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Surface heat flux feedback controlled impurity seeding experiments with Alcator C-Mod's high-Z vertical target plate divertor: performance, limitations and implications for fusion power reactors

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Abstract. The Alcator C-Mod team has recently developed a feedback system to measure and control surface heat flux in real-time. The system uses real-time measurements of surface heat flux from surface thermocouples and a pulse-width modulated piezo valve to inject low-Z impurities (typically N_2) into the private flux region. It has been used in C-Mod to mitigate peak surface heat fluxes >40 MW/m^2 down to <10 MW/m^2 while maintaining excellent core confinement, H_{98}>1. While the system works quite well under relatively steady conditions, use of it during transients has revealed important limitations on feedback control of impurity seeding in conventional vertical target plate divertors. In some cases, the system is unable to avoid plasma reattachment to the divertor plate or the formation of a confinement-damaging x-point MARFE. This is due to the small operational window for mitigated heat flux in the parameters of incident plasma heat flux, plasma density, and impurity density as well as the relatively slow response of the impurity gas injection system compared to plasma transients. Given the severe consequences for failure of such a system to operate reliably in a reactor, there is substantial risk that the conventional vertical target plate divertor will not provide an adequately controllable system in reactor-class devices. These considerations motivate the need to develop passively stable, highly compliant divertor configurations and experimental facilities that can test such possible solutions.

1. Introduction

Control of the boundary plasma in tokamak fusion devices remains one of the great challenges to economic fusion energy production. Although we presently have no first-principles model for precisely quantifying the challenge, a multi-machine empirical scaling law indicates that the boundary heat flux width scales inversely with the poloidal magnetic field and is independent of machine size [1]. This scaling suggests that essentially all reactor- and burning plasma-class devices will have unmitigated parallel heat flux densities in the boundary plasma exceeding 10 GW/m^2 [2]. Surface heat flux engineering limits for steady state power exhaust are three orders of magnitude less than this at ~10 MW/m^2. Present tokamaks do not provide such high heat flux plasmas allowing us to study this incredible challenge, to mitigate reactor-level parallel heat fluxes. However, Alcator C-Mod comes close with ~1–2 GW/m^2 unmitigated parallel heat flux densities at the divertor plate [3].

The present state of the art in boundary heat flux solutions is the conventional vertical target plate divertor with extrinsic, low-Z impurity injection. The highly oblique magnetic field line incidence angle (down to ~1°, although ITER will be ~3–4° due to engineering challenges [4]) on the vertical target plate can reduce the heat flux density to the divertor surface by a factor of ~50. But the additional factor of ~20 can only be attained with the help of volumetric dissipative processes, such as low-Z impurity line radiation and plasma-neutral interactions. If successfully controlled in this way, the plasma can become 'partially-detached', i.e., attain significant plasma pressure loss and heat flux density reduction near the strike point region while further out into the SOL the plasma pressure and heat flux densities are relatively unchanged.

Full detachment is actually desirable for mitigating divertor heat flux (and may be required to eliminating sputtering erosion in a reactor) but it has been found to be problematic for maintaining good core plasma performance. In this situation, the location of the 'detachment front' – i.e. the region of cold, high-density, radiating plasma – can intrude onto closed flux surfaces near the x-point region forming an 'x-point MARFE' [5,6]. Since this zone is well connected thermally to the pedestal region, radiation in this zone reduces the power flowing across the pedestal where a transport barrier forms – an essential feature of high confinement H-modes. This in turn reduces the pedestal top temperature. As a result, core plasma confinement, which is intrinsically tied to the pedestal top temperature, tends to degrade [6,7] (here quantified as the H-mode confinement factor, $H_{98,y2}$ [8]). Therefore, significant attention is focused on developing divertor detachment *control strategies*, such as active feedback control of low-Z impurity seeding. The idea is to identify a reliable set of measurements (ideally, including divertor surface heat flux), employ fast timescale sensors to monitor them and actuate a fast-valve that injects just the right amount of impurity gas to mitigate the divertor heat flux while maintaining good core plasma confinement.

A number of techniques have been successfully employed. Sensors have ranged from bolometers [5,7,9] and vacuum ultraviolet spectroscopy [10] chords, to tile current shunts [9,11,12], Langmuir probes [13], and surface thermocouples [14]. All of these systems have had varying degrees of success controlling impurity injection under relatively steady conditions and some have done so while demonstrating maintenance of good core confinement.

However, transients, and the ability of such systems to handle them, have not been considered in much detail. As illustrated by experiments discussed in this paper, we have found that feedback-controlled impurity puffing schemes appear to have an important, inherent limitation: they cannot possibly respond fast enough to many of the most important transients (e.g., intrinsic impurity injections, H-L transitions). This is due to the very fast nature of the transients relative to the response time of impurity gas actuation as well as the small operational target window (i.e., the combination of incident plasma heat flux, plasma density, and impurity density) in which a partially-detached plasma can be maintained on a conventional vertical target plate divertor [15,16]. Loss of control and/or heat flux mitigation, even for very short periods of time, has severe consequences. Machine protection systems, such as a massive gas injection system [17], must be

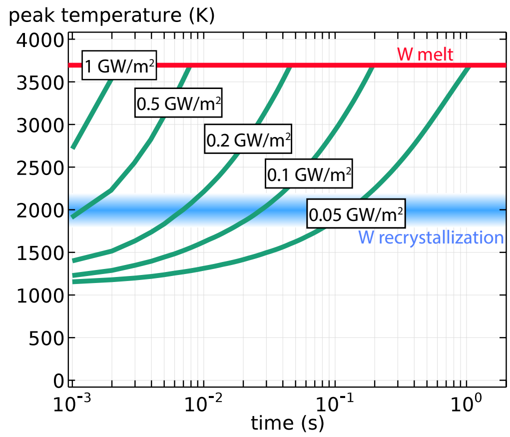

Figure 1 Peak surface temperature from a heat pulse (surface heat flux values indicated in boxes) to an ITER-like tungsten monoblock starting at 10 MW/m² with active cooling. Any reattachment of the plasma must be responded to promptly (<1 s for response to mitigate heat flux) to avoid melting and permanently damaging the surface. Even faster responses are required to remain below the recrystallization temperature.

used for events where the divertor heat flux mitigation system cannot respond fast enough. Although, even massive gas injection may not be fast enough to ensure protection in all scenarios.

For events that cause reattachment of the plasma to a solid metal target, there are both the problems of immediate damage due to melting (which causes both surface deformation and loss of material) as well as long term damage due to erosion. Reattachment of the unmitigated heat flux will cause permanent melt damage to the divertor plate in well under 1 s (Figure 1). The surface will surpass the recrystallization temperature even quicker. Even a small amount of melt damage in high-heat flux regions is likely unacceptable, since it can run-away, accelerating damage and preventing operation with plasma contacting the melt-damaged surface [18]. Repeated reattachment events will severely accelerate erosion rate by transiently increasing both the sheath potential and plasma flux [19]. Both melting and erosion could potentially be mitigated with a renewing liquid metal wall [20]. However, such a solution has its own technological challenges; in this paper, we consider only operation with and concerns for solid metal walls.

Control of plasma confinement must be maintained at all times. Plasma confinement (H_{98}) is such a strong 'control knob' for fusion power gain (Q) that nearly any degradation is unacceptable for a power reactor. Most power plant designs rely on having $H_{98} > 1$. Note that the best plasmas with x-point MARFEs typically have lower H_{98} values ~0.7–0.9 [6]. For ITER, degradation of H_{98} from 1.0 to 0.8 would result in a degradation of Q from 10 to 5 [21]. Additionally, reduction of confinement and/or enhancement in core radiation (seeding-induced) could cause a thermal instability in self-heated, burning plasma regimes – a situation not encountered in present devices. In this scenario, a reduction in core plasma burn and/or increase in core radiation results in decreased power through the pedestal, which in turn reduces core confinement.

This paper examines the performance and the limitations of an impurity feedback control system recently developed for divertor heat flux mitigation and core plasma performance optimization in Alcator C-Mod. It is unique in that it employs direct, real time measurements of surface heat flux and a fast-acting piezo valve. Due to C-Mod's small size, the closed-loop response time of this system is short – certainly much shorter than an equivalent system for a reactor scale device. Thus, the results presented here provide important guidance. A brief overview of Alcator C-Mod and the feedback system are given in Section 2. As discussed in Section 3, the system has demonstrated the ability to automatically mitigate surface heat fluxes in steady L- and H-mode down to preprogrammed levels. Section 4 describes the performance limitations under transient conditions. Section 5 discusses implications for reactor-class devices and outlines some possible paths toward developing robust and controllable systems. Section 6 contains a brief summary and conclusion.

2. The Alcator C-Mod tokamak and the heat flux feedback system

Alcator C-Mod is a compact (R=0.67 m), high-field (B_T<8 T, B_p<1.5 T), high-power (P_{aux}<6 MW) tokamak [22], which allows it to reach boundary heat fluxes parallel to the magnetic field higher than any other present machine; ~1–2 GW/m^2 is commonly attained [3]. The vertical target plate divertor was pioneered on Alcator C-Mod [15] and, due to its adoption as the base-line design for ITER [23,24], has been the major focus for world research on boundary and divertor plasma physics. In recent years a significant amount of research on feed forward injection of impurities into the C-Mod divertor has been done [25–28]. One of the key results of these investigations was the demonstrations of divertor impurity seeding to >90% radiated power fraction with partial detachment while maintaining good core plasma confinement, H_{98}>1.

Although feedback control is highly desirable, its implementation can be difficult. It requires the identification of a suitable set of sensors that, ideally, directly relate to the quantity that is being controlled – in this case, surface heat flux. With this goal in mind, the Alcator Team developed surface thermocouples into a reliable diagnostic for real-time measurements of surface temperature and heat flux [14,29]. The surface thermocouples are simply a coaxial, refractory metal (Mo/W-Re) thermocouple with the thermojunction directly exposed to the divertor plasma. The surface thermocouple face is flush with the divertor surface and grounded to the divertor, ensuring that it received the same surface heat flux as the rest of the divertor and its temperature responds in the same manner as the rest of the molybdenum divertor surface. The sensors are in a special set of ramped tiles to ensure that they are not shadowed. The magnetic field incidence angle is typically ~3°. The surface thermocouples have a very good time response (few ms) due to the small mass of the thermojunction and sufficiently large signal-to-noise to measure surface heat fluxes down to ~1 MW/m^2, depending on the timescale of interest.

The surface heat flux is calculated from the surface thermocouple temperature measurements in two different ways: (1) A digital computation is performed after every plasma pulse and (2) an analog computation is performed in real-time during the plasma pulse using a very simple and accurate 7-node RC-network [14]. The analog computation makes use of the direct relations of voltage and current in electrical diffusion with temperature and heat flux in thermal diffusion; the resistor and capacitor values are scaled to match the thermal diffusivity of the material of interest (here molybdenum). The surface thermocouple based surface heat flux measurements have distinct advantages over IR cameras; there is no need to correct for a loosely-bound surface layers [30] nor the need to re-calibrate due to time-evolving surface emissivities [31].

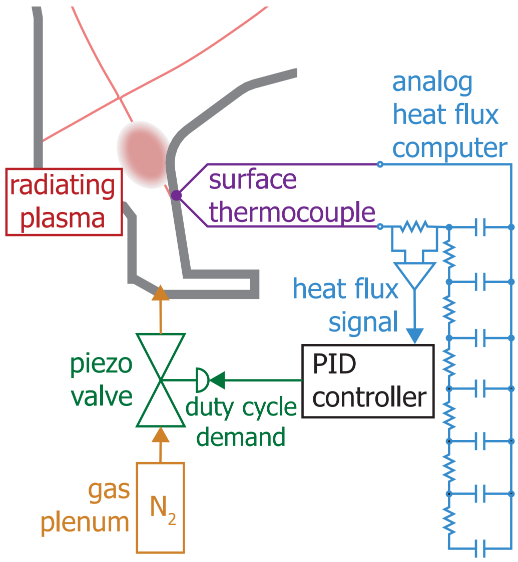

Figure 2 Schematic of the heat flux impurity seeding feedback system.

Real-time surface heat flux signals are sent via analog fiber optic links to the C-Mod digital plasma control system where a PID controller is implemented. It sends out a demand signal to adjust the pulse-width modulated duty cycle (and thus time-averaged injection rate) of a piezo valve, which injects impurity gas into the private flux region of the divertor. A schematic of the integrated system is shown in Figure 2.

Nitrogen and neon are the two gases that have been used with the most success for heat flux mitigation in C-Mod. Nitrogen is found to behave as a partially-recycling gas whereas neon is a fully-recycling gas. From a control system point of view, nitrogen is preferred since it has a short lifetime in the plasma. In contrast, once neon atoms that are introduced into the vacuum vessel, the core neon content stays essentially the same for the remainder of the discharge.

3. Feedback control of impurity seeding in L- and H-mode plasmas

L-mode observations

L-mode plasmas tend to be more amenable to impurity seeding, having a shorter impurity confinement time and lacking an edge pedestal that is sensitive to radiating impurities.

Although L-mode is rarely considered reactor-relevant from a core energy confinement standpoint, it still serves as an excellent test bed for studying the physics of the boundary plasma. We find the thermocouple-based feedback system to be an excellent tool to set and maintain essentially any level of divertor heat flux dissipation, Figure 3, and to observe its effects on other parts of the boundary plasma.

Already, one of the most salient features of the feedback system can be seen in the L-mode tests: the close-loop response time of the system is ~100–200 ms. This is limited by the travel of gas down the seeding tube (~2 m), into the volume behind the divertor module, and through the plasma and vacuum regions around the torus. Given the compact size of C-Mod, these are likely near the fastest obtainable response times; reactor-scale devices will only be slower due to the longer distances traveled. For example, the valve for ITER's gas system is ~20 m away from the plasma. The gas takes 500 ms to travel the length of the tube and another 500 ms before it reaches ~2/3 of the maximum flow rate [32,33].

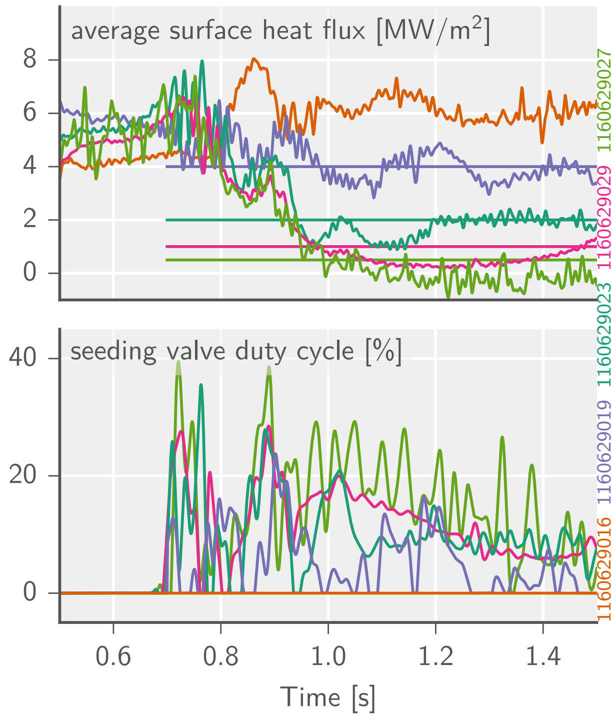

Figure 3 Use of the feedback system to control surface heat flux in L-mode plasmas to various levels of mitigation for otherwise repeated shot conditions. Solid horizontal lines indicate the heat flux demand level.

Tuning of the system was relatively straightforward: A feedforward duty cycle was applied to the system based on the plenum pressure and previous experience. The response of the sensors to this duty cycle was noted and used for a by-hand calculation of the P-gain. This was then applied to the system for the next plasma pulse. What resulted was an unstable oscillation that was only halted by the end of the plasma pulse. The Ziegler-Nichols method along with the oscillation period were used to determine approximate values for the D- and I-gains. A low-pass filter was implemented on the sensor signal to reduce the high-frequency contributions to the D-gain. Further refinement was done on the gains, obtaining satisfactory performance in ~5 shots. Unfortunately, not enough run time was available to determine how sensitive the gain values were to plasma conditions.

Practical considerations for H-mode in a reactor scenario

For H-mode plasmas, the major challenge with regard to divertor seeding is to maintain the temperature pedestal. The temperature pedestal height is closely related to the amount of power flowing through it and, due to stiff core energy transport, core confinement is tightly coupled to the temperature at the top of the pedestal [34,35], so, it is highly desirable to minimize (or ideally decouple if that is even possible) the impact of the divertor impurity seeding on the pedestal. Also, H-mode plasmas typically have a much longer impurity confinement time than L-modes. For C-Mod the core impurity confinement time is the same order or greater than the feedback response time [36]. Therefore, if a seeding puff were to raise core impurities too high, it would take a few core impurity confinement times to recover.

In examining experimental results and projecting current experimental results to reactor regimes, there is an additional complication that should be pointed out. In a fusion reactor, alpha heating, not external heating, dominates the power balance. In this case, a step increase in pedestal or core radiation can lead to a thermal instability: when radiation increases, power through the pedestal decreases, core confinement decreases, fusion power production decreases, power through the pedestal decreases, and so on until there is insufficient power to sustain the pedestal and an H-L back transition occurs. Therefore, an additional level of burn control is needed to counteract this feedback loop, such as varying the external power or D-T mix. In an externally heated plasma, when pedestal or core radiation increases the external heating system (usually) keeps on injecting power. If the radiation is not too large, the plasma is thermally stable and remains in H-mode. As such, present experimental demonstrations of heat flux "solutions" are missing this key feature.

H-mode Observations

Using the feedback system in EDA H-mode plasmas (steady-state, ELM-free [37]), we were able to reproduce the performance that was demonstrated in feed-forward experiments: mitigation of the plasma surface heat flux while maintaining excellent core confinement. In an example case, Figure 4, the peak surface heat flux was mitigated from ~50 MW/m^2 (~1 GW/m^2 parallel to the magnetic field) down to ~10 MW/m^2 with H_{98}~1.05.

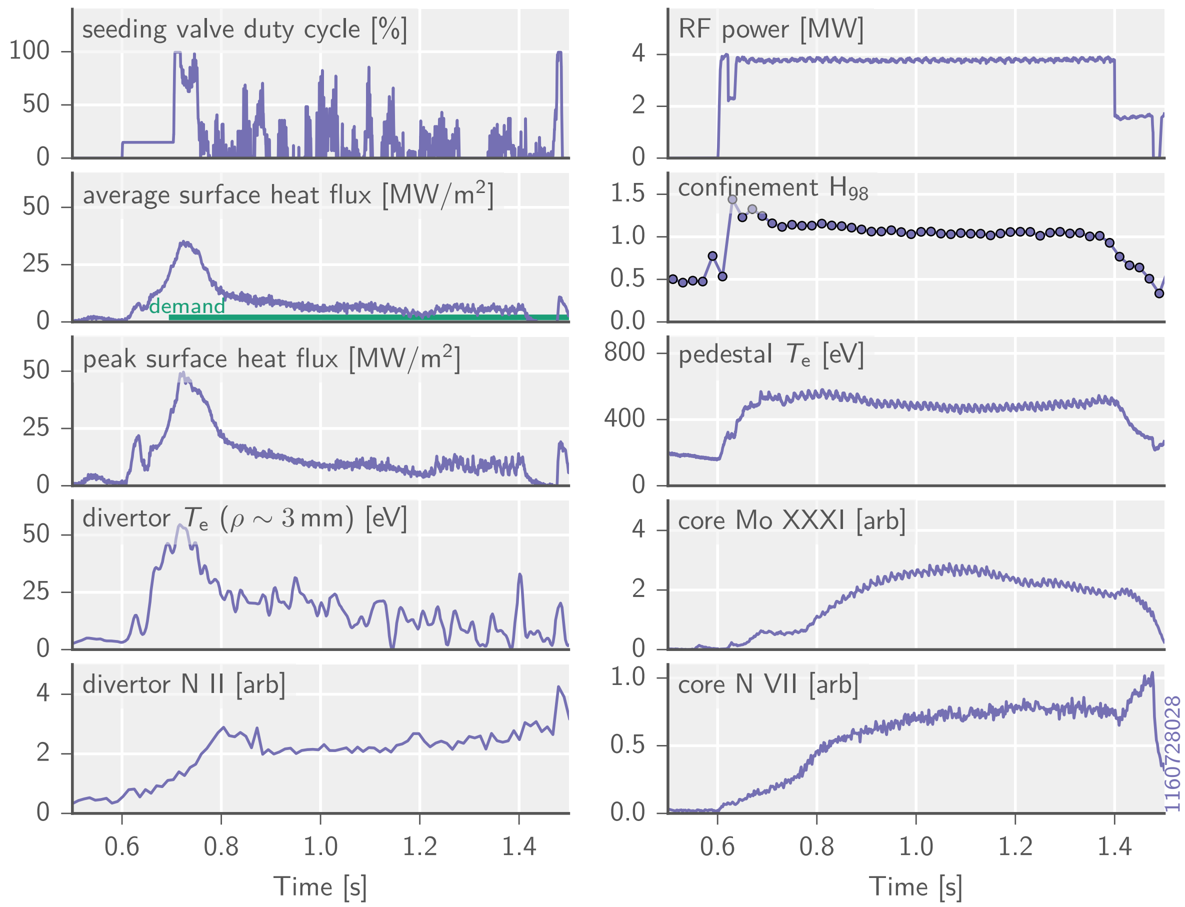

Figure 4 Feedback control of N_2 impurity seeding during an EDA H-mode. Peak surface heat flux was controlled down to <10 MW/m^2 while maintaining H_{98}>1.

In this case, the PID gains were tuned such that there was no over-shoot in seeding, bringing the heat flux down to near the demand level within ~250 ms. The integral gain could have been larger to bring it all the way down to the set-point before the end of the discharge. Limited run time did not allow for this level of fine-tuning. The divertor nitrogen spectroscopy reached steady state within ~100 ms whereas the core nitrogen took ~250 ms. Over the later ~100 ms increase in core nitrogen the edge electron temperature and H_{98} drops slightly.

Through the heat flux mitigation, the divertor electron temperature is significantly reduced, from ~50 eV to <15 eV. Additionally, the core molybdenum content started to decrease ~400 ms after the start of heat flux mitigation. The decrease in core molybdenum appears to result in a recovery of edge electron temperature. Unfortunately, due to the short pulse length, the core molybdenum did not decrease to a steady-state value during heat flux mitigation. Had it continued long enough, confinement may have increased back to its pre-seeding level.

Looking at the profiles across the outer divertor, Figure 5, before and during feedback heat flux mitigation we see that the surface heat flux and electron temperature are significantly reduced across the divertor profile. The plasma pressure is reduced at the strike point but remains roughly the same in the far SOL as is typical of a 'partially-detached' plasma. Chord-integrated divertor molybdenum brightness is reduced at the strike point but remains roughly the same in the far SOL as is typical of a 'partially-detached' plasma. And, of course, the chord-integrated divertor N_2 influx (due to recycling) increases. The peak of nitrogen influx appears to be above the nominal strike point position.

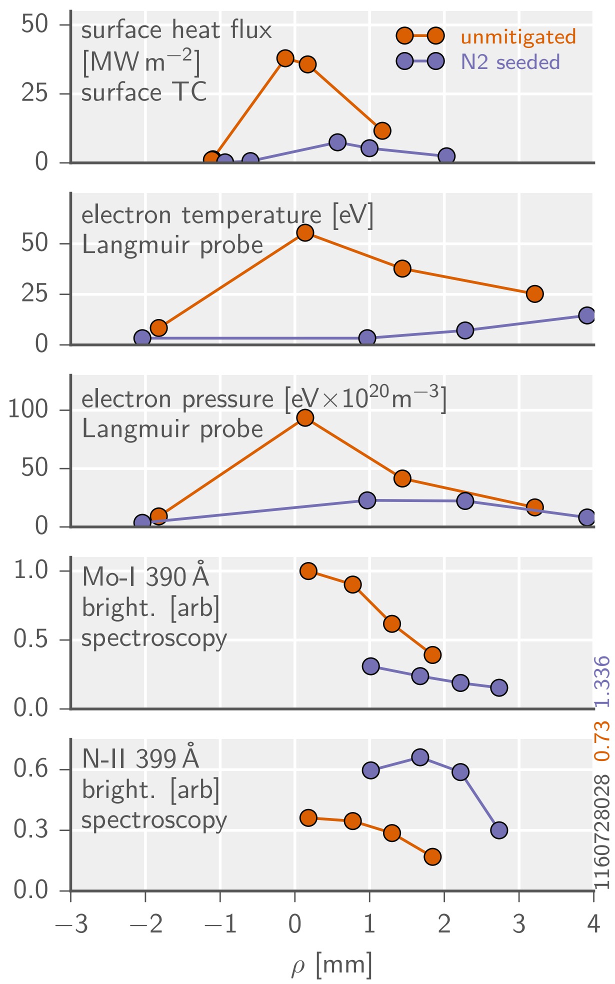

Figure 5 Profiles across the outer divertor before (orange) and during (purple) the feedback heat flux mitigation within a single plasma pulse. Probe measurements are at the divertor plate and flux-surface mapped to the outer mid-plane (ρ). Spectroscopic measurements are line-integrals through the divertor and mapped to the ρ position where they intersect the outer divertor plate. Shifts in measurement point locations are due to slowly evolving magnetic equilibrium.

Using multiple sensors ensures that the system still functions if there are movements of the magnetic equilibrium. As seen in Figure 1, the position of the plasma profile has shifted ~1 mm relative to the sensors. The system still functions even if the peak heat flux is not measured. As long as sensors remain in regions of heat flux above the demand level the system will put in impurities until the heat flux is decreased to the demand. If any area away from the peak is decreased, then the peak is also decrease to at least that level.

These results are encouraging; they are essentially what is needed to obtain a divertor solution for ITER under relatively steady conditions: peak surface heat flux reduced from >40 MW/m^2 to ~10 MW/m^2 while maintaining core confinement, with the additional benefit of reduced divertor target plate sputtering and core high-Z impurity content. However, this 'solution' does not scale to a DEMO-solution, where erosion must also be eliminated.

4. Feedback system performance during transients

Although demonstrating control under relatively steady conditions is an important achievement for a feedback system, operation under transients is the more important test for successful control. The consequences for loss of control in a reactor are severe, as discussed in the introduction. Here we report the system response for three different types of transient events: (1) over-injection of the seed impurity gas; (2) a slowly increasing impurity source from the vessel wall due to melting; and (3) a spontaneous natural impurity injection from the vessel wall.

4.1 Over-injection of seed impurity gas

While heat flux mitigation requires a certain impurity concentration, a feedback system of this type only directly controls the injection rate. A faster response requires a larger injection rate. Yet, this come at an increased risk of ruining core confinement. The lag – due to slow gas transport – makes it very easy to over-shoot heat flux mitigation. Needing to operate very deep into detachment means that almost any overshoot will cause the plasma to completely detach, increasing impurity penetration to the pedestal.

An example of a poorly tuned controller with an unmanageable seeding rate is shown in Figure 6, where the proportional gain of the PID controller was set too high. What results is an interesting case of a rectified (since it cannot go negative), damped oscillation in the control quantity (surface heat flux) along with a 'state transition' between dominant divertor and x-point radiation locations and a recovery back to divertor radiation. There was a large proportional error since the feedback was turned on after the large surface heat flux was established. The large proportional gain forced the valve open at 100% duty cycle (i.e., maximum flow rate) for ~60 ms, over-mitigating the divertor heat flux with a resultant deleterious effect on the core plasma, significantly degrading the pedestal (H_{98}~0.8). The time between reaching the heat flux demand and degrading core confinement was <30 ms. If the core heating was from fusion reactions rather than external RF, this would have surely resulted in a H-L back transition, although the timescale for this to unfold would probably be much longer. This plasma recovers ~250 ms later after the nitrogen transports out of the core.

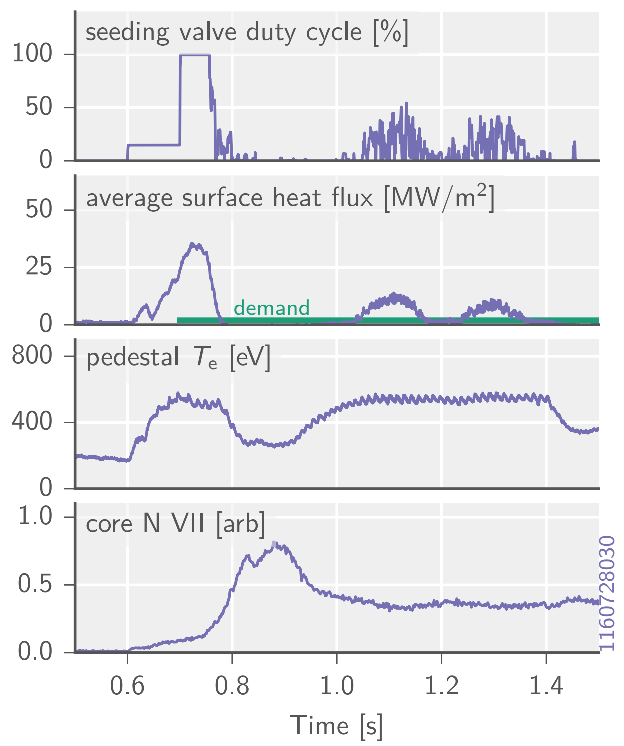

Figure 6 Demonstration of a reduction in core confinement due to over-puffing the seed impurity, forming a radiating x-point MARFE.

With this aggressive injection rate, the divertor heat flux was mitigated within ~40 ms, a 5-fold improvement in response time from cases that do not over-shoot. However, due to rapid rate at which the divertor responds and the slower time for the remaining gas to flow in from the seeding tube, the result is to inject too much N_2 into the divertor. This leads to a limit on the injection rate that is an important restriction on the fastest time response which an impurity gas injection system can operate. Even in the ideal case of gas reservoirs and valves directly adjacent to the plasma, it is likely that a very high injection rate cannot be used. A more complex control model (similar to the 0D model in Ref. [12]) or one that includes the effects of system lag may be able to provide improvements, limiting the overshoot. But the ultimate response of the system is still limited by mass transport of the seeding gas. Determining the maximum injection rate and the physics which controls it is beyond the scope of this work, although it is an important parameter to consider in the specification of feedback control schemes.

4.2 Slow melting source from the wall

A slowly melting source from the wall (Figure 7) presents an interesting challenge to a heat flux feedback control system: The overall response behaves much like an *over-injection* of the seed impurities, exhibiting a slowly decreasing pedestal temperature (where 'slow' is a 50% pedestal temperature reduction in ~100 ms). However, since the core-damaging impurity source is due to part of the wall overheating, it is actually caused by an *under-injection* of the seed impurity. Yet they both result in a decrease of pedestal temperature and surface heat flux.

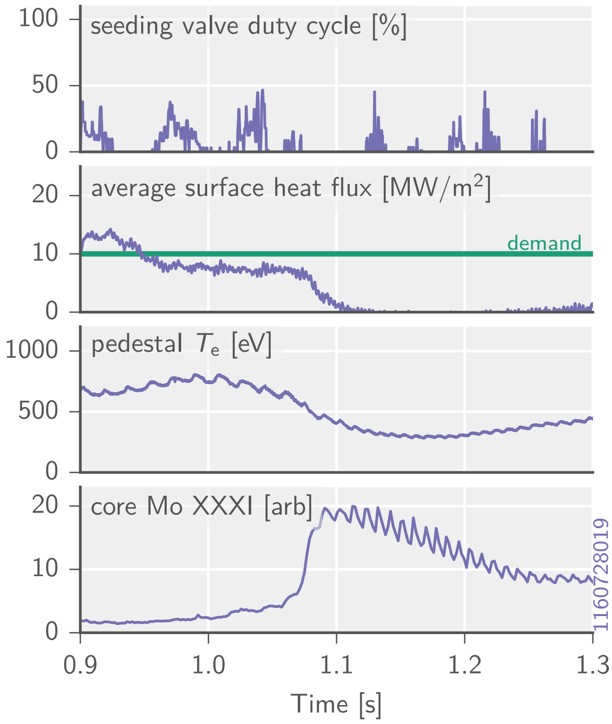

Figure 7 Evolution of parameters during a slow molybdenum melt event.

When first considering a heat flux mitigation feedback system on C-Mod, it was thought that the pedestal electron temperature could be used as an indication of over-seeding. A reduction in pedestal temperature could be an early warning that core confinement was decreasing. But, as illustrated in this case of a melting wall, it is clear that more than two measurements (e.g., surface heat flux and perhaps pedestal electron temperature) are necessary to discern the cause of core degradation events. Information on what impurity species (intrinsic versus extrinsic) is causing the decline in core confinement is necessary to properly react with an increase or decrease in seeding rate. Additionally, the nature of the intrinsic impurity source must be known; the appropriate mitigation response for (1) surface melting due to too much plasma heat flux and (2) sputtering due to an RF source may be different. However, recognizing that the diagnostic set will be extremely limited in the neutron environment of power reactor, obtaining such exquisite knowledge of the impurity injection event may be challenging in practice.

Finally, it is important to examine the time scales and the system response time. The initial, slow rise in core molybdenum and decrease of pedestal temperature occurs over ~50 ms. This is followed by a very rapid rise in core molybdenum (<10 ms) and a continued slow decrease in pedestal temperature. While it may be possible for a finely tuned impurity injection system to track this event on Alcator C-Mod, it is far too fast for an ITER-like system, by about a factor of ~20.

This was just one example of a melt event which happened to occur over the course of these feedback experiments. More generally, the time response of a melt event is dictated by the surface heat flux and geometry local to the melt event (see Figure 1 for an example). The likelihood that an unmanageably fast melt event will occur increases with power density. Scaling this divertor solution to a reactor, the occurrence of melt-events beyond the response capabilities of a feedback system may be unavoidable.

4.3 Rapid 'natural' injection from the wall

Rapid (<10 ms) release of material from the first wall into the plasma and its consequences will be one of the great challenges for a steady state reactor. These are typically cause by a small 'flake' or melted droplet released from the first wall [18].

One such example is shown in Figure 8, where a rapid (~3 ms) injection of molybdenum from the wall kills the pedestal and causes an H-L back transition. A significant fraction of the plasma stored energy is lost to the divertor, increasing the peak surface heat flux from ~1 MW/m^2 to 40 MW/m^2 within ~3 ms. In this case the RF heating remained on and the plasma transitioned back to H-mode after losing ~20% of the stored energy. For a self-heated burning plasma, the effect would undoubtedly be much more severe.

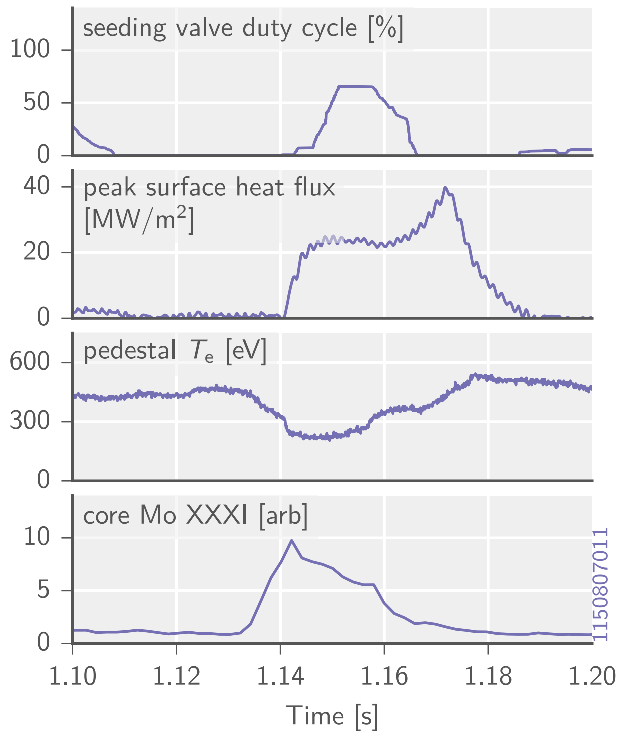

Figure 8 Response of plasma to rapid injection of molybdenum.

One advantage that a reactor has compared to smaller experiments is that the effect on the core plasma should be smaller. The size of a melt droplet injection is set by the wall material properties and is expected to remain similar in size (~100 μm), which is expected to result in ~1 MW of radiated power [18]. An increase of radiation by 1 MW has a much more detrimental impact on the compact C-Mod (~1 m^3 and ~0.2 MJ) than it would on the larger ITER (~800 m^3 ~200 MJ).

5. Challenges going towards a reactor

Although we have demonstrated a feedback controlled system that successfully mitigates divertor heat flux while maintaining core confinement, there are clear challenges going forward towards developing a truly reactor-relevant system. The three main challenges are: 1) a reactor-relevant set of sensors, 2) an integrated system with sufficient time response, and 3) a system that adequately suppresses erosion (in addition to integration with core control as well as machine protection). These issues are considered in the following sections where the integrated system is split into three main components: sensors, actuators, and plasma-system.

5.1 Sensors

For a heat flux mitigation system, there must be an appropriate set of sensors to assess the state of the system. They may either be direct measurements of the quantities of interest, such as surface heat flux in this paper, or less direct measurements related through some model.

Although we have sensors that can measure many useful parameters with sufficient response rates (~1 ms), at present, there are no sensors used for heat flux mitigation qualified for use in a fusion nuclear environment: Electrical measurements – such as thermocouples, Langmuir probes, or tile current shunts – all require a material for electric insulation. In a fusion radiation environment, insulators will degrade as well as suffer from prompt radiation-induced voltages/currents [38]. Optical measurements – such as bolometers, IR cameras, and spectroscopy – all will require a plasma-facing mirror. Baring a mirror surface renewal technique, the intense plasma environment is expected to preclude the use of any short-wavelength (UV, visible, IR) optical technique [39]. Longer-wavelength techniques – such as ECE and reflectometry – may be possible. Reflectometry, in particular, could be useful for measuring the density front associated with a radiation front. However, neither have been used as the sensor for a heat flux mitigation system.

In addition to surface heat flux related diagnostics, spectroscopy may be needed [40]. As described above, spectroscopic measurements will be important to differentiate between intrinsic and extrinsic impurity causes of core degradation. Spectroscopy may also be required for assessing the erosion rate. Looking forward, the community should find the minimal set of measurements needed for control of the boundary plasma and then focus on developing radiation and plasma hardened techniques for these measurements.

5.2 Actuators

A heat flux mitigation system must have a control parameter, some way of affecting the plasma state. There are four main ways to do this: 1) upstream plasma density, 2) impurity concentration in the boundary, 3) power flowing into the boundary, and 4) magnetic equilibrium.

Upstream plasma density

The upstream plasma density is determined by the core plasma scenario and is not a viable option for controlling the boundary plasma independently. However, it is an important parameter for determining the surface heat flux and measurement of it will likely be needed if a more complex control system based on a model of the boundary plasma is used.

Impurity concentration

Control of boundary plasma heat flux mitigation has been done largely through impurity concentration by puffing of extrinsic gas species (e.g., N_2, Ne, Ar). (There are some examples of puffing of hydrogen isotopes, but this is not considered sufficient for reactor conditions.) The system for this is relatively simple, consisting of a gas plenum, valve, and length of pipe. Ideally, the valve would be neutron resistant and located close to the plasma. One of the main reasons for the piezo valve in ITER being located so far from the plasma is to protect it from neutrons. But there do appear to options for neutron-resistant piezo transducers [41]. Additionally, an inertially-actuated valve could be located very near the plasma while shielding the sensitive components from neutrons [42,43]. However, as noted above, even with locating the valve very near the plasma, the time response of such systems remains a serious limitation, being orders of magnitude longer than most transients of practical interest, and there is little room to improve this.

In addition to the impurity injection rate, the impurity pumping rate is also comparatively slow. The pumping system in a reactor (necessary for helium ash exhaust) will, among other things, provide an important means for reducing the concentration of recycling impurities. Calculations for ITER suggest that neon will have a residence time in the divertor of ~60–600 ms, depending on the divertor neutral pressure achieved (~10–1 Pa, respectively) [33]. The 'puff and pump' technique has been demonstrated to decrease the residence time of impurities [44]. However, it comes at the expense of increased mass flow rates through the gas processing systems that must handle tritium.

Power flowing into the boundary

The power flowing into the boundary has some room for control. It is often assumed, due to empirical evidence [45], that the minimum power flowing through the pedestal to the boundary plasma must be above the H-L threshold power. This then sets a maximum on core radiated power and the minimum power into the boundary. For a machine like ITER operating near the H-L threshold, there is not much room for core radiation. More reactor-like, higher-Q machines are likely to operate with fusion alpha powers well above the H-L threshold and thus allow for more core radiation before an H-L transition. However, a core-radiating gas impurity injection system will suffer from the same poor time response as edge-radiating systems. The time response of the core plasma will be much slower than the divertor due to longer impurity confinement timescales.

One could envision an actuator consisting of active addition of power to boundary plasma, much like the use of ECH to control neoclassical tearing modes in the core [46]. This would be to counteract either the addition of too much seed impurity gas or a reduction of power flowing from the core plasma to the boundary. But controlling boundary heat fluxes through a core heating system (including fusion power) would be a poor choice, since response times would be dominated by the core energy confinement time (~1–10 s for reactors). However, RF power systems are available that can deposit significant power at relevantly fast (~1 ms) time scales in the boundary plasma, e.g., LHRF [47].

The amount of reserve power necessary for such a control system is likely to a significant fraction of the power flowing into the boundary plasma (at least a few 10's MW in an ITER-sized device and much more in a GW-class device) to ensure a safe margin for control. Having this large of a power system sitting on standby is an undesirable engineering choice given the large capital cost of equipment, low utilization, and increased recirculating power. This technique also introduces the dilemma of putting more power into a system from which it was already a struggle to handle removing the large amount of power.

Magnetic equilibrium

Movement of the plasma equilibrium may be an option to avoid a melt event. Feedforward strike point sweep programming is a common technique in today's machines to reduce the chance of melting. (It is not a viable steady-state solution for reactors due to the cyclic thermal stress.) But even if such a technique for reducing target temperatures were possible, it is too slow to react for transient heat flux mitigation in an ITER-class device: Changes in magnetic equilibria take ~1 s due to the large inductance of the poloidal field coils and the slow time for fields to sink through the thick, conducting vacuum vessel and blanket [48].

Quite simply, there is presently no viable actuator to improve system response sufficient to keep up with common transients.

5.3 Plasma system

Although the sensors and actuators both have their challenges, it is the plasma portion that is the most limiting part of the whole system. The plasma part sets the time scales and windows of operation for the integrated system. The majority of transient events occur on the order of the sound speed. Given similar plasma conditions, this suggests that the time scales of transients may scale with the linear size of the machine. Since the response of the impurity gas injection system also scales with the linear size, the ratio of time scales remains approximately the same.

There is a very small window in parameters (i.e., power into the boundary, upstream density, and impurity fraction) for operation of a partially detached divertor [15,16]. When the divertor goes beyond partial detachment, it can form a confinement-degrading x-point MARFE [49]. This makes for a challenging system to control – especially since the divertor plasma will need to be held below ~5 eV to minimize divertor target erosion. There may not even be a window for detachment and acceptable erosion rate without a significant reduction in core confinement.

The plasma system may be the greatest opportunity for innovation in developing a controllable system. There have been a variety of 'advanced' magnetic divertor geometries proposed over the last ~10 years [2,50–52]. These concepts move beyond a simple diverted plasma with a conventional vertical target plate to modifying the magnetic geometry. Typically, they contain some combination of an extended leg and additional poloidal field null(s).

One with particular promise for a more robust integrated system is the x-point target divertor [2], which takes advantage of the greatest weakness of present conventional vertical-target plate divertors: the rapidity with which the detachment and radiation fronts move to the divertor x-point. Inserting a second x-point in the plasma in front of the target easily facilitates complete detachment from the target (thus taking care of both the heat flux and erosion problems) [53]. And having a long target leg could lead to a more robust plasma front location control, possibly affording a factor >10 window increase in the operational window for the control variables: power into the SOL, upstream density, and impurity fraction based on magnetic geometry considerations alone [16].

Initial simulations with the 2D plasma-neutral fluid code UEDGE suggest that the power window for obtaining a stable, fully detached divertor condition for a double-null x-point target in the proposed divertor test tokamak ADX [2] spans ~0.5–7.0 MW of power into the boundary [53]. The combined effects of long-leg magnetic geometry, enhanced gas-plasma interactions and the presence of a secondary magnetic x-point are found to contribute to this factor of 10 enhancement compared to conventional divertors – an unprecedented result. It would be interesting to see it the large power window could be passively compliant to ELMs as well.

Such a system potentially makes the slow performance of the gas puffing system acceptable: The nominal position of the radiation front would be maintained ~midway in the leg: Since there is a very large compliance window in maintaining the radiation front in the leg, the leg would be able to handle very large changes in power into the boundary without having the radiation front move to the core plasma or the plasma reattach to the divertor. A transient faster than the gas system would can respond to simply moves the front position within the leg. The position would then be readjusted to the ~midway point by the gas system on its own, slower time scale. Additionally, the state of such a system might be diagnosed by detecting the position of the detachment front with reflectometry [54,55], one of the few reactor-relevant diagnostic systems.

6. Discussion and Conclusions

A feedback system for injection of seed impurities has been used on Alcator C-Mod to mitigate peak divertor heat fluxes from >40 MW/m^2 down to that required by the limits of power exhaust <10 MW/m^2 while maintaining excellent core confinement with H_{98}>1. The major uncertainty in projecting these results to a reactor scenario is the details of impurity transport under reactor conditions. The effective impurity residence time, which determines the radiative effectiveness above coronal equilibrium [56], are not within present experimental capability nor do first-principle methods exist for theoretical projection. Reactor-level pedestals are expected to be much hotter than present devices and divertor radiating impurities may be fully stripped by the time they reach the pedestal. The direction of impurity transport in the pedestal in a reactor is predicted to be different than in current experiments. In particular, transport of tungsten is predicted to be outwards [57]. Such a self-shielding pedestal, if extended to impurity species used for plasma control, would help decouple divertor impurity and core plasma impurity content. However, such fine detail of what the pedestal will be like in reactor-class devices remains beyond the realm of present knowledge.

Since C-Mod is not able to access ELMing regimes with the outer strike point on the vertical target plate, they could not be included in this study. However, interaction of the impurities with ELMs is an important open question, from the standpoints of both the divertor and pedestal. The solution in the divertor must be able to handle the steady-state as well as the ELM heat flux pulses. There's also a complex interplay between the impurities and ELMs in the pedestal [6].

Although the system demonstrated here reaches steady state heat flux density values needed for ITER, it does not address the long-term erosion problem for reactors. To reduce erosion to acceptable levels requires that the net erosion rate to be below a level that allows for surface to last between replacement times. It may be that such low erosion rates requires detachment over the entire divertor (complete detachment as opposed to the partial detachment needed for ITER). There has been success in maintaining an H-mode with a completely detached divertor [6]. However, this has come at the expense of a drop of core confinement. If this is the path we must follow, reactor designs should self-consistently include it, almost certainly at an increase of cost and size to achieve the same power output.

Finally, divertor detachment control system struggles with transients. This is due to the slow response of the integrated system (~150 ms in C-Mod) and the fast nature of transients (~10 ms or less) as well as the relatively small operational window in control variables that keeps the radiation front away from the core plasma with a conventional vertical target plate divertor. It is unlikely that any new actuator can provide such performance in a reactor with much longer actuator delivery times. There are promising 'advanced' divertor concepts that may be able to address both the erosion and controllability issues. However, the fusion community needs a flexible new divertor test tokamak to assess the viability of these concepts before committing them to a reactor.

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