects of sawteeth is due to high β_p or to a flat shear near $q = 1$ (also related to β_p through the bootstrap current) then the current evolution aspects of the scenario should translate directly to future machines. For the tearing mode limited scenario at higher q_{95} , the two key issues are the saturated mode width and the mechanism of voltage generation at the island. As discussed above, both of these effects may be linked through the Rutherford equation [19], which should describe the tearing physics in future devices. If the ELMs are the

mechanism by which the voltage is generated, then there is a substantial uncertainty in predicting what will happen in future machines. It is important to remember that this uncertainty is also attached to the standard baseline scenario, primarily in the confinement projection. Of the three classes of high performance discharges discussed in Sec. II, the tearing mode limited scenario would appear to have the least uncertainty in the projection of the current profile evolution to future burning plasmas. The attainment of this scenario in JET recently helps establish this assertion [31].

For confinement projection, the two key issues are the impact of T_i/T_e and toroidal rotation on the confinement multipliers derived from DIII-D. The DIII-D discharges have $T_i/T_e > 1$ in all cases shown here, with largest values occurring at higher q_{95} . Theoretical and experimental observations both indicate that $T_i/T_e > 1$ should have a favorable effect on confinement. Scans of density [21] indicate a fairly weak dependence of τ_{th} on T_i/T_e . Similar confinement quality has been observed in high density discharges in the ASDEX-Upgrade tokamak [28]. Experiments with electron heating by ICRH or ECH are planned on JET, ASDEX-Upgrade, and DIII-D which should address this issue. The DIII-D discharges are also heated by neutral beam injection which imparts a toroidal rotation to the plasma. Shear in this rotation is also expected on theoretical and experimental grounds to have a favorable impact on confinement. Calculations with and without rotation using the GLF23 model [39] indicate that the effect is modest; however, the best agreement with experiment is obtained with rotation shear included. The experiments with electron heating discussed above and balanced injection experiments on JT-60U would help shed light on the importance of this effect.

Finally, the impact on the divertor of the low and high q_{95} cases is arguably different. In the low q_{95} case, the stored energy is higher. If the pedestal energy is proportionately higher, then the energy available in each ELM is also increased. Since this is already challenging for future devices it may not be practical to operate at the higher accessible β_N . For the higher q_{95} , the prospects are somewhat different. The absolute density is certainly lower for ITER due to the dependence of the density limit on plasma current. However, the connection length and flux expansion are larger which should help disperse the heat load. In addition, the ELM energy loss may be somewhat smaller at higher q_{95} . This must be investigated further. For the FIRE design, the reduced density limit is of little consequence since the absolute densities are already high.

V. CONCLUSIONS

The discharges presented in this paper challenge some of the basic premises of tokamak operation and projected performance. It is widely assumed that the presence of low n tearing modes is inconsistent with high performance operation. While this is clearly true for the $m = 2/n = 1$ tearing mode, excellent normalized performance is obtained in the DIII-D tokamak and other tokamaks in the presence of $n = 2$ or $n = 3$ tearing modes. Indeed, their presence in part facilitates the stationary current profiles conducive to the high performance. Further, it is widely assumed that β_N must be limited to fairly modest values in the presence of sawteeth. As shown here, this premise is true for conventional operating scenarios where the auxiliary heating is not applied until after the onset of sawteeth. However, as shown in Figs. 3 and 7, it is possible to initiate the discharge differently and obtain profiles which are stable to the $m = 2/n = 1$ tearing mode up to 50% higher β . This would translate to more than doubling the fusion power. Finally, due largely to the previously mentioned beliefs it is assumed that a tokamak must operate at $q_{95} \sim 3$ to obtain reasonable fusion power and gain. The discharges discussed here do indicate the fusion gain does maximize at low q_{95} ; however, performance approaching that expected for ITER baseline mission has been demonstrated in stationary discharges with $q_{95} > 4$.

Projections to ITER and FIRE of these scenarios developed on DIII-D are positive. For ITER, performance similar to the discharge shown in Fig. 3 would give at least 700 MW fusion power with $Q_{\text{fus}} > 10$ even with the pessimistic ITER89P L-mode scaling. The Ohmic solenoid in the present ITER design could sustain this for > 30 min, longer than the present heat sink specification in the design can handle. For FIRE at full parameters, the same discharge projects to $Q_{\text{fus}} > 10$ with the standard scaling and possible ignition if the scaling is truly gyroBohm. Projection of the discharge in Fig. 1 to ITER would yield 500 MW fusion power for > 1.5 hours, assuming an appropriate heat sink. In the event of confinement scaling like the pure gyro-Bohm scaling, this could be maintained with $Q_{\text{fus}} > 10$. This would be an attractive candidate for pulsed net electric operation of ITER at a demonstration scale if suitable blankets and balance-of-plant equipment were available at the ITER site. It would be wrong to conclude that these discharges should drive a redesign of the ITER or FIRE core. Rather they represent strong indications that these designs have significant potential to meet and exceed their performance goals by taking advantage of continuing progress in present-day devices.

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