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PhD Thesis

XXXIII Engineering for Energy and Environment

***Studies of divertor magnetic configurations
and control systems for fusion energy
experiments***

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Abstract

The divertor is one of the most essential components in tokamaks and sustaining the excessive heat power strike. The sputtered impurity particles from the damaged divertor surface materials can dilute and contaminate the plasma, degrading the confinement and fusion efficiency. Thus, the divertor overheating has been one of the most challengeable issues for the fusion reactors like ITER.

The divertor heat load is determined by the edge plasma parameters. Using the edge transport code can help us to explore the available way to generate the detached plasma. We used the transport code SOLEDGE2D-EIRENE to simulate the radiative divertor plasma and the quasi-snowflake (QSF) plasma on EAST. SOLEDGE2D-EIRENE is a typical 2-D fluid code of the edge plasma by solving Braginskii equations with the Bohm boundary conditions. Modeling result indicates that the neon seeding increased the radiated power both in the plasma core and the divertor region, and the mitigation of the divertor heat load still can be improved. The QSF magnetic configuration is a special advanced divertor shape on EAST which is similar to snowflake minus(SF-) configuration. Longer poloidal connection length of the scrape-off layer(SOL) of QSF plasma increases the radiated power loss compared with the normal single null(SN) configuration, which is beneficial for reducing the divertor heat load.

The detachment feedback control is also developed on EAST. We firstly used the CD₄ as the working gas to realize the divertor electron temperature ($T_{e,div}$) feedback control. CD₄ shows a different physical mechanism of forming detachment compared to the medium Z impurity like neon(Ne) and argon(Ar). The integrated control method of $T_{e,div}$ - P_{rad} joint feedback was achieved in H-mode plasma: P_{rad} feedback control increases the radiated power, while $T_{e,div}$ monitor judges the divertor detached situation. Through this way, the robustness of the feedback control system is improved, using less impurity seeding amount still can keep the detachment and having less influence on the core confinement.

Key words: divertor; SOL; transport; simulation; feedback control

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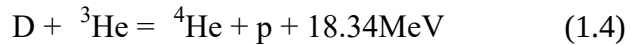
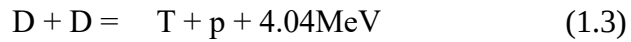
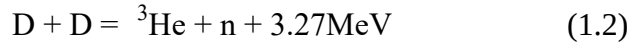
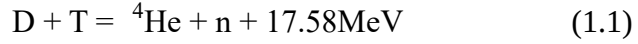
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Chapter 1. Introduction

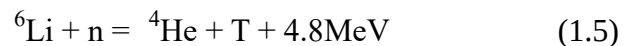
1.1 The fusion ignition condition

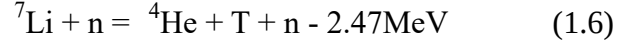
Nuclear energy is the energy released by the reactions of atomic nuclei, and there are two main ways: fission and fusion. The fission reaction is a process in which heavy nucleus splits into light nuclei. The continuous fissions, i.e. the chain reactions, can release a mass of energy, so the fission reactors have already become one of the most important electricity sources. However, the extremely radioactive fuel and products are easily causing the environment contaminations or even catastrophes, and there are a lot of tragedies in history caused by the nuclear leakage.

Contrary to the fission, the fusion reaction is the process which the light nucleus with the small atomic numbers colliding with other light nucleus to produce the heavy nucleus and release the binding energy. Considering the collision cross section, the mainly interested fusion reaction is about the hydrogen isotopes (H, D, T) and the helium isotopes (^3He , ^4He) as following^[1]:



The early fusion reactor works by the deuterium-tritium (D-T) reaction described in equation (1.1). Figure 1.1 is the D-T reaction schematic diagram. Each fusion reaction can release about 17.58MeV of energy, so the estimated energy produced by fusion fuel of 1kg is about $10^8\text{kW}\cdot\text{h}$. The fuel and products of the fusion reaction are not highly radioactive, which is beneficial for protecting the ecological environment. Besides, the fusion fuel like deuterium(D_2) can be acquired from the ocean. Theoretically, there is about 10^{14} tons of deuterium element, which fully meets the human need^[2]. Meanwhile, tritium can be gained from the following reactions:





The thermal neutrons generated from the D-T fusion reaction (equation (1.1)) then react with lithium (Li) and produce more T atoms (equation (1.5) & (1.6)) to join the D-T reactions again, the fusion reactor thus can work in long term with high efficiency.

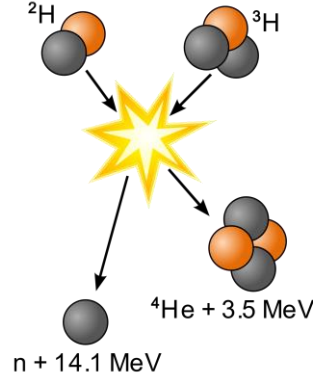


Figure 1.1 Schematic diagram of D-T fusion reaction

The key to develop the controlled fusion is heating and constraining the intensive D-T plasma. The condition of fusion reaction is extremely high temperatures and pressures, the temperature is of the order of 10keV. Under these conditions, deuterium and tritium particles will exist as the plasma: composed by ions and electrons but still in the electrically neutral state, and shows the collective characteristic as the gas^[3]. Based on equation (1.1), the energy of 17.58MeV released by D-T fusion is gained by α particles (i.e., ${}^4\text{He}$ nuclei) of 3.52MeV and neutrons of 14.06MeV. Only the energy of α particle part can be captured to heat the plasma further into the self-sustaining burning state.

The fusion energy density of D-T reaction can be expressed by the following equation:

$$P_{T_n} = n_D n_T \langle \sigma v \rangle \epsilon \quad (1.7)$$

In which, n_D and n_T denote the density of the deuterium and tritium ions separately, $\langle \sigma v \rangle$ denotes the cross section of the fusion reaction, and ϵ denotes the energy released by a single fusion process. If setting the ratio of deuterium and tritium

density is 1:1, the total ion density is n , equation (1.7) can be changed into equation (1.8):

$$P_{T_n} = \frac{1}{4} n^2 \langle \sigma v \rangle \epsilon \quad (1.8)$$

Considering the plasma volume as V and setting ϵ_α as the power density of α particles gained from the D-T reaction, the heating power from α particles to the D-T plasma is:

$$P_\alpha = \frac{1}{4} n^2 \langle \sigma v \rangle \epsilon_\alpha V \quad (1.9)$$

There is a continuous energy loss in the fusion plasma as expressed by equation (1.10):

$$P_{\text{Loss}} = \frac{W}{\tau_E} \quad (1.10)$$

In which τ_E denotes the energy confining time, W denotes the total energy stored in the plasma:

$$W = 3nTV \quad (1.11)$$

In which n and T represent the ion density and temperature separately. Equation (1.10) indicates that due to the finite energy confining time, the fusion plasma energy loss must be compensated by the additional input power:

$$P_{\text{input}} = P_{\text{Loss}} \quad (1.12)$$

Considering the heating from α particles shown in equation (1.9), the power balance in equation (1.12) can be expressed as equation (1.13):

$$P_H + P_\alpha = P_{\text{Loss}} \quad (1.13)$$

Where P_H represents the additional input power. Substituting equations (1.9), (1.10) and (1.11) into (1.13), the expression of P_H is:

$$P_H = \left(\frac{3nT}{\tau_E} - \frac{1}{4} n^2 \langle \sigma v \rangle \epsilon_\alpha \right) V \quad (1.14)$$

The self-sustaining burn of the fusion plasma means that the fusion power offsets the power loss completely, without the additional input power (i.e., $P_H \leq 0$). Thus, the ignition conditions of the fusion plasma, known as the Lawson Criterion, is obtained^[1]:

$$n\tau_E \geq \frac{12}{\langle \sigma v \rangle} \frac{T}{\epsilon_\alpha} \quad (1.15)$$

When the temperature is between 10 to 20 keV, the value of the collision cross

section $\langle\sigma v\rangle$ is about $1.1\times 10^{-24} \text{ T}^2 \text{ m}^3 \text{ s}^{-1}$. Thus, the plasma ignition condition under this condition is:

$$nT\tau_E \geq 3\times 10^{21} \text{ keVsm}^{-3} \quad (1.16)$$

Therefore, the technical aim of the controlled fusion is to make the fusion plasma parameters satisfy equation (1.16), i.e., the Lawson's criterion. Up to now, there is still a large gap between the aim and the experimental results.

1.2 The magnetic confined fusion and tokamak

As mentioned in 1.1, fusion needs the extreme conditions of high temperature and pressure. However, there is no kind of surface material can sustain such a high temperature ($\geq 10\text{keV}$). It is necessary to find other ways to “fill” the fusion plasma. The main available way are the inertial confinement and the magnetic confinement.

The inertial confinement fusion is compressing the fusion fuel into a small space, like the target pellet, and bombarding it by multiple high intensity lasers to generate the high-temperature dense plasma rapidly close to the ignition conditions, which is a transient energy explosion^[4]. On the contrary, the magnetic confined fusion uses the high-intensity magnetic field to constrain the fusion plasma, and the fusion energy is released smoothly for a long time under the quasi-steady state.

The charged particles of the plasma in a strong magnetic field condition will have the Larmor precession and move along the magnetic field lines. In the macro perspective, it can be regarded as the plasma “freezing” on the magnetic field line^[5]. This effect can reduce the displacement perpendicular to the magnetic line. If generating a toroidal-closed magnetic field, the motion of the plasma in the direction parallel to the magnetic field line will be constrained, and that is the basis of the magnetic confined fusion^[1].

There are various magnetic confined fusion devices, such as the tokamak, stellarator^[6], reversed field pinch^[7] and so on. They differ in the plasma confinement performance, the instability, the magnetic field structure and so on. Tokamak has been researched world widely with the longest study history. It was put forward in 1950s, means a toroidal vacuum vessel with multiple magnetic coils. In general, tokamak is

composed by multiple components and subsystems, including the toroidal and magnetic field coils, vacuum chamber, various auxiliary heating systems, control system, various diagnostics and so on.

Up to now, dozens of different tokamaks have been built around the world. The representative devices include JET in the UK, TCV in Switzerland, ASDEX/ASDEX-Upgrade in Germany, TFTR and DIII-D in the US, and JT-60U/JT-60SA in Japan. JET is the tokamak with the largest size. It has executed the D-T fusion experiments and generated a fusion output power of about 1.7MW^[8-9]. TFTR is the first tokamak generating the plasma and has also implemented the D-T fusion experiments^[10]. ASDEX realized the H-mode plasma for the first time in 1982^[11]. Besides, the other tokamaks also contribute a lot on the study of the magnetic controlled fusion.

The International Thermonuclear Experimental Reactor (ITER) project is one of the most important fusion research projects. It is planned to build the world's largest full-superconducting experimental tokamak. The primary goal is to verify the feasibility of self-sustaining burn of the fusion plasma, and to produce plasma with fusion gain coefficient(Q) higher than 5 which can run for a long time stably^[12]. Figure 1.2 is the structure schematic of ITER tokamak. According to the current engineering progress, ITER is expected to discharge for the first time in 2025^[13].

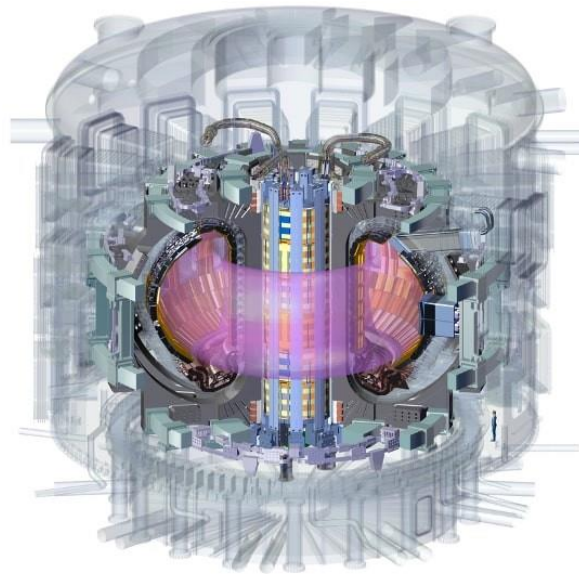


Figure 1.2 The schematic of ITER project

In this thesis, the main work is finished on Experimental Superconducting Advanced Tokamak (EAST). EAST is the first fully-superconducting tokamak with non-circular cross section around the world. EAST holds the world record of the L-mode plasma discharge for 400 seconds^[14], and the world record of the stable H-mode plasma discharge for 100 seconds^[15]. EAST first wall is made by molybdenum (Mo), and owns the tungsten(W) upper divertor with water cooling system, which is same to ITER. In 2021, the lower divertor was also upgraded from graphite(C) to tungsten^[16].

1.3 The plasma-wall interaction challenge on tokamak

Tokamak can be regarded as a container for the fusion plasma, in which the plasma and the tokamak wall materials exchange the energy and particles during the discharge continuously, i.e., the plasma-wall interactions(PWIs) happen. The influence of PWIs on tokamak is bi-directional: On the one hand, the surface material is corroded, which reduces the tokamak lifetime as shown in Figure 1.3; On the other hand, the impurity particles produced by PWIs contaminate the fusion plasma.



Figure 1.3 Damage of EAST target plate material caused by the plasma heat flux

PWIs can be classified into two types: the heat transfer and the particle interaction. The heat transfer mainly includes heat conduction and heat convection. The heat flow from the main plasma arrives at the divertor target plates and the other first-wall surface like the limiter, causing the thermal damages to the material, such as radiation enhanced sublimation and the other damaging processes^[2]. The gas adsorbed on the wall will also enter the recycle of the neutral particles through the thermal desorption.

The charge-exchange process, elastic/inelastic scattering and the chemical reactions occur between the high-energy charged particles and neutral particles near the wall lead to the physical or chemical sputtering. The sputtered impurity particles will degrade the core confinement^[17]: For the high Z impurities like tungsten (W) and copper (Cu), they will lead to the disruption even with very small amount; For the medium and low Z impurities, the long-term accumulation will also dilute the fusion fuel and reduce the fusion reaction efficiency. For example, the heat flux of ITER design on the divertor targets will be as high as $20\sim 40\text{MW/m}^2$ ^[18], while the divertor surface materials can only sustain the peak heat flux of about $5\sim 10\text{MW/m}^2$ ^[19-20]. Thus, the divertor overheating is a significant challenge for the magnetic confined fusion research.

In this issue, the edge plasma transport plays an important role. Most of the PWIs happens in the scrape-off layer (SOL) which connects the plasma core and the divertor targets^[20]. In the second chapter of this thesis, the key contents of SOL will be clarified in detail. By understanding the SOL physical process, we can design the technical way to reduce the heat load of the divertor targets and the other PFCs.

1.4 The thesis structure

The main aim of this thesis is studying the tokamak divertor in both physics and engineering: studying the divertor and SOL physics by the edge transport code modeling, and implementing the detachment feedback control on EAST. Chapter 2 states the physical model of SOL and the modeling code we used. Chapter 3 shows the modeling results of the EAST radiative divertor experiments from SOLEDGE2D-EIRENE code, including the radiation feedback control and the quasi-snowflake (QSF) discharges. Chapter 4 analyzes the current radiative divertor feedback control experiment results on EAST. Chapter 5 summarizes all the work and gives an outlook.

Chapter 2. The SOL physics and the radiative divertor method

As the statement in the section 1.3, understand the SOL physical characters can help us to solve the excessive heat load issue of the divertor. In this chapter, the two-point model is used to describe the energy and particle transport processes of the edge plasma, and a summarize of the radiative divertor conception is given.

2.1 The main functions and structure of the divertor

Modern divertor systems have four main functions^[20]: 1. The energy dissipation of plasma; 2. The vacuum pump installed in the divertor area is used to remove impurities and fusion products "Helium ash"; 3. Reducing the generation of impurity particles and 4. Impurity shielding, to prevent the impurity particles entering the main plasma.

A complete divertor is usually made up of the target plate, the magnetic coil, the pump, the water-cooling system, the gas puffing valves, etc. The divertor geometric structure and closure are the essential factors which affects the plasma neutral recycling and the impurity shielding^[21-22]. Figure 2.1 shows the structure of the EAST upper tungsten divertor.

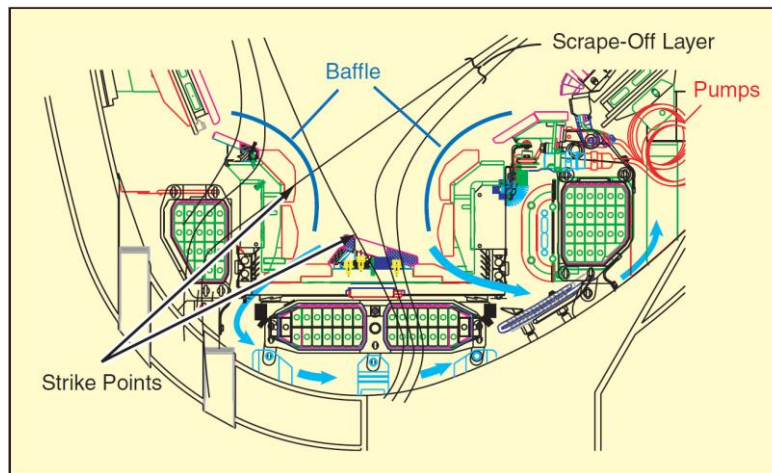


Figure 2.1 The schematic of the modern divertor structure (quoted from reference^[23]).

The magnetically confined plasma shape is mainly determined by the magnetic field configuration, and also influenced by the tokamak geometry. Due to the strength of the magnetic field is finite, only the plasma inside the last closed flux surface (LCFS) is constrained well. Outside LCFS, there is an open magnetic field region, the charged particles move along the magnetic field line and travel to the vacuum chamber wall.

Under the limiter shape, LCFS is very close or even tangent to the limiter surface. Such a straight contact between the main plasma and the surface materials makes the sputtered impurity particles enter plasma core easily, degrading the core confinement. Under the divertor configuration, the divertor coil produces a current with the same direction to the plasma current, the two currents generate X point(null point) between the divertor and the main plasma separatrix, in which the poloidal magnetic field(B_p) is zero. The cross magnetic lines which are passing through X point form LCFS of the plasma under the divertor configuration. The plasma shape forms two "legs" connected with the divertor and forms strike point (SP) on the divertor target plate, as shown in figure 2.2.

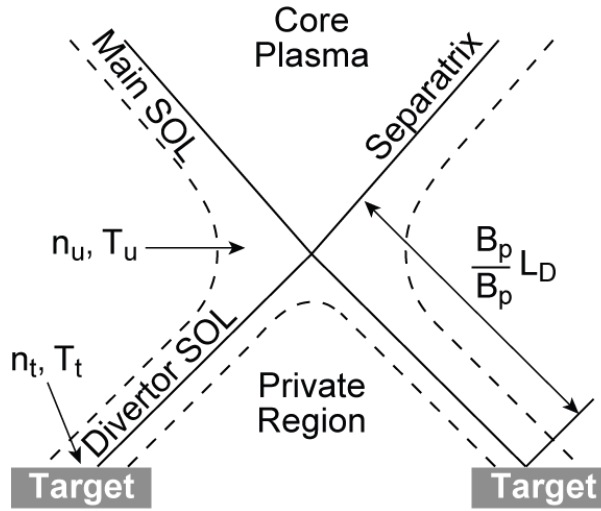


Figure 2.2 The LCFS (separatrix), X-point and SOL in the divertor configuration.

The divertor configuration keeps the region where PWIs happen away from the main plasma. Besides, the elongated plasma shape shows better confinement compared to the limiter shape^[23]. Because under the divertor configuration, the open

magnetic field lines connects to the divertor target, the heat and particle fluxes from the main plasma will mainly flow to the divertor target. The region near the divertor target surface thus is the strongest area where PWIs happen.

2.2 The two-point model of SOL

2.2.1 The basic definition of the two-point model

In theory, the plasma can only move along the magnetic field lines. However, because the thermodynamic characters of the plasma and the finite magnetic field strength, the power and particle fluxes in the open magnetic field region will also have a cross-field transport.

The intensity of the cross-field transport is much weaker than the parallel transport along the magnetic field line, so the competition between both transports forms a thin flux tube outside the separatrix, so-called the scrape-off layer(SOL). Most of the heat and particle fluxes from the main plasma move to the divertor target along SOL finally. Considering the toroidal symmetry of the tokamak and assuming that the plasma pressure is conserved along the magnetic field lines, the poloidal profile of SOL can be simplified to the one-dimensional model.

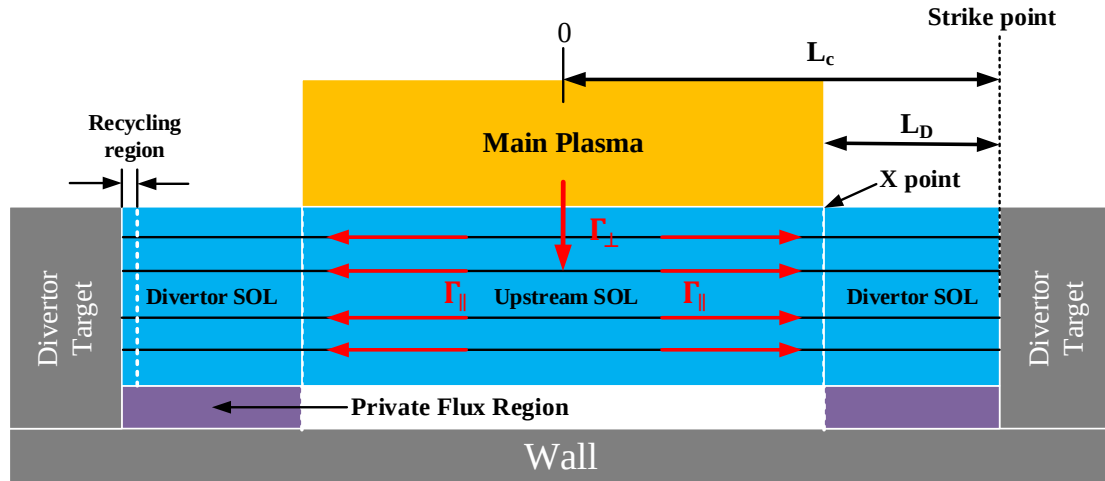


Figure 2.3 Schematic diagram of one-dimensional SOL two-point model. The main plasma is connected to the divertor target by the radial and parallel transport. SOL is divided into the upstream and the divertor region, demarcated by X point, and private flux region is squeezed between the two strike points (purple region). The two-point

model is symmetrical to the targets of the inner and outer divertor, which is shown in the figure as left-right symmetry. It should be mentioned that this figure does not show the actual geometrical width and area.

This model can analyze the transport process of the plasma boundary area between X point (regarding the upstream of X point is unified) and the strike point, estimating the density and energy distribution on the divertor targets, and it is called the two-point model (TPM)^[20, 24-27]. Figure 2.3 is the schematic of the 1-D two-point model.

2.2.2 The cross-field transport in SOL

The energy conservation of plasma can be expressed as following^[20] :

$$P_{\text{SOL}} = P_{\text{in}} - P_{\text{rad main}} - \frac{dW}{dt} \quad (2.1)$$

In this formula, P_{in} represents the total input power, $P_{\text{rad main}}$ represents the radiated power of the main plasma, $\frac{dW}{dt}$ represents the loss rate of the main plasma stored energy.

The energy in SOL can be divided into two parts: various radiation processes consume a part of the SOL energy, and the rest reaches to the divertor targets. Defining P_{div} as the heat power on the divertor target, we have:

$$P_{\text{SOL}} = P_{\text{rad SOL}} + P_{\text{div}} \quad (2.2)$$

The cross-field transport of particles in SOL region can be expressed by the following formula^[28]:

$$\frac{d}{dr} (D_{\perp} \frac{dn}{dr}) = \frac{nc_s}{L_c} \quad (2.3)$$

where $D_{\perp}$ represents the diffusion coefficient of particles in the cross-field transport, c_s represents the thermal velocity during particle diffusion, and L_c represents the connection length from the SOL upstream to the divertor target plate. The width of sheath is ignored because it is too much smaller compared with the SOL connection length. Doing integration to the equation (2.3) in SOL:

$$n(r) = n(a) \exp\left[-\frac{r-a}{\lambda_n}\right] \quad (2.4)$$

where a represents the small radius of LCFS, and λ_n represents the folding length of the radial particle transport in SOL, i.e., the width of SOL:

$$\lambda_n = \left(\frac{D_{\perp} L_c}{c_s} \right)^{1/2} \quad (2.5)$$

It can be seen that the width of SOL is related to the strength of the cross-field transport of particles, the upstream and downstream connection length and the electron&ion temperature (particle sound velocity), where $D_{\perp}$ is about the Bohm diffusion magnitude^[25]:

$$D_{\perp} = \frac{cT_e}{16B} \approx 0.06 \frac{T_e(\text{eV})}{B(\text{T})} \quad (2.6)$$

Equation (2.1) indicates that the energy of SOL are completely from the cross-field transport of the main plasma which penetrates the separatrix, so the linear approximation of SOL energy is:

$$P_{\text{SOL}} = A_p n \chi_{\perp} \left(\frac{T}{\lambda_T} \right) \quad (2.7)$$

Where $A_p \approx (2\pi R_0)(2\pi a)\kappa^{1/2}$ represents the area of the main plasma, κ represents the elongation ratio of the plasma, R_0 and a represent the large and small radius of the plasma separately, $\chi_{\perp}$ represents the cross-field thermal diffusion coefficient, λ_T represents the characteristic decay length of SOL temperature. It is difficult to obtain an accurate expression of λ_T because the temperature distribution is affected by many factors (including some unknown factors). If the contribution of ions to heat transport is ignored, the linear approximation relationship between λ_n and λ_T can be obtained^[2]:

$$\frac{D_{\perp}}{\lambda_n} \left(\frac{5}{2} + \frac{\chi_{\perp}}{D_{\perp}} \frac{\lambda_n}{\lambda_{Te}} \right) = \frac{\gamma_s^e c_s}{2L_c} \left(\frac{1}{\lambda_n} + \frac{1}{\lambda_{Te}} \right)^{-1} \quad (2.8)$$

Equation (2.8) is only used for the qualitative analysis, the simple mathematical relationship $\frac{\lambda_p}{\lambda_T} = \frac{2}{7}$ is used more often. The upstream parallel heat flux of SOL is determined by the plasma configuration and P_{SOL} ^[20]:

$$q_u = \frac{1}{4\pi a \kappa^{1/2} N} \frac{q_{95} P_{\text{SOL}}}{\lambda_p} \quad (2.9)$$

In which, N represents the number of the null points of plasma, and q_{95} represents the safety factor of 95% flux surface. Equation (2.7) is substituted into Equation (2.9) and

then we can obtain:

$$\lambda_p = \left[\left(\frac{2\pi R_0 q_{95}}{7N} \right) \frac{n_{\perp} T}{q_u} \right]^{1/2} \quad (2.10)$$

It can be seen from the above equation that, for the plasma with the fixed magnetic topological structure, the decay length of radial power of SOL is determined by the density of upstream particles(n), temperature(T), cross-field transport intensity($\chi_{\perp}$) and the parallel heat flux(q_u), especially the relationship between the strength of the cross-field heat conduction and the parallel heat convection.

According to the calibration method of Eich et al., the power decay scale of H-mode plasma SOL in the experiment following the following calibration law ^[29]:

$$\lambda_p \propto B_T^{-0.8} q_{95}^{1.1} P_{SOL}^{0.1} R_{geo}^0 \quad (2.11)$$

This formula is widely used in physical analysis.

2.2.3 The parallel transport in SOL

As mentioned in 2.2.1, the strength of the cross-field(radial) transport is much weaker than parallel transport, so it can be considered that the density, momentum and energy conservation in direction parallel to the magnetic lines in SOL ^[21]:

$$\frac{d}{dx}(nv) = S_p(x) \quad (2.12)$$

Equation (2.12) represents particle conservation, in which n is the particle density in SOL, v is the velocity, and $S_p(x)$ on the right is the particle source from the main plasma. Ignoring the other particle sources, the particle density in SOL is completely determined by the upstream cross-field diffusing particle source;

$$\frac{d}{dx} [nm_i v^2 + n(T_e + T_i)] = -m_i v \overline{\sigma v}_{in} n n_n \quad (2.13)$$

Equation (2.13) represents the conservation of momentum and can also be regarded as the conservation of parallel pressure from upstream to the divertor in SOL. In this equation, the first term on the left represents the dynamic pressure of the ions, and the electron mass is so small that is neglected. The second term on the left represents the thermal pressure, where the electron and ion temperature (T_e , T_i) has been multiplied by Boltzmann constant. The right side of the equation represents the

pressure loss caused by the friction between the ion and the neutral particle, $\overline{\sigma v}_{in}$ represents the collision cross section between the ion and the neutral particle, and n_n represents the density of the neutral particle. In theory, the plasma viscosity effect should also be considered, and here is ignored;

$$\frac{d}{dx} \left\{ \left[\frac{1}{2} m_i v^2 + \frac{5}{2} (T_e + T_i) \right] n v - \kappa_0 T_e^{5/2} \frac{dT_e}{dx} \right\} = Q_R + Q_E \quad (2.14)$$

Equation (2.14) represents the energy conservation. In the equation, the first term on the left curly bracket represents the parallel thermal convection of electrons and ions. The kinetic energy of ions is also transported to the target plate, and the electron term is ignored as a very small variable. The second term on the left in curly bracket indicates the parallel heat conduction of the electrons. The temperature of the ions is generally considered to a constant and thus is ignored. The two subtitles on the right of the equation represent Joule heat and heat flux from the main plasma.

In equation (2.14), Based on the work of Spitzer and Harm, the parallel electron heat transfer in SOL can be expressed as following (the heat transfer coefficient of ions is much smaller than electrons and thus can be ignored, so the electron subindex is omitted^[30]):

$$q_{//} = -\kappa_0 T^{5/2} \frac{dT}{dx} \quad (2.15)$$

By integrating the equation (2.15) from the symmetry center of the two-point model to the divertor target (connection length L_c), we can obtain:

$$T_u^{7/2} \approx T_t^{7/2} + \frac{7q_u L_c}{2\kappa_0} \quad (2.16)$$

In this formula, T_u and T_t are the upstream and downstream temperatures in SOL region separately, q_u is the SOL upstream heat flux, and constant κ_0 is defined as following:

$$\kappa_0 = \frac{(4\pi\epsilon_0)^2}{e^4 \ln \Lambda m_e^{1/2} Z_{eff}} \quad (2.17)$$

where m_e is the electron mass, $\ln \Lambda$ is the Coulomb logarithm, which is usually set to 15, and Z_{eff} is the effective charge number in the plasma. Assuming that the electron temperature of the divertor target is much lower than the upstream electron temperature, equation (2.16) can be simplified as:

$$T_u \approx \left(\frac{7q_u L_c}{2\kappa_0} \right)^{2/7} \quad (2.18)$$

Under this situation, the downstream SOL(i.e. the divertor) plasma has no influence on the upstream SOL plasma.

According to Equation (2.13), if neglecting the pressure loss caused by the friction between ions and neutral particles, assuming that $T_e \approx T_i = T$, the momentum conservation equation (2.13) can be changed into:

$$\frac{d}{dx} [nm_i v^2 + 2nT] = 0 \quad (2.19)$$

Integrating equation (2.19) in the parallel direction:

$$nT(1+M^2) = \text{const} \quad (2.20)$$

where M represents the Mach number. According to Bohm criterion^[31] which will be clarified in the section 2.2.4, when plasma reaches sheath entrance, $M = 1$, so the relation between upstream and downstream SOL is as following:

$$n_u T_u = 2n_t T_t \quad (2.21)$$

If considering the friction momentum loss, a more general form can be obtained:

$$f_m n_u T_u = 2n_t T_t \quad (2.22)$$

In the above formula, the value of f_m is between 0 and 1, a higher value means a smaller pressure loss.

2.2.4 The relations between the main plasma, SOL and the divertor target

The structure of the two-point model shows that the entrance of SOL is the main plasma separatrix, and the sheath is the downstream boundary of the main SOL region. Therefore, the SOL particle and energy transport can be divided into three parts:

1. The core plasma determines the density and temperature at the LCFS of the plasma, i.e., the SOL entrance. According to the phenomenological convection model, the relation between particle density and temperature at the separatrix and them of the core plasma is as following^[2]:

$$n_e(a) = \frac{\lambda_n \langle \sigma v \rangle_i \langle (n_e) \rangle^2}{2f^2 v_0} \quad (2.23)$$

$$T_e(a) = 0.3 \langle T \rangle \left(1 - \frac{P_{\text{rad main}}}{P_{\text{in}}} \right) \frac{\tau_p}{\tau_E} \quad (2.24)$$

In the above two equations, λ_n represents the width of SOL, $\langle\sigma v\rangle_i$ represents the collision cross section of neutral particle and ion, $\langle n_e \rangle$ represents the average density of the core plasma, f represents the peaking degree of the plasma density, v_0 represents the penetration velocity of the neutral particle. Equation (2.23) qualitatively indicates that the upstream density of SOL is proportional to the square of the average density of the plasma at the core. τ_p and τ_E represent the confinement time of the core particles and energy separately. The confinement time decreases with the increment of the plasma density, and the SOL temperature will also decrease.

2. The cross-field(radial) and parallel transport of SOL are described in 2.2.2 and 2.2.3.
3. The effect of sheath to the divertor heat flux. Sheath is the last part of downstream SOL connecting to the divertor target. Because electron mass is very smaller than ion and the thermal velocity of electron is much higher, the electrons arrive earlier to the divertor target than ions. This difference forms a negative potential electric field with a thin width. This electric field accelerates ions while reduces the electron velocity to achieve the dynamic balance. In addition, there exists a pre-sheath area to accelerate ions before entering the sheath boundary^[24].

If ignoring the radiation loss in SOL, the sheath is the only "sink" of the heat and particle fluxes. Figure 2.5 shows the sheath structure and electric potential variation schematic diagram. The thermal velocity of gas diffusion is determined by Equation (2.25)^[5]:

$$c_s = \left(\frac{2T}{m} \right)^{1/2} \quad (2.25)$$

Setting the potential at the sheath entrance is 0, the sheath electric field can be described by the Poisson equation^[2,25] :

$$\frac{d^2\phi}{dx^2} = \frac{e}{\epsilon_0} (n_e - n_i) \quad (2.26)$$

Where n_e and n_i represent electron and ion density separately, and ϵ_0 represents dielectric constant, which can be obtained from Boltzmann distribution and ion continuity equation:

$$n_e = n_0 \exp\left(\frac{e\phi}{T_e}\right) \quad (2.27)$$

$$n_i = n_0 \left(\frac{\frac{1}{2} m_i v_0^2}{\frac{1}{2} m_i v_0^2 - e\phi} \right)^{1/2} \quad (2.28)$$

Where n_0 is the plasma density at the sheath entrance, v_0 is the velocity of ions entering sheath entrance. Compared with the plasma, the electric potential of the sheath is very small and the electric potential in the sheath changes slowly. Substituting equations (2.27) and (2.28) into (2.26), the approximate form can be obtained:

$$\frac{d^2\phi}{dx^2} = \frac{\phi}{\lambda_D^2} \left(1 - \frac{T_e}{m_i v_0^2} \right) \quad (2.29)$$

Where:

$$\lambda_D = \left(\frac{\epsilon_0 T_e}{n_0 e^2} \right)^{1/2} \quad (2.30)$$

λ_D is Debye radius, indicating that sheath thickness is on the same order of Debye radius. When equation (2.5) has a stable solution and meets the condition that ϕ changes very slowly, it must have:

$$v_0 = \left(\frac{T_e}{m_i} \right)^{1/2} \quad (2.31)$$

The right part of equation (2.31) is same to the acoustic velocity c_s of ions, which proves that ions enter sheath with the sound speed. This conclusion is called Bohm Criterion^[31]. When the SOL pressure is conserved, the sheath determines the plasma parallel heat flux arriving at the divertor targets.

The potential difference between the sheath entrance and the divertor target surface for the deuterium (D) plasma is $V_f \approx -\frac{3T_e}{e}$ ^[25]. Therefore, the ions arriving at the wall also gain the additional kinetic energy of sheath acceleration of $-eV_f$. Therefore, the parallel heat flux on divertor targets is expressed as following:

$$q_{// \text{ sh}} = n_0 c_s (2T_e + 2T_i - eV_f) \quad (2.32)$$

In general, considering the collision cross section of electrons and ions and the secondary electron excitation coefficient, parallel heat flow equation (2.32) can be rewritten as^[20]:

$$q_{//sh} = \gamma_s \Gamma_{//} T_e \quad (2.33)$$

In which particles $\Gamma_{//} = n_0 c_s$, deuterium plasma the sheath energy transmission coefficient of D plasma γ_s is about 7^[20].

In conclusion, TPM provides us the available way to analyze and control the particles and temperature in SOL. Theoretically, by adjusting the energy balance of SOL (mainly adjusting the radiation power loss) and the upstream plasma density, the divertor heat and particle flux distributions can be changed correspondingly, and the divertor temperature also can be decreased.

2.3 The recycling state of SOL and the detachment

In the section 2.2, the relation between the upstream and downstream density and temperature of SOL, equation (2.21) is obtained by neglecting the pressure loss caused by the collision between ions and neutral particles. However, the momentum loss between ions and neutral particles will increase with the decrease of the divertor temperature, and the heat conduction process is also enhanced correspondingly. In this case, the sheath will no longer completely determine the divertor heat flux, and the SOL characteristics will also change. Assuming that there is no energy and momentum loss in the process of SOL parallel transport, and the divertor heat flux is completely determined by the sheath, the relationship between the SOL upstream and downstream temperatures can be obtained by combining the equations (2.18), (2.21) and (2.33):

$$\frac{T_u}{T_t} = \frac{1.5 \gamma_s^2 n_u^2 L_c^{6/7}}{m_i \kappa_0^{6/7} q_u^{8/7}} \quad (2.34)$$

The physical meanings of the parameters in the equation have been explained in the above sections. According to equation (2.34), if there is a large temperature gradient in SOL, such as $T_u > 3T_t$, we have:

$$\frac{n_u^2 L_c^{6/7}}{q_u^{8/7}} > 2 \times 10^{31} \quad (2.35)$$

Equation (2.35) can be used as the criterion to divide SOL into three recycling states:

1. Sheath limited SOL

Also called low recycling SOL. For the high-heated low-density plasma, the equation (2.35) is invalid, and it can be considered that the SOL temperature is almost constant:

$$T_t \approx T_u \quad (2.36)$$

, which means the upstream and downstream temperatures of SOL are equal, and according to Equation (2.21), it can be obtained:

$$n_t \approx \frac{1}{2} n_u \quad (2.37)$$

In this case, the heat flow of divertor target plate is completely determined by the heat flux of sheath, i.e.,

$$q_t \approx q_u \approx q_{//\text{ sh}} = \gamma_s n_u T_u^{3/2} (2m)^{-1/2} \quad (2.38)$$

substituting it into Equation (2.10), the radial power decay length in the low recycling state can be obtained:

$$\lambda_p = 1.8 \frac{m_i^{1/5} R_0^{3/5} a^{1/5} \kappa^{1/10} q_{95}^{2/5} n_u^{1/5} \chi_{\perp}^{3/5}}{\gamma^{2/5} N^{2/5} P_{\text{SOL}}^{1/5}} \quad (2.39)$$

2. Heat conduction limited SOL

When the plasma density further increases and the power outflow from the main plasma decreases, satisfying the criterion equation (2.35), the influence of thermal convection on the downstream temperature is weakened due to the increase of the collision degree of the target particles, and the heat conduction process is enhanced. In this case, the SOL temperature gradient increases. The collision friction process of the target particles is enhanced, but the resulting momentum loss can still be ignored (f_m is still 1). In this case, equation (2.18) can be considered to be completely valid, and the target plate temperature can be obtained by substituting it into Equation (2.34):

$$T_t = 0.98 \frac{m_i}{\gamma_s^2} \left(\frac{\kappa_0}{L_c} \right)^{4/7} \frac{q_u^{10/7}}{n_u^2} \quad (2.40)$$

From Equation (2.21), the density of the target plate can be obtained as

$$n_t = 0.73 \frac{\gamma_s^2}{m_i} \left(\frac{L_c}{\kappa_0} \right)^{6/7} \frac{n_u^3}{q_u^{8/7}} \quad (2.41)$$

All the parameters are substituted into Equation (2.10) to obtain the radial power decay length under the high recycling state:

$$\lambda_p = 6.4 \frac{1}{\gamma^{2/5} \kappa_0^{2/9}} \frac{R_0 a^{5/9} \kappa^{5/18} q_{95}^{4/9} n_u^{7/9} \chi_{\perp}^{7/9}}{N^{4/9} P_{SOL}^{5/9}} \quad (2.42)$$

It can be seen from the equation that:

$$T_t \propto \frac{1}{n_u^2} \quad (2.43)$$

$$n_t \propto n_u^3 \quad (2.44)$$

The upstream density plays a decisive role on the divertor density and temperature distribution. Increasing the upstream plasma density can reduce the divertor temperature effectively, while the divertor plasma density also increases greatly. Besides, the sputtering of the divertor target is suppressed, the impurity production is reduced. Under this recycling state, the neutral recycling is enhanced significantly, so the heat conduction limited SOL is also called the high recycling SOL.

3. Detachment

If the divertor temperature decreases further from the high recycling state, the charge exchange and elastic scattering processes between plasma and neutral particles on the divertor target will exceed the ionization of neutral particles. Such collision and friction processes will lead to bigger momentum loss of ions, and the pressure from SOL to the target plate will no longer maintain the conservation. Under this SOL condition, the heat flow will decrease greatly. Theoretically, there is a possibility that the plasma and the divertor target will detach completely in the thermal contact, which is called detachment.

The detachment has a strong effect on mitigating the divertor heat load. Experimental studies have found that when the electron temperature of the target

plate ($T_{e, \text{div}}$) drops below 10eV, the pressure reduction of the target plate caused by friction is very strong^[32]. Once $T_{e, \text{div}}$ is lower than 4 ~ 5eV, the complete detachment will be achieved^[19], the plasma is easily becoming unstable, this conclusion is called Lackner-Borrass limitation^[33-34]. Detached plasma is easy to return to attachment or even disruption^[35], but it is also a good solution to lower the divertor heat load. Therefore, achieving the stable partial detachment is more practical in experiment^[32,36-38].

The calculation of the detachment degree is^[39]:

$$\text{DOD} = \frac{C \cdot \bar{n}_e^2}{I_{D^+}^{\text{measured}}} \quad (2.45)$$

In this formula, C represents the calibration constant, which is generally gained from the experimental results. $\bar{n}_e$ represents the average density of the main plasma, and $I_{D^+}^{\text{measured}}$ represents the diagnostic data of the ion flux of the divertor target.

The physical mechanism of detachment is still not completely clear, but according to the above analysis, we have two main methods to generate detached plasma in experiment:

- (1) Increase the upstream plasma density n_u of SOL. As shown in equation (2.23), when the core plasma density(also separatrix density) is high and no disruption occurs, the particle flux on the target plate rises substantially and drops significantly after the plasma enters the detachment, which is called "roll-over" process^[40-41]. This method has been realized in many tokamak experiments^[32-33,42-43].
- (2) Increase the SOL radiated power, which is in the name of the radiative divertor method. According to equation (2.2), when the SOL inflow power is fixed, the increment of the radiation consumption will reduce the divertor power load, and reduce the divertor electron temperature to few eV, in which the plasma enters the detachment phase. The radiation process in SOL is very complex and will be stated in the section 2.5. In theory, the radiated power ratio can

reach 100% of the total SOL power, the experiment has achieved radiating 90% of the plasma input power^[44]. The radiated power of the intrinsic impurity and the working gas particles is very difficult to achieve such a high radiation level, the additional impurity injection is necessary to realize the active detachment plasma.

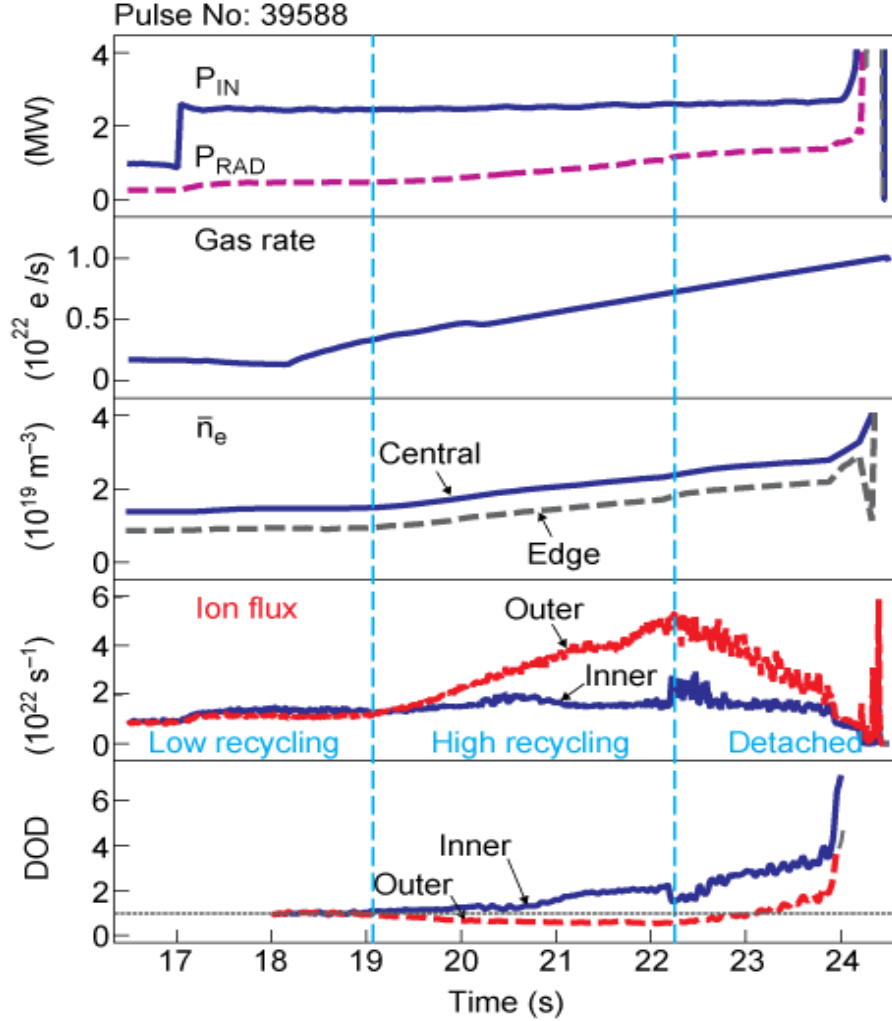


Figure 2.4 The detachment Roll-over process of the Ion Flux on JET plasma (quoted from reference ^[42]).

It should be mentioned that both the detachment methods may degrade the plasma confinement or even lead to the disruption. Some instability, like the multi-level asymmetric radiation (MARFE) ^[45-47], is easily appearing with the high radiation level or the high edge density gradient. It is better to apply a

feedback control to achieve the detachment, which can make the detachment phase more stable, keep the compatibility of the high core confinement and the divertor detachment.

In the detachment state, the neutralization rate of the ions in the SOL recycling region increases. Taking the deuterium plasma as an example, the deexcitation of the deuterium ions increases, the intensity of D Balmer line emission such as D_γ thus will increase. In the detachment state, the divertor electron temperature and the divertor surface temperature will decrease correspondingly, while the particle flux will go through the so-called “roll-over” process as stated in the section 2.3. Figure 2.4 shows a roll-over example on JET^[42]. In the detachment state, SOL radiation will also be enhanced, and the radiation will gradually move from the area near the divertor target to the X point region^[34]. The above physical parameters can be used as the controlled variable.

2.4 The basic characteristics of SOL radiation

The selection of the impurity gases used for the radiative divertor are mainly the medium- and low-Z impurities such as Nitrogen (N_2), Neon (Ne), Argon (Ar), Deuteride methane (CD_4), etc. Due to the challenging power consumption of the large-size fusion devices like ITER and DEMO, the feasibility of using high-Z impurities, such as Krypton (Kr), is also being studied^[38,48]. Considering that only SOL power consumption cannot meet the demand of the future reactors, the research of increasing the core plasma radiation has also been carried out. It is expected that achieving the two-step power exhaust both of the plasma core and SOL reduces the divertor heat flux to the level which the PFCs withstand for a long operating time^[49].

The SOL radiation process can be described by the coronal model or the collision-radiation (C-R) model^[50]. The main differences between these two models are: the coronal model is suitable for high-temperature and low-density plasma, while the C-R model is suitable for high-temperature and high-density plasma. In coronal or

C-R model, the radiation process can be divided into three parts: the line emission of atoms and molecules, the cascade recombination and the Bremsstrahlung. Radiation power loss thus can be expressed by the following formula^[2,46]:

$$P_{\text{rad}} = n_e n(X) \sum_{Z=0}^{Z_0} \frac{n(X^{Z+})}{n(X)} (P_L^{(Z)} + P_R^{(Z)} + P_B^{(Z)}) \quad (2.44)$$

where $n(X)$ is the density of element X , $n(X^{Z+})$ is the X ion density with the Z valence, Z_0 is the valence state of X ion which is completely stripped, $P_L^{(Z)}$, $P_R^{(Z)}$ and $P_B^{(Z)}$ are the power loss of the line emission of the ions in Z valence, the recombination and the bremsstrahlung radiation separately.

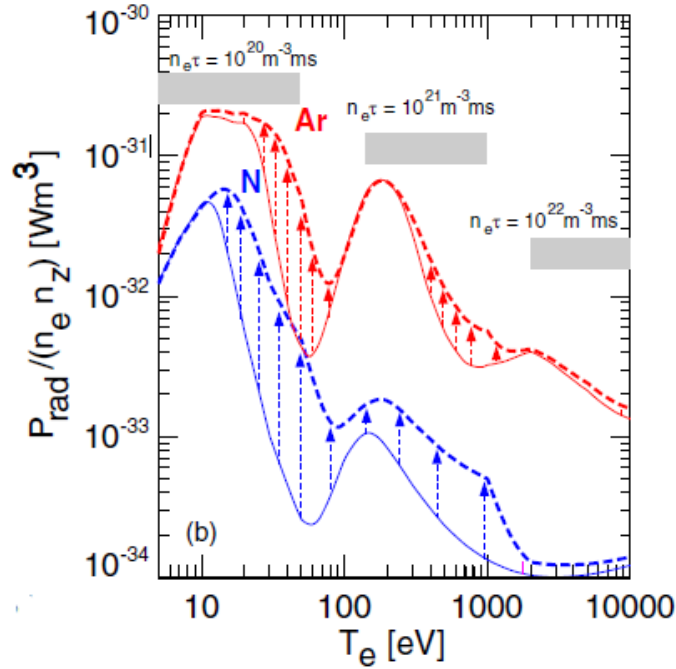


Figure 2.5 Comparison of radiation functions of N and Ar impurities in corona equilibrium and non-equilibrium states. The radiation energy of impurities in non-equilibrium state is higher (picture quoted from^[38])

In general, setting $L_z(T_e) = P_L^{(Z)} + P_R^{(Z)} + P_B^{(Z)}$ to represent the radiated power of a single impurity particle at a given electron temperature, which is defined as the so-called radiation function, also known as the cooling rate. It is obvious that L_z is a function only related to the electron temperature, the total radiated power of the impurity can be determined as long as the electron temperature and density of the

plasma are fixed.

The corona model describes the radiation under the dynamic balance state. However, the population time of the impurities in SOL is not enough to be stripped completely, that is, it is always in a non-equilibrium state and will continue to radiate more energy. Therefore, the calculation results of the corona model are generally lower than the actual radiated power of the SOL impurities, figure 2.5 is the revised radiation function of Nitrogen (N) and Argon (Ar)^[38].

2.5 The control method of the radiative divertor

One of the ultimate goals for the radiative divertor is to mitigate the heat load on the divertor target under the long-pulse high-performance plasma discharge, and further to achieve the stable detached discharge. Except the control methods mentioned in the section 2.3, the divertor magnetic configuration also helps to optimize the divertor heat load or generate the detachment. The typical advanced divertor configuration, such as the Snowflake Divertor (SFD)^[51], Super X-Divertor^[52] have been proved the effect on the divertor heat load mitigation by increasing the flux expansion, connection length and some other SOL topological parameters. EAST develops the Quasi Snowflake (QSF) divertor which is suitable for the fully superconducting tokamaks^[53]. In chapter 3 the QSF plasma will be modeled by SOLEDGE2D-EIRENE.

Radiative divertor control experiments have been implemented on many tokamaks. ASDEX-U has obtained important experimental results in this field^[36,44,54-55]. ASDEX-U has achieved the capability of dual-feedback control of divertor region radiation and the core plasma radiation^[56]. ASDEX-U also achieved the detachment plasma by injecting the different kinds of impurity gases^[49,57-58]. DIII-D injected D₂ into the divertor area to achieve radiation enhancement^[59]. DIII-D also built the small angle slot (SAS) divertor to lower the divertor particle flux by the strike point control and the special geometry structure^[60]. JT-60U achieves stable radiation control in the 30s long pulse H-mode discharge^[61], and TCV achieves combined experiment of the snowflake discharge and radiative divertor, proving that it is easier to obtain plasma

with high radiation ratio under snowflake configuration, and the effect of increased radiation on the divertor heat flux is stronger than that of the single null configuration^[62].

2.6 Summary

In this chapter, the physical models of divertor and SOL are stated. The upstream and downstream of SOL are connected through the parallel and cross-field(radial) transport. The parallel of the magnetic field transport is much stronger than the cross-field transport, so the strength of cross transport determines the width of SOL. The electron temperature and density of the divertor target can be adjusted by controlling the parameters of the upstream plasma, such as the radiated power and the main plasma density. Detachment is a special SOL recycling state, which has an obvious effect on mitigating the heat load on the divertor target. According to the physical mechanism, the detachment can be achieved by increasing the plasma density or greatly increasing the plasma radiated power ratio. The goal of the radiative divertor is to achieve stable and partially detached plasma with a high radiation ratio. Different physical parameters can be selected as control variables to achieve this goal, and the relevant work has been carried out in many tokamaks around the world.

Chapter 3. SOLEDGE2D-EIRENE modeling for EAST

3.1 The 2-D fluid model and SOLEDGE2D-EIRENE

The plasma modeling has two basic ways: the dynamic simulation is trying to solve the Vlasov equation of the charged particles in the electric-magnetic field, which will cost huge computing resources; the other more available way is the fluid modeling which regards the plasma as a fluid composed by the charged particles. In this way, the modeling code is to solve the Braginskii equations with the Bohm boundary conditions, which describe the conservation of the charged particles, momentum and power in SOL^[63].

The Bohm boundary condition describes the sheath effect near the divertor, limiter and the other PFCs surfaces. It includes three items: 1. the ions own acoustic velocity at the sheath entrance, $M=1$; 2. the ion temperature is a small constant in the sheath, i.e., $\frac{\partial T_i}{\partial s_{\parallel}} = 0$; 3. the gradient of the electron temperature is $\frac{\partial T_e}{\partial s_{\parallel}} = -\frac{2}{\kappa_{0e}} T_e^{-3/2} \Gamma$, in which Γ is the particle flux entering the sheath.

Thus, the 2-D fluid model are almost same to the two-point model stated in the section 2.2 but in both magnetic parallel (poloidal direction) and the cross-field (radial direction) dimensions. In this situation, the cross-field transport is considered into the equations rather than only calculate the parallel one. Therefore, the particle, momentum and power conservation in SOL can be described as the following equations:

$$\frac{\partial}{\partial s_{\parallel}} (n v_{\parallel}) = \frac{1}{r} \frac{\partial}{\partial r} r \left(D_{\perp} \frac{\partial n}{\partial r} + n v_{\text{pinch}} \right) + S_{iz}(r, s_{\parallel}) \quad (3.1)$$

In equation (3.1), $s_{\parallel}$ represents the distance unit along the magnetic field line, while r represents the cross-field distance unit, v_{pinch} is the pinch velocity, so the first item in the bracket on the right side is the particle cross-field transport and the second item is the ion particle source. In general, the main particle source is the plasma, thus equation (3.1) describes the dynamic equilibrium of the parallel, cross-field transport and the particle outflow from the plasma core;

$$\begin{aligned} \frac{\partial}{\partial s_{\parallel}} [nm_i v^2 + n(T_e + T_i)] = & -m_i v \bar{\sigma} v_{in} n n_n + \frac{1}{r} \frac{\partial}{\partial r} r \left[m_i v_{\parallel} \left(D_{\perp} \frac{\partial n}{\partial r} + \right. \right. \\ & \left. \left. n v_{pinch} \right) \right] + \frac{1}{r} \frac{\partial}{\partial r} r \left(\eta_{\perp} \frac{\partial v_{\parallel}}{\partial r} \right) + S_{mom}(r, s_{\parallel}) \end{aligned} \quad (3.2)$$

Equation (3.2) represents the momentum conservation in the 2-D model. Compared to the equation (2.13), the second item in the square bracket on the right side and the third item on the right side in equation (3.2) are the radial momentum transport. The last item on the right side is the momentum source.

$$\begin{aligned} \frac{\partial}{\partial s_{\parallel}} \left[\left(\frac{5}{2} T_i + \frac{1}{2} m_i v_{\parallel}^2 \right) n v_{\parallel} \right] = & -n v_{\parallel} E_{\parallel} + \frac{1}{r} \frac{\partial}{\partial r} r \left[n \chi_{\perp}^i \frac{\partial T_i}{\partial r} - \right. \\ & \left. \left(\frac{5}{2} T_i + \frac{1}{2} m_i v_{\parallel}^2 \right) \Gamma_r \right] + Q_{eq} + Q_{Ei}(r, s_{\parallel}) \end{aligned} \quad (3.3)$$

$$\begin{aligned} \frac{\partial}{\partial s_{\parallel}} \left[\left(\frac{5}{2} n T_e v_{\parallel} - \kappa_0 T_e^{5/2} \frac{dT_e}{dx} \right) \right] = & -n v_{\parallel} E_{\parallel} + \frac{1}{r} \frac{\partial}{\partial r} r \left[n \chi_{\perp}^e \frac{\partial T_e}{\partial r} - \frac{5}{2} T_e \Gamma_r \right] - Q_{eq} + \\ & Q_{Ee}(r, s_{\parallel}) \end{aligned} \quad (3.4)$$

Equation (3.3) and (3.4) represent the ion and electron energy conservation separately. In equation (3.3), because of the Bohm condition, the heat conduction by the T_i gradient is ignored, but ions gain the kinetic energy from the sheath of $\frac{1}{2} m_i v_{\parallel}^2 n v_{\parallel}$. In the second item in the square bracket on the right side, $n \chi_{\perp}^i \frac{\partial T_i}{\partial r}$ is the cross-field heat conduction by the T_i gradient, $\chi_{\perp}^i$ is the cross-field thermal diffusion coefficient of the ion energy, and $(\frac{5}{2} T_i + \frac{1}{2} m_i v_{\parallel}^2) \Gamma_r$ is the cross-field heat convection by the radial particle flux Γ_r . Equation (3.4) is same to (3.3) except the subindex is electron, and because the electron mass is too small, the parallel kinetic energy is ignored.

All the above equations, boundary conditions and the radiation model stated in the section 2.4 compose the physical principle of the 2-D SOL simulation. Up to now, there are many different transport code for the edge plasma, such as the SOLPS/SOLPS-ITER^[64], EDGE2D^[54], UEDGE^[65], etc. SOLPS is the most widely used in the different tokamaks, such as TCV^[66], ASDEX-U^[67], JET^[68] and so on. It should be mentioned that there also some the 3-D modeling codes like EMC3^[69] and TOKAM3X^[70] which consider the toroidal asymmetric are also used in the different magnetic confinement devices.

In this thesis, we use SOLEDGE2D-EIRENE code on EAST for the first time.

Soledge2D is a typical 2-D edge modeling code and coupled with the Monte-Carlo code EIRENE to compute the neutral particle recycling^[71]. In the fluid modeling code, the various tokamak wall structure enforces remeshing the grids near the wall, which costs a lot of computing sources. SOLEDGE2D uses the “penalization technique” to mitigate this defect^[72]. The penalization technique is a kind of parameterizing method. By introducing a mask function in the Braginskii equations and Bohm conditions, the wall structure can be easily changed with the mask function parameters, and no need to rebuild the mesh on the wall structure, which improves the computing efficiency and the application range^[73]. SOLEDGE2D-EIRENE now has been applied on many tokamaks, such as TCV^[74], HL-2M^[75], and WEST^[76].

We use the SOLEDGE2D-EIRENE to model the EAST radiative feedback control experiment and the QSF plasma. The main goal of the radiative control experiment simulation is to analyze the impurity gas seeding effect in the edge plasma and the divertor targets. Modeling results will help to mitigate the control system. The QSF simulation is to compare the difference between the QSF shape and the normal USN shape. The SOLEDGE2D-EIRENE application will enrich the modeling tools on EAST, and will be applied on CFETR in the future.

3.2 Simulation for the neon seeding experiment

EAST has realized the radiative feedback control in the H-mode long-pulse discharges^[77-78]. The basic mechanism is using the real-time radiation signal from the Absolute Extreme Ultra Violet (AXUV) array as the controlled variable, and the PID controller adjusts the impurity gas injection from different gas puffing valves at different locations to increase the radiation power loss and reduce the divertor heat load. In order to understand the impurity seeding effect, we choose the shot with radiation feedback control and its reference shot without impurity seeding to simulate, the initial aim is fit the main diagnostic data, i.e., reproduce the actual experimental situation.

3.2.1 Experimental parameters and the modeling setup

The experimental conditions are almost same for these two analyzed upper single null (USN) plasmas. Both them are in H-mode and the plasma current $I_p=0.4\text{MA}$, the averaged line electrical density $n_e \approx 4 \times 10^{19}\text{m}^{-3}$. Figure 3.1 shows the shapes of these two discharges at 3.5s, they are almost same. All these discharges have the same upwards drift direction ($\mathbf{B} \times \nabla B \downarrow$). Shot #71019 is in the radiation feedback control scenario, and the impurity gas was neon (Ne). Ne gas is seeded with a pulse length of 100ms in the feedforward (FF) stage from the gas puffing valve located at the upper inner divertor at 3.0 s shown in figure 3.2(b), the gas puffing speed $\Gamma_{\text{UIV}} \approx 4 \times 10^{20}\text{s}^{-1}$. Instead, shot #71022 has no impurity seeding. Figure 3.2 also shows under this discharge conditions, in #71022, the background total radiated power without Ne seeding measured by the AXUV is about 0.5MW. In #71019, the set level of the radiation control is 0.8MW, the feedforward Ne injection made the total radiated power close to the set level, then the small feedback Ne pulses keep this level until the control phase end.

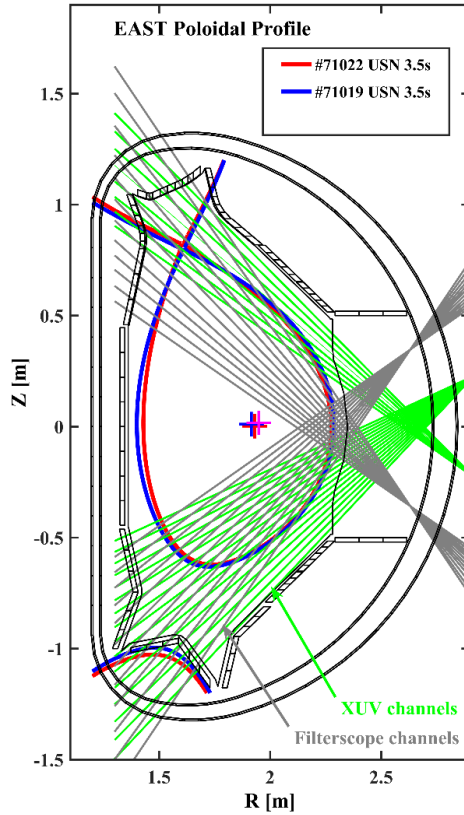


Figure 3.1 The poloidal shape of #71019 (red line) and #71022 (blue line). This figure also shows the edge AXUV channels (green lines) and the filter scope channels

(grey lines)

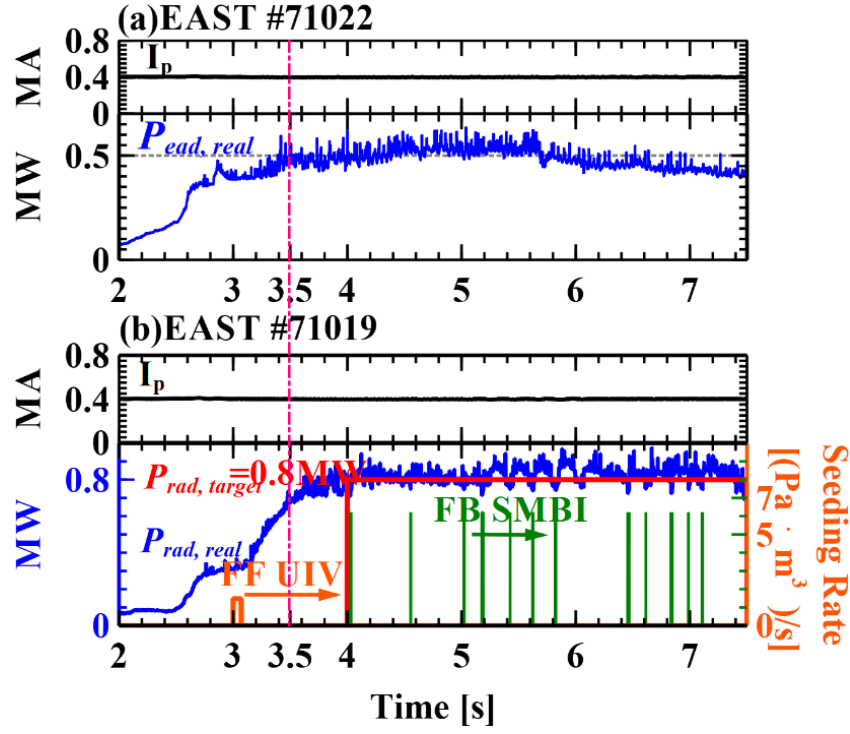


Figure 3.2 The discharge parameters of #71019 (with neon seeding) and #71022 (without neon seeding). In figure (b) the orange line shows the feedforward neon pulse at 3.0s with pulse length of 100ms, lifts the radiated power (blue line) close to 0.8MW, which is the target value of the radiation feedback control (red line). In figure (a), without neon seeding, the radiated power is almost flat of 0.5MW.

It is obvious that the feedforward Ne gas produced the additional radiated power of about 0.3MW. However, we lack of the diagnostic data to distinguish the ratio of the radiation increment in the core plasma and the upper divertor region. We choose the #71019 at 3.5s as the simulation time point. At this time, the input power is stable, the FF Ne injection was finished and no other gas injection happens. The radiation increment generated by this injection is close to the maximum, compared to the simulation of #71022 at the same time point, we can easily analyze the effect of the Ne impurity.

3.2.2 The effect of Neon seeding

Table 1 shows the main cross-field transport parameters setting in SOLEDGE2D modeling for these two shots. The cross-field parameters have no big difference between these two shots because they have the same basic discharge parameters, the only variable is w/wo neon seeding.

Table 1. Transport parameters in USN simulations.

Parameters	$D_{\perp,e\&i} [m^2 s^{-1}]$	$\nu_{\perp,e\&i} [m^2 s^{-1}]$	$\chi_{\perp,e\&i} [m^2 s^{-1}]$	$P_{in} [MW]$
#71022	1.8	1.8	1.0	1.78
#71019	1.5	1.5	1.0	1.96

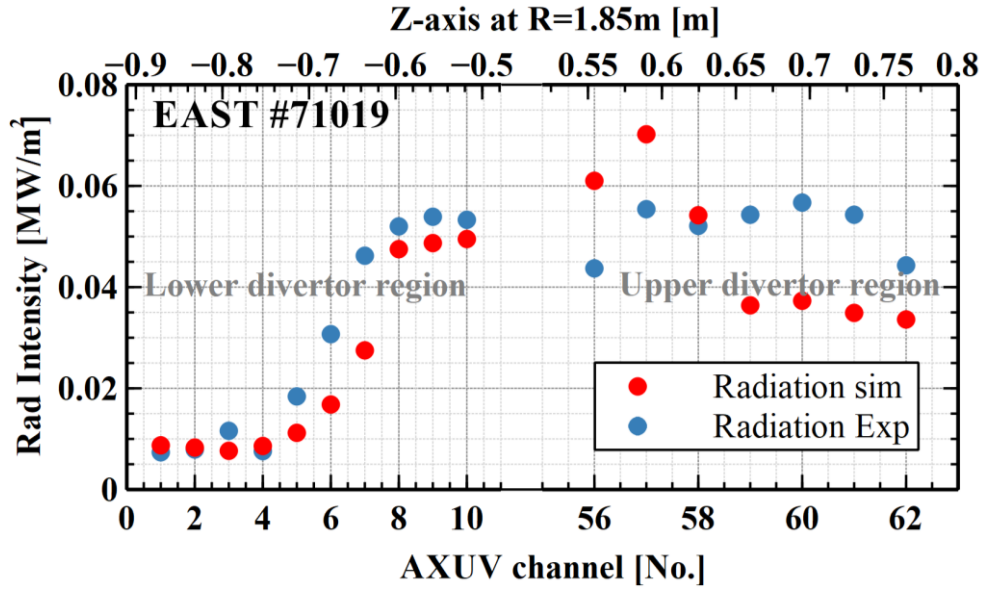


Figure 3.3 The SOLEDGE modeling radiation distribution in the lower and upper divertor region (red points) and the AXUV diagnostic data (blue points). The AXUV channels No.1-10 view the lower divertor region, and AXUV channels No. 56-62 view the upper divertor region

SOLEDGE2D-EIRENE cannot simulate the whole core plasma very well, so in the simulation of EAST#71019, the initial aim is making the modeling radiated power consistent with the radiated data from the AXUV single channels which are viewing the edge plasma as shown as the green lines in figure 3.1. Modeling result is consistent with the AXUV measurement in the upper and lower divertor region as shown in figure 3.3. Based on this result, the radiated power in the divertor region is about 0.23MW. At 3.5s, the total radiated power is about 0.7MW, so 32.9% of the total radiated power is from the divertor region. For the current research, the first aim

of the impurity seeding is increasing the radiated power in the divertor region rather than the core, the modeling result indicates that the ratio of the divertor radiation and the total radiated power still has a big improving space. The low edge electron temperature at the outer mid plane separatrix ($T_{e,u}$) is one of the main reasons to allow more Ne particles penetrating into the separatrix and increasing the core radiation substantially. Figure 3.4 shows the modeling electron temperature T_e profile of EAST #71019, T_e at the outer midplane separatrix is about 76~90eV.

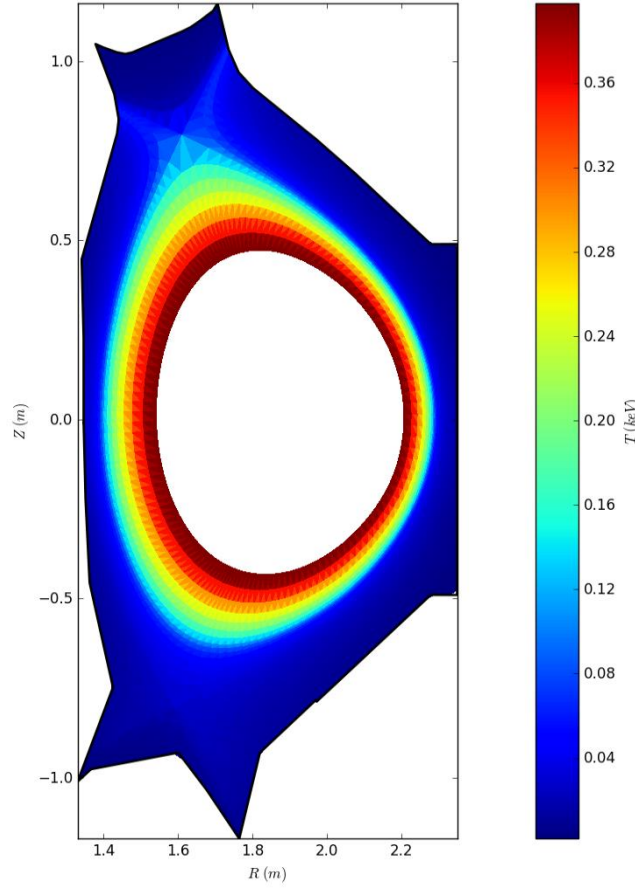


Figure 3.4 Modeled 2-D T_e distribution in shot #71019 with Ne impurity.

The simulated data of the divertor heat flux is almost same to the data measured by Langmuir probes as shown in figure 3.5. The additional Ne seeding provides the reduction of the divertor peak heat flux of about 0.15 MW/m^2 . The peak $T_{e,\text{div}}$ also declines about 10eV. In the series of radiation feedback control experimental, a slight increment of W line emission is measured during Ne seeding in another pulse^[77], which indicates that a few of W influx is in the main plasma after Ne seeding and

partly contributes to the core radiation. However, whether the neon seeding indeed cause the additional sputtering of W still lack of the enough experimental data, or just an accidental event.

To sum up the modeling results for shot #71019 and #71022 on EAST. The Ne seeding indeed plays a role on the divertor heat load mitigation despite it still has an improving space, the Ne gas volume should be higher to strengthen the radiation power loss. Because the temperature of EAST plasma is not high enough, the impurity shielding is not strong and let more Ne particles penetrate the main plasma separatrix. If the plasma density and temperature is higher, more Ne particles will be completely stripped in SOL, more radiated power will be in the divertor region and plasma edge, which is better to keep the main plasma confinement. Another way is trying to use the low-Z impurity gas as the working medium of the feedback control, like CD_4 , In Chapter 4 we will use CD_4 to control the divertor electron temperature ($T_{e,\text{div}}$) and compared with the neon seedings.

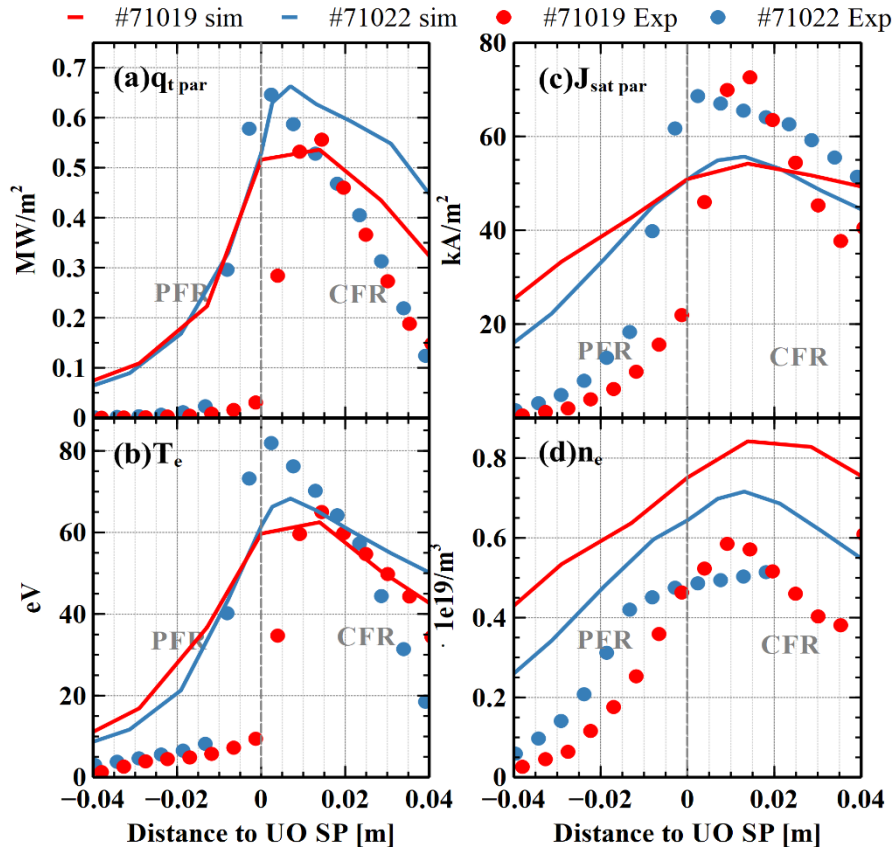


Figure 3.5 The simulation (line) and diagnostic (dot) data of the divertor parameters.

(a) parallel heat flux; (b) electron temperature; (c) saturation ion current; (d) electron density in #71019 (red) and #71022 (blue).

3.3 Simulation for the QSF discharge

The snowflake (SF) configuration was first established on TCV^[79]. It is characterized by a second-order null point(X point) in the poloidal magnetic field, in which both magnetic field and its derivative are zero($B_p=0$, $\nabla B_p=0$). This X point splits the separatrix into six segments: two enclose the plasma core and the other four connects to the divertor targets. The poloidal profile of the magnetic flux surfaces with a hexagonal null-point is similar to the snowflake. Theoretical studies indicate that the SF magnetic geometry lead to longer connection length in SOL, wider flux expansion and the bigger volume of the divertor SOL, which are beneficial for increasing the radiated power loss, suppressing the edge local mode(ELM) and facilitating the detachment phase^[80-81]. However, the standard SF configuration needs the precise value(I_{d0}) of the current of the divertor coil(I_d). Once the divertor coil current is not equal to I_{d0} , the standard SF will change into SF- ($I_d < I_{d0}$) and SF+ ($I_d > I_{d0}$) configurations, one second-order null point splits into two adjacent first-order null points^[82]. SF- and SF+ configurations also have the mitigation effect for the divertor heat load and more practical in experiment because they don't need the meticulous control of the divertor coil current.

Due to EAST does not have the specific divertor coils, and the superconducting components of EAST make the poloidal coils away from the core, it is more difficult on EAST to establish the SF configuration. EAST thus develops the substitution, so-called Quasi-Snowflake (QSF) configuration without need the strong divertor coils and the shaping control capability, which is more suitable for the fully superconducting tokamaks^[54]. Figure 3.6 shows the differences of the shape between SF and QSF, it is obvious that QSF is similar to SF- but the distance between two X points is longer.

Using the code modeling can help us to understand more about the SOL characteristics of QSF configurations

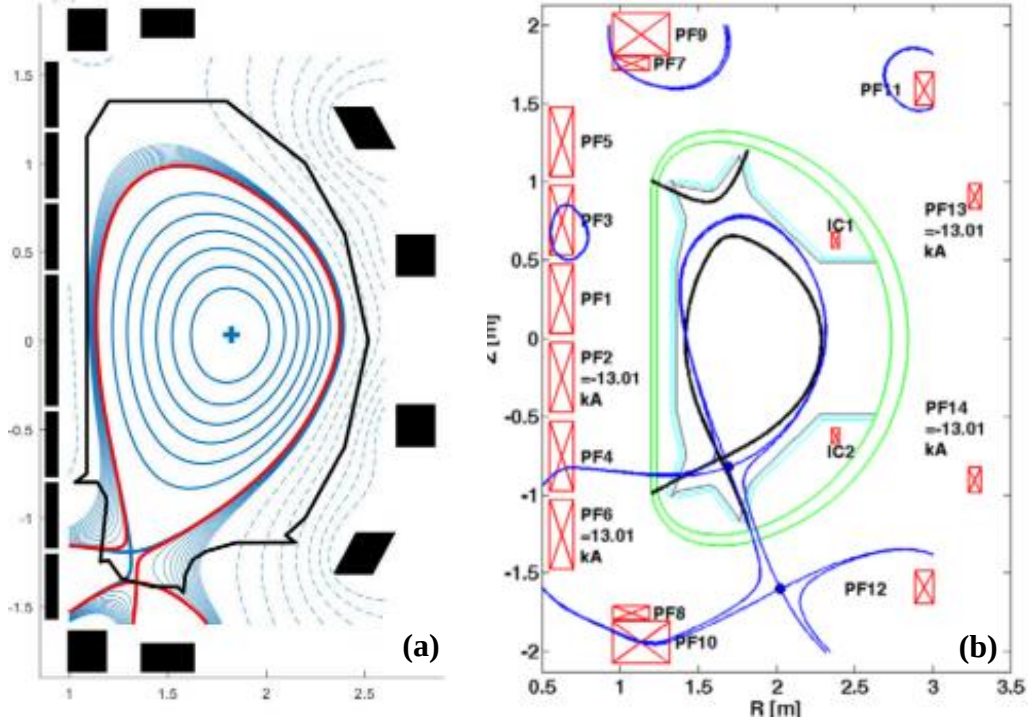


Figure 3.6 (a) The SF- configuration on HL-2M(quoted from reference^[75]), red lines is the hexagonal null points; (b) The QSF configuration on EAST (quoted from reference^[54]), blue lines is the QSF separatrix, the second X point is away from the tokamak structure.

3.3.1 The modeling setup of the typical QSF configuration

A typical QSF discharge on EAST #81562 is chosen to simulate, and another upper single null (USN) discharge #81565 is simulated as the reference shot. The plasma current(I_p) of both EAST#81562 and #81565 are 0.25MA, toroidal magnetic field $B_t=2.25T$. The line averaged electron density n_e is about $3.0 \times 10^{19} m^{-3}$. Figure 3.7 shows the main discharge parameters of EAST#81562. We chose the time point of 5.7s to reproduce the experimental situation because at this time the diagnostic data is complete.

Figure 3.8(a) shows poloidal profile of the shot #81562, the distance between the two one-order null X-point is beyond 1m. Thus, in the mesh building in SOLEDGE2D-EIRENE, the second X-point out of the tokamak actually is cut as shown in figure 3.8(b)

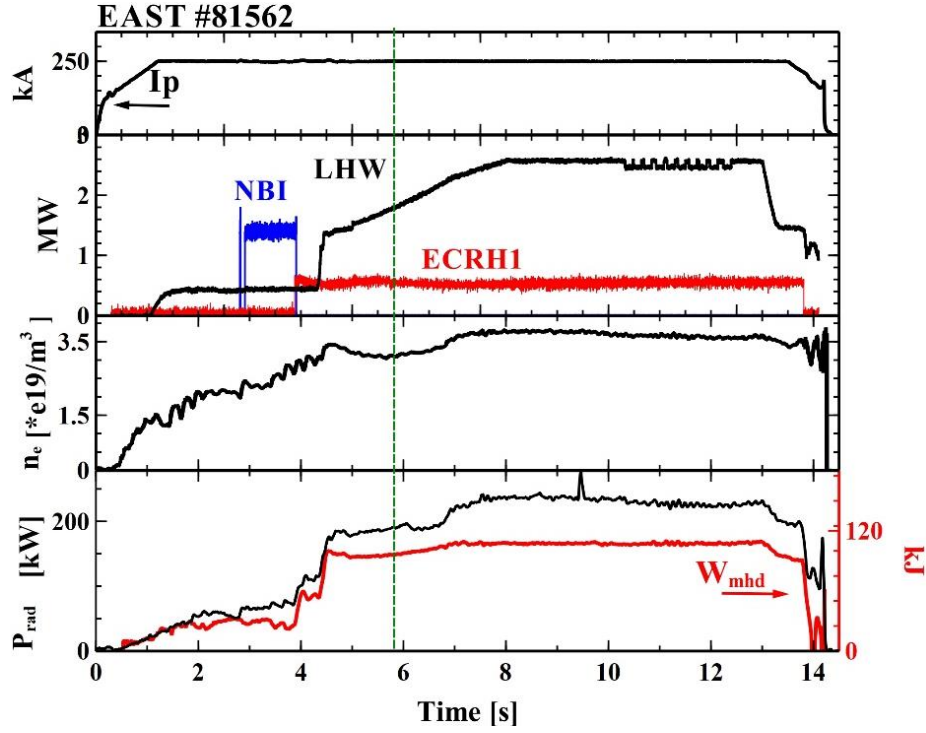


Figure 3.7 The basic discharge parameters of QSF shot #81562

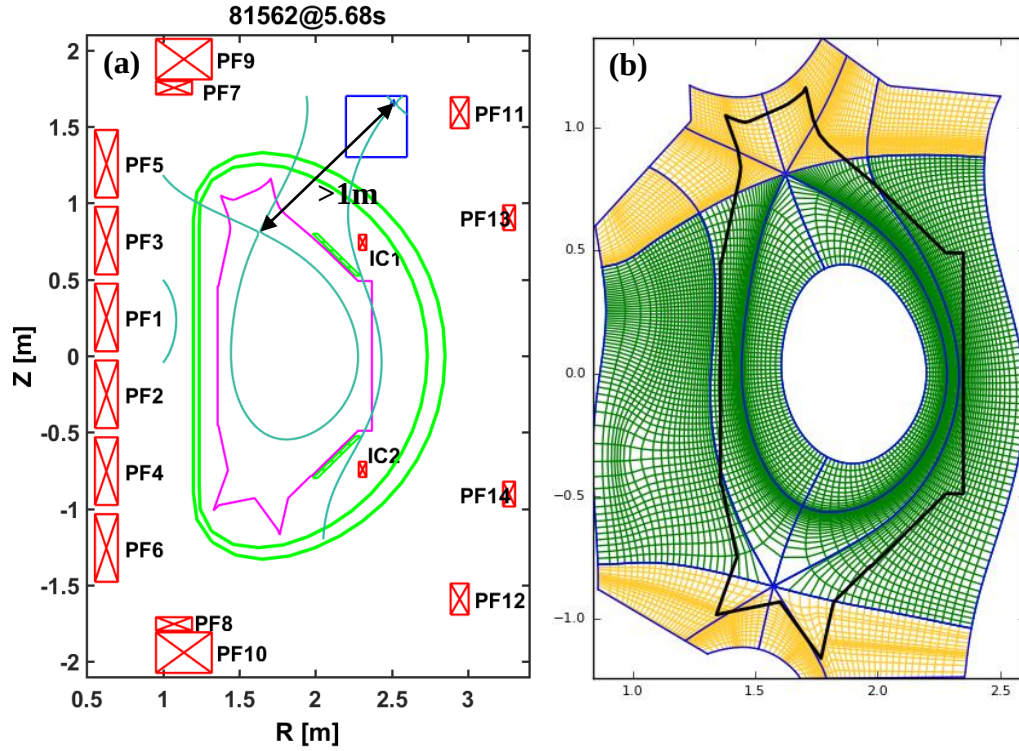


Figure 3.8 (a) The poloidal profile of the QSF shot #81562, the second X-point is lined out by the blue box; (b) The mesh built in the SOLEDGE2D-EIRENE of shot #81562, the second X point is already cut.

3.3.2 The QSF modeling results

Figure 3.9 shows the parallel connection length in SOL from the outer midplane to the divertor target (L_{par}) in the different EAST shots. The L_{par} of #81562 is longer than 2 times to L_{par} of shot #71022 (165m vs 75m). The longer connection length of QSF configuration is beneficial to increase the SOL volumetric power loss. However, because the discharge parameters of these two shots are completely different, it is hard to compare the mitigation effect on the divertor heat load of the QSF configuration.

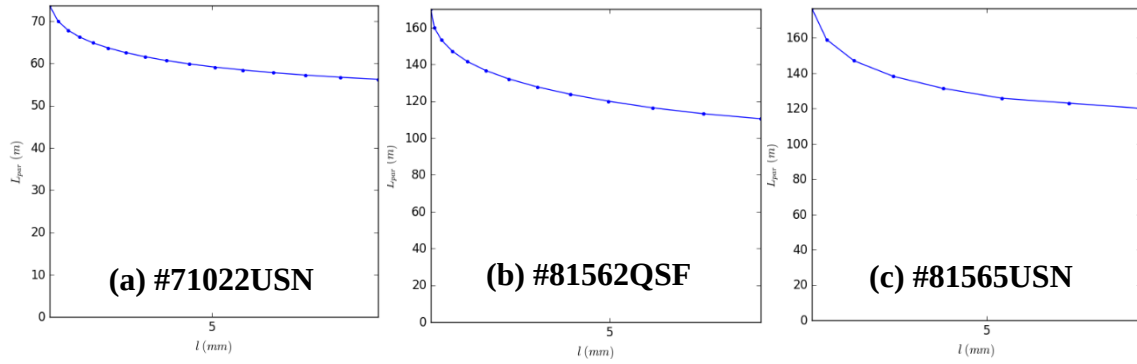


Figure 3.9 The parallel SOL connection length in 3 discharges, the x-axis is the radial distance to the outer midplane separatrix.

Figure 3.10 shows the modeling data and the diagnostic data of the outer midplane density ($n_{e,u}$) and temperature ($T_{e,u}$) profiles, the divertor target density ($n_{e,t}$) and temperature ($T_{e,t}$) distributions. The $n_{e,u}$ and $T_{e,u}$ inside the separatrix fit not well. Because part of the Langmuir probes are broken due to the long-term experiments, the original diagnostic data are not so precise, the simulation and diagnostic data cannot fit so well. We do the density scanning in the shot #81562 simulation, the electron density at the main plasma separatrix $n_{e,sep}=0.55-0.6\times 10^{19}/\text{m}^3$, the reproducing results relatively best. It is also can see that under this discharge conditions, when the $n_{e,sep}$ is higher than $0.8\times 10^{19}/\text{m}^3$, the SOL is into the high recycling state, $n_{e,t}$ is increased greatly, while $T_{e,t}$ is declined sharply, figure 3.11 shows the density scanning results.

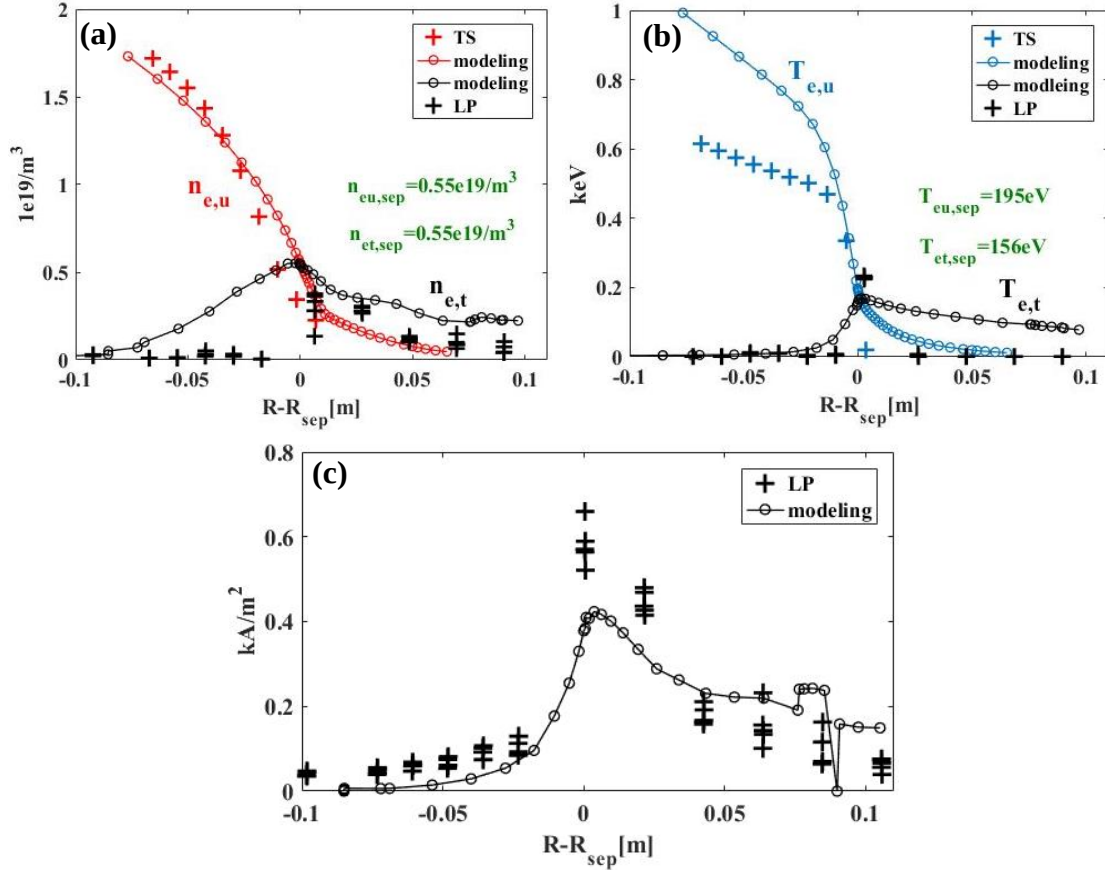


Figure 3.10 (a) The modeling $n_{e,u}$ profile (red solid line with circle), the $n_{e,u}$ profile measured by Thomson Scattering (TS) (red cross), the modeling $n_{e,t}$ distribution (black solid line with circle) and the $n_{e,t}$ distribution measured by the Langmuir probes (black cross). (b) The modeling $T_{e,u}$ profile (blue solid line with circle), the $T_{e,u}$ profile measured by Thomson Scattering (TS) (blue cross), the modeling $T_{e,t}$ distribution (black solid line with circle) and the $T_{e,t}$ measured by the Langmuir probes (black cross). (c) The modeling saturated ion flux (Γ_{ion}) (black solid line with circle) and the Γ_{ion} measured by the Langmuir probes (black cross).

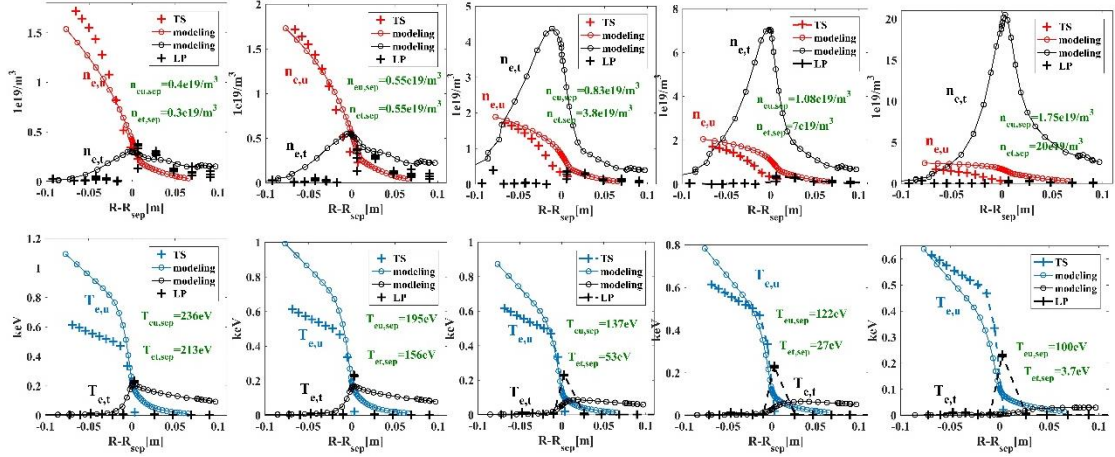


Figure 3.11 Density scanning for the shot #81562, the $n_{e,sep}$ is changed from $0.4 \times 10^{19}/m^3$ to $1.7 \times 10^{19}/m^3$.

It is an unexpected result that the #81565 has the same SOL parameters of #81562, they have the same SOL connection length (165m vs 170m) as shown in figure 3.9(b)&(c). The original view is despite the shot #81565 use the new shape generation method, the shot #81565 still should be similar to the normal USN shot #71022. The modeling results indicate that this new method also enlarges the connection length, and this issue needs more simulation work to deeply study in the future.

The long distance between two null X points in shot #81562 attenuates the optimizing function of the quasi-snowflake configuration to the divertor heat load, makes the QSF configuration more like a single null configuration. In the future experiment, the QSF configuration should shorten the two null X point distance as much as possible.

3.4 Summary

SOLEDGE2D-EIRENE is a typical 2-D fluid modeling code which is used to simulate the edge plasma transport. The penalization technique helps SOLEDGE2D code be suitable for all kinds of tokamak geometry structure. We use the SOLEDGE2D code to model the neon seeding and QSF configuration experiments. The neon seeding increases both the core plasma and divertor region radiation, and

the heat load mitigation effect still have improving space. In the QSF configuration, the second X point is away from the first one, which weakens the heat load mitigation effect.

Chapter 4. The detachment feedback control experiments on EAST

4.1 The EAST detachment feedback control system

EAST develops a comprehensive feedback control system of the radiative divertor to keep stable detachment phase. The construction of EAST radiative divertor feedback control system includes the data processing subsystem, the feedback control algorithms and the data transport network of various diagnostics.

The various diagnostic data are used as the controlled variables in this control system, which includes AXUV^[83], the divertor Langmuir probe (LP)^[84] and the infrared (IR) camera^[85-86], providing the radiated power(P_{rad}), the divertor electron temperature $T_{\text{e,div}}$, the saturated ion flux (Γ_{ion}) and the surface temperature of the divertor target plate(T_{surface}). By establishing a data processing subsystem, the values of various physical quantities required by the control system are obtained.

Control algorithms are implemented into EAST Plasma Control System (PCS)^[87-89]. All the control algorithms are using the proportion-integration-derivative(P-I-D) controller. During the discharge, EAST PCS call the corresponding real-time computing node to execute the control algorithm according to the preset before the discharge. The execution process of the detachment control is injection of the impurity gas or the working gas. The actuators of the control system thus are the gas puffing system on EAST, mainly including the divertor piezo-electrical valves and the supersonic molecule beam injection(SMBI)^[90]. Figure 4.1 is the schematic of the EAST detachment feedback control system.

By applying this control system into EAST long-pulse plasma, we have realized the radiative feedback control for the total radiated power and the local radiated power^[77-78], the divertor particle flux feedback control^[91]. The divertor heat load is reduced by all the above control methods effectively, and also realizing the controllable partial detachment phase.

In this thesis, we focus on the other detachment feedback control methods, the $T_{e,div}$ and the target surface temperature ($T_{surface}$) control method. In the $T_{e,div}$ feedback control, we firstly use the low-Z impurity CD₄ as the working medium to take place the neon and argon which are more widely used on EAST.

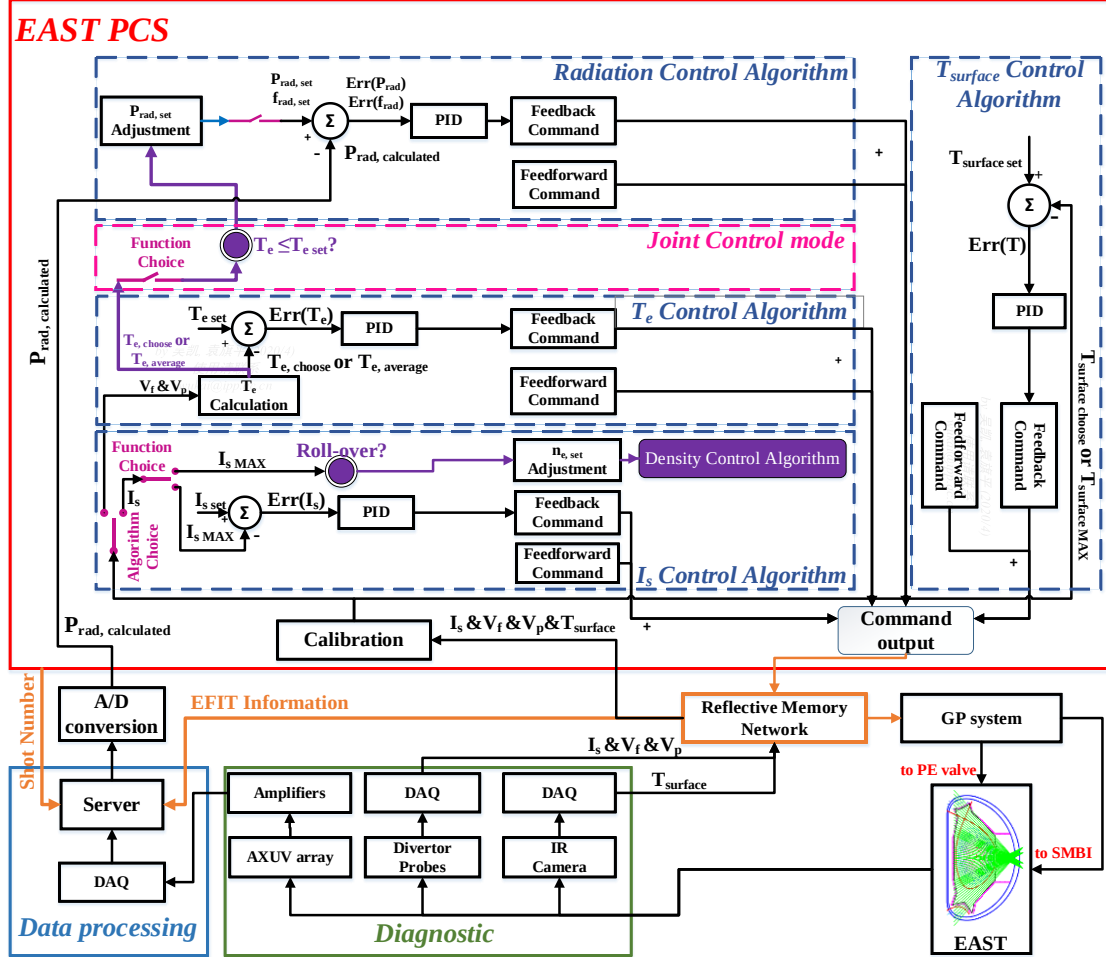


Figure 4.1 The schematic of EAST radiative divertor feedback control system

4.2 Achievement of $T_{e,div}$ feedback control by CD₄ seeding

4.2.1 Feedback control logic

The realization of the control method is also described in the reference^[92]. Figure 4.2 shows the flow chart of the CD₄ $T_{e,div}$ control proceeding. Under $T_{e,div}$ control phase, EAST PCS receives the real-time data of the Langmuir probes(LPs) and then calculates the $T_{e,div}$ in a fast way^[93], figure 4.2 shows the geometric positions of the divertor LPs. After acquiring the real-time $T_{e,div}$ data, the P-I-D controller then

calculates the feedback control commands and sends to the gas puffing valves. The feedforward command setting manually will be sent together with the feedback control commands.

The CD_4 gas can be injected in the pulse length or the pulse amplitude mode. In the pulse length mode, the gas puffing valves work under the fixed aperture. The impurity gas is injected as a series of the independent square pulse. In the pulse amplitude mode, the aperture of the gas puffing valves can be continuously adjusted, so the impurity gas is continuously injected with different flow speed.

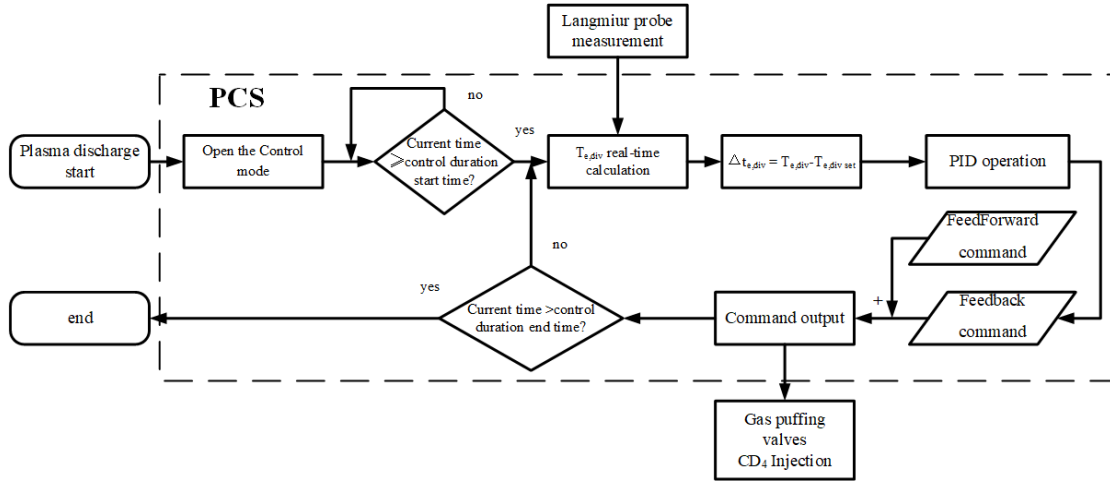


Figure 4.2 The flow chart of the CD_4 $T_{e,div}$ feedback control proceeding.

4.2.2 Experimental setup

The CD_4 $T_{e,div}$ feedback control was operated under H-mode plasma with USN shape. The input power is from 3 sources: the 2.45GHz low hybrid wave(LHW) of about 0.8MW, the 4.6GHz LHW of about 1.6MW and the ECRH of about 0.8MW. The plasma current $I_p=0.55MA$, and the Greenwald density $n_G \approx 8.64 \times 10^{19}/m^3$. The toroidal magnetic field $B_T \approx 2.25T$, the safety factor $q_{95} \approx 4.8$, and the parallel SOL flow direction is upwards ($B \times \nabla B \downarrow$).

The upper outer(UO) divertor gas puffing valve executes the impurity seeding in the pulse length modulation mode: fixing the valve working voltage and adjusting the pulse length automatically. In EAST PCS, the gas puffing command is executed one time in each working duration. The working duration time is set to the fixed value, and the pulse length, i.e., the duty cycle in one working duration can be adjusted by

the PID controller automatically.

In this experiment, the control duration has two kinds: 0.2s and 0.4s. In such setting mode, if the gas puffing pulse length is too short or even zero, the feedback control will enter the smaller working duration to reduce the empty time. When the pulse length is from 0.02 to 0.18s, the control duration is 0.2s, i.e., the off-time is from 0.18s to 0.02s. When the pulse length is from 0.2s to 0.38s, the control duration is 0.4s, i.e., the off-time is from 0.2s to 0.02s, so in both working durations, the off-time is almost same. The injecting position is away from the UO divertor corner about 12cm as shown in the figure 4.3(b).

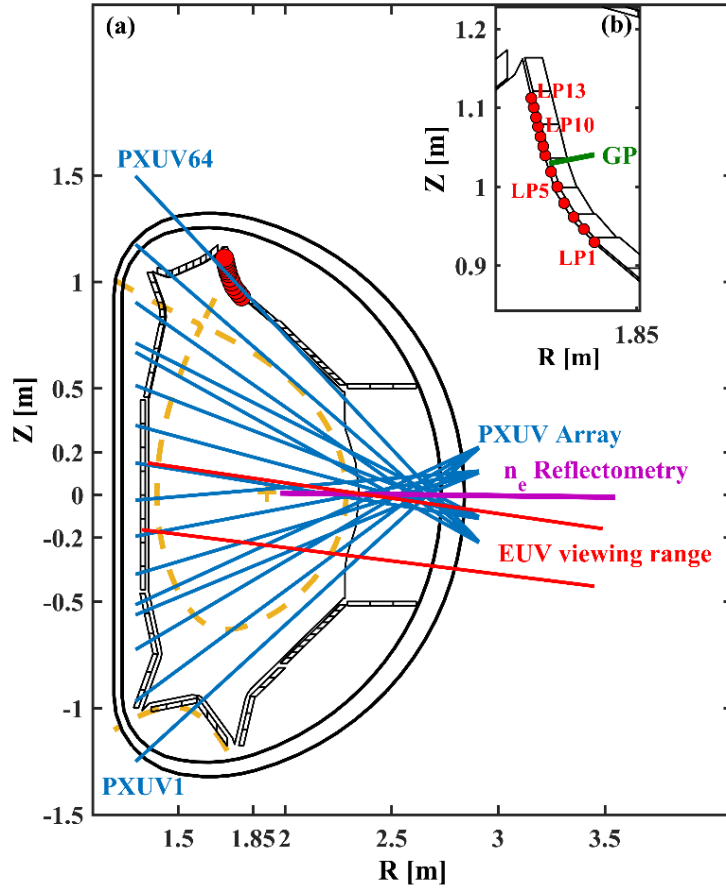


Figure 4.3 The main diagnostics in CD_4 $T_{e, \text{div}}$ feedback control experiments. (a) Including the AXUV array (blue lines, total 64 channels, not shown completely), the view line of the reflectometry (purple line), the EUV viewing range (between red lines); (b) The positions of the UO divertor Langmuir Probes (red spots) and Gas puffing position (green line).

Except the above mentioned AXUV, LP and IR camera, the following are the other main diagnostics used in the experiments. The edge plasma density profile at the midplane is detected by the EAST reflectometer^[94]. The covering frequency range is 32-110GHz, and the sample frequency is 64MHz. A fast-time-response EUV spectrometer working at 20-500Å are used for observation of line emissions from impurity ions, e.g. CV (C4+) at 40.27Å, CVI (C5+) at 33.73Å, and the tungsten unresolved transition array (W-UTA) at W²⁷⁺-W⁴⁵⁺ in 45-70Å^[95-97]. Figure 4.3(a) shows all the above diagnostics.

In the EAST radiative divertor experiments, the neon (Ne) and argon (Ar) are widely used^[77,98]. The CD₄ is firstly used as the working medium in the control system. CD₄ is one of the typical low-Z impurity gas, the cooling factor peak of the Carbon (C) is located at electron temperature $T_e \sim 8\text{eV}$ ^[38]. The C atoms are completely stripped in the SOL, which means the radiated process are mainly in the SOL. The N₂ is the another widely used impurity gas and beneficial to reduce the divertor heat flux^[55,61], and its cooling factor is located at $T_e \sim 10\text{eV}$ ^[38]. However, the Li coating on the first wall^[99] limits the N₂ application on EAST. The N₂ will erode the Li and generate the unstable compound Li₃N, and the secondary product NH₃ will also erode the metal wall. Since the cooling factor peak value of CD₄ is at the same T_e range with that of N₂, and their peak value are similar^[38]. In this regard, CD₄ is one of the best replacements of the N₂ on EAST.

4.2.3 Control experiment results

The CD₄ $T_{e, \text{div}}$ feedback control experiments yielded the expected result. The set level of the UO $T_{e, \text{div}}$ is 5eV, the plasma will be in partial detach phase in such level. The chosen LP probe signal is close to the UO strike point (SP). The control period started from 4s with the feedforward CD₄ pulse length of 0.5s. Figure 4.4 shows the control results in EAST shot #93805 and #93808. The interval between feedforward and feedback process is about 0.2s. Shot #93808 has a higher CD₄ injection amount because during the control phase, the total pulse length of CD₄ injection of shot #93808 (3.573s) is much higher than #93805 (2.727s). With the higher injection

volume of CD_4 , the waveform of the actual UO $T_{e, \text{div}}$ in the shot #93808 is more flat, the absolute error is smaller, which shows a more stable control effect. The integration gain G_I in #93808 is adjusted much higher than that in #93805 because the accumulate error in #93805 is big. Lifting the G_I extends the single pulse length to lower the $T_{e, \text{div}}$ to 5eV, and thus in #93808 the feedback duration shows a fixed duty cycle CD_4 seeding. During the feedback control, the gas puffing valve always executed the highest pulse length command (0.38s, shown in figure 4.4(b)).

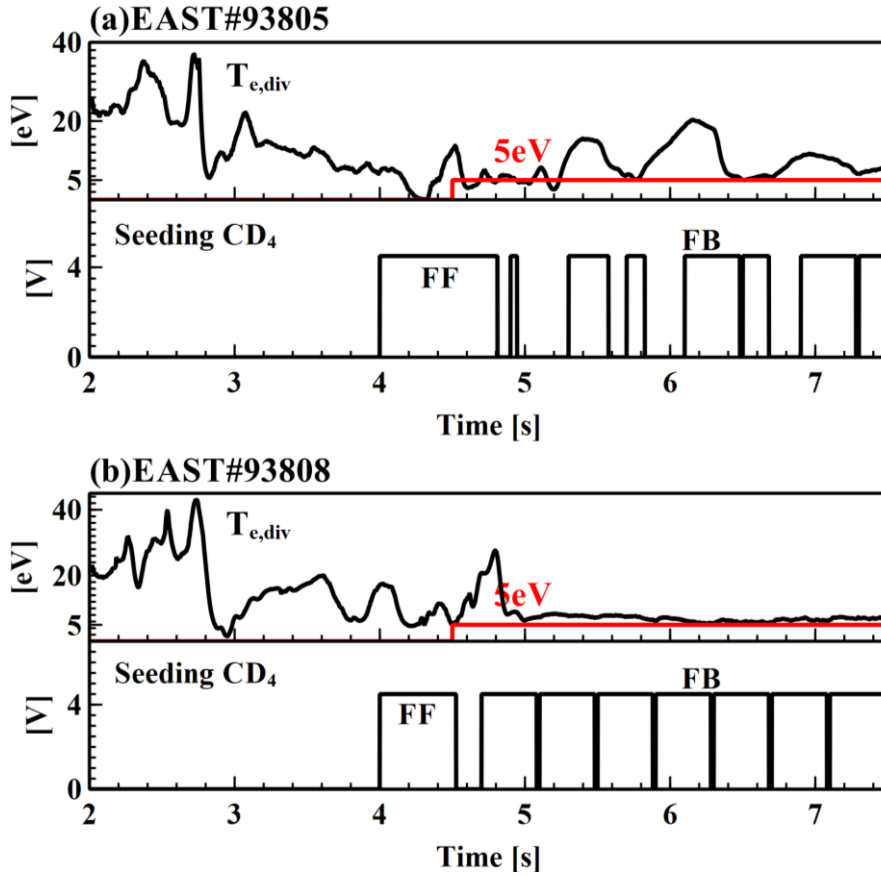


Figure 4.4 The CD_4 $T_{e, \text{div}}$ feedback control results in EAST shot#93805 and #93808. The $T_{e, \text{div}}$ target value is 5eV (red line). The control period started at 4.0s with the feedforward (FF) injection. After the feedforward injection, the CD_4 injections were operated by feedback (FB) control. The y-axis of the CD_4 waveform is the working voltage of the gas puffing valve, unit: V.

In the shot #93808, after the feedforward CD_4 seeding, the $T_{e, \text{div}}$ has a reduction of about 12eV with a small oscillation at about 4.4s. At the end of the feedforward

process, the $T_{e, \text{div}}$ rebounded to the peak value of about 27eV at 4.8s. The feedback period started at 4.7s and restarted the CD₄ injecting with the feedback control. After 4.8s, the $T_{e, \text{div}}$ was lowered close to 5eV and kept until the end of the control period.

Comparing to the Ne and Ar experiments^[77-78,98], in the pulse width control mode, both the injecting volume of single CD₄ pulse and the total CD₄ injecting volume during the control duration are far beyond than those of Ne or Ar. The CD₄ brought a lot of D atoms into plasma, the main plasma density n_e thus was lifted during the control phase. Figure 4.5(b) shows the time evolution of the Greenwald fraction $f_G = \frac{n_e}{n_G}$. At the control phase beginning, the f_G is about 0.4. At the end of feedback control, it was already up to 0.7, the actual n_e is about $6.04 \times 10^{19} \text{m}^{-3}$. The increment of the upstream plasma density influences the SOL recycling state, the $T_{e, \text{div}}$ and the divertor electron density $n_{e, \text{div}}$ thus are changed in the control period. Section 4.2.4 will analyze it. Considering about the big injecting amount, the seeding efficiency was may degraded in the second half of the feedback control duration and lowering the power exhaust effect, that is why at about 6.4s, the actual $T_{e, \text{div}}$ was in the smallest value of about 5.3eV, and gradually lifted in a small step shown in figure 4.4(b).

Figure 4.5(a) shows the time evolution of the divertor surface temperature measured by the IR camera. It is obvious that the T_{surface} has a reduction of about 150°C during the control period and the tendency is synchronized to the change of the $T_{e, \text{div}}$, which is also an evidence for the control effect of CD₄. For the Ne and Ar, the power dispersal effect still exists in a time after the seeding processes, while for CD₄ this is not observed. The time evolution of $T_{e, \text{div}}$ and T_{surface} in figure 4.4(b) & 4.5(a) suggests that the lasting effect of CD₄ is not strong. Once the injection pauses, the divertor temperature rises soon. The radiated power shown in figure 4.5(b) has two peaks in the control phase, but at the flat segment in the feedback period (about 5.5s to 7.5s), the P_{rad} is only about 0.3MW (lower than 20% of the input power) and has almost no change, the single CD₄ effect on the radiation actually is not strong even with such a higher amount compared to the Ne and Ar. The big injection volume of CD₄ also made a degradation on the plasma confinement. During the control phase,

the plasma stored energy had a reduction of about 30kJ, which is much higher than in the Ne experiments^[77]. It is necessary to find an effective way to keep the good plasma confinement when implementing the CD₄ T_{e,div} feedback control.

The CD₄ injection also optimized the LHCD coupling in a few degree, which was also found in JET CD₄ seeding experiments^[100]. Figure 4.5(c) shows that the reflected energy of 4.6GHz LHW was reduced, while the incident energy was kept a constant. The reduction of reflected energy means more LHW energy were absorbed by the main plasma.

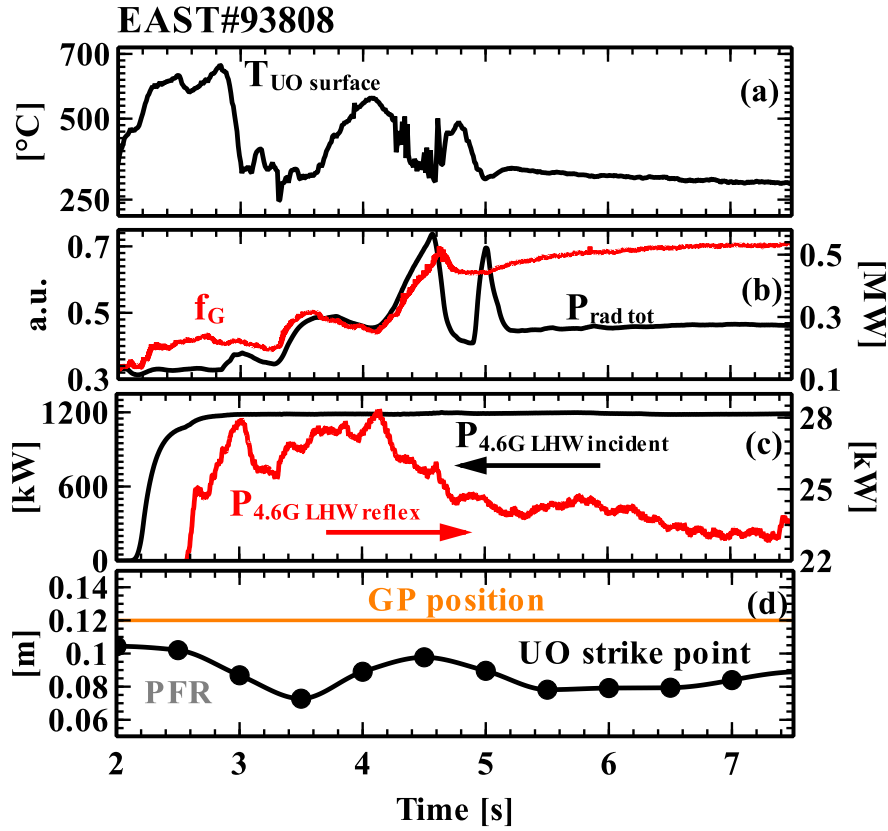


Figure 4.5 The control period in shot#93808. (a) The UO divertor target plate surface temperature measured by the IR camera; (b) The time evolution of the Greenwald fraction (red line) and the total radiated power(black line); (c) The incident power of the 4.6GHz LHW (black line) and the reflex power of the 4.6GHz LHW (red line); (d) The time evolution of the UO strike point position (line with spots) and the gas puffing position (orange line).

4.2.4 The change of SOL recycling and WUTA intensity

In theory, the SOL recycling state are decided by the SOL thermal convection and conduction processes^[20].

In the CD4 $T_{e,div}$ feedback control experiment, the time evolution of the upstream and divertor plasma density profile are shown in figure 4.6. Choosing 6 time points: 4s (Feedforward start), 4.5s(in the feedforward phase), 4.6s(the interval between feedforward and feedback period) and 3 time points in the feedback phase (5.5s, 5.8s and 6.25s). The feedforward with high injecting volume lifted the electron density of the upstream plasma (i.e. the main plasma) n_{eu} profile obviously compared to figure 4.6(a1) & 5(b1), the upstream electron density inside the separatrix has an appreciable increment after CD4 seeding, so the n_{eu} profile near the separatrix has an obvious change, but the change of the upstream plasma density at the separatrix $n_{eu,sep}$ can be neglected, which was around $1e19/m^3$. In the meantime, the divertor plasma density at the strike point $n_{et,sp}$ was increased in a factor of about 2. At 4.5s, the $n_{et,sp}$ is about $4.2e19m^{-3}$. This increasing tendency was terminated with the end of the feedforward impurity pulse.

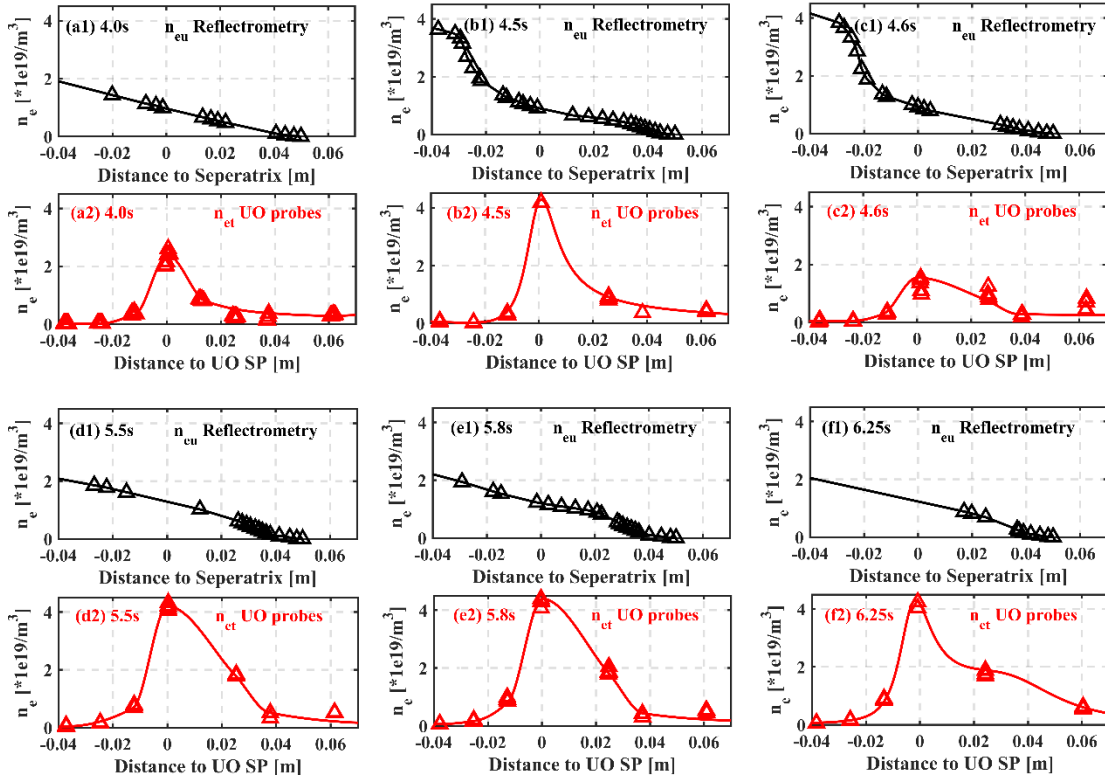


Figure 4.6 The upstream plasma density $n_{e,u}$ (red triangles) and the divertor plasma

density $n_{e,t}$ (black triangles) profiles near the separatrix and the upper outer strike point (UO SP) at different time points and their fitting curves. (a1)&(a2): 4s (Feedforward start), (b1)&(b2): 4.5s(in the feedforward phase), (c1)&(c2): 4.6s(the interval between feedforward and feedback period) and 3 time points in the feedback phase: 5.5s (d1)&(d2), 5.8s (e1)&(e2) and 6.25s(f1)&(f2).

Comparing to the figure 4.6 (b1), (b2), (c1) & (c2), at the interval of the feedforward and feedback phase (4.6s), without the CD_4 injection, the $n_{et, sp}$ dropped substantially in 0.1s. In the feedback control phase, the divertor plasma density n_{et} rose again and the $n_{et,sp}$ was almost kept a constant of about $4.3e19m^{-3}$, as shown in figure 4.6 (d2), (e2)&(f2), which are higher than the $n_{eu,sep}$ at the same time as shown in figure 4.6 (d1),(e1)&(f1). From figure 4.6(c) to figure 4.6(d), the feedback control phase changed the SOL from low recycling state (at 4.6s) to the high recycling state (at 5.5s), and kept this state until the end of the feedback control phase (figure 4.6(e) and figure 4.6(f)).

The symbol of the high recycling SOL is the influence to the $n_{e, div}$ and $T_{e, div}$ from n_{eu} getting stronger as stated by equation (2.43) & (2.44). The evolution in figure 4.6 indicates that the CD_4 injection change the SOL from low recycling state at 4.0s into high recycling state during the feedforward injection phase. The SOL rebounded back to low recycling state after the feedforward phase and the feedback control phase made it into high recycling again until the end of control period.

Figure 4.7(a) and 4.7(b) shows the time evolution of $T_{e, div}$ distribution and the saturated ion flux j_{sat} on the UO divertor target at the same time points in figure 4.6. It should be mentioned that figure 4.5(d) shows the movement of the outer strike point(SP) during the control phase, the $T_{e, div}$ signal used in the feedback control is not at the actual SP, so the $T_{e, div}$ at the SP is actually higher than the controlled value in figure 3(b). It is obvious that the change of $T_{e, div}$ distribution was synchronized to the control process, i.e., synchronized to the CD_4 pulse injections. At 4.6s, the increment of the $T_{e, div}$ proved that the SOL returned to low recycling state, also demonstrated that the lasting effect of the CD_4 injection is limited. Once the injection stops, the

$T_{e,div}$ rebounds quickly. The j_{sat} shows a general decline during the feedforward and feedback control phase, which is consistent with the evolutions of the $T_{e,div}$ shown in figure 4.7(a).

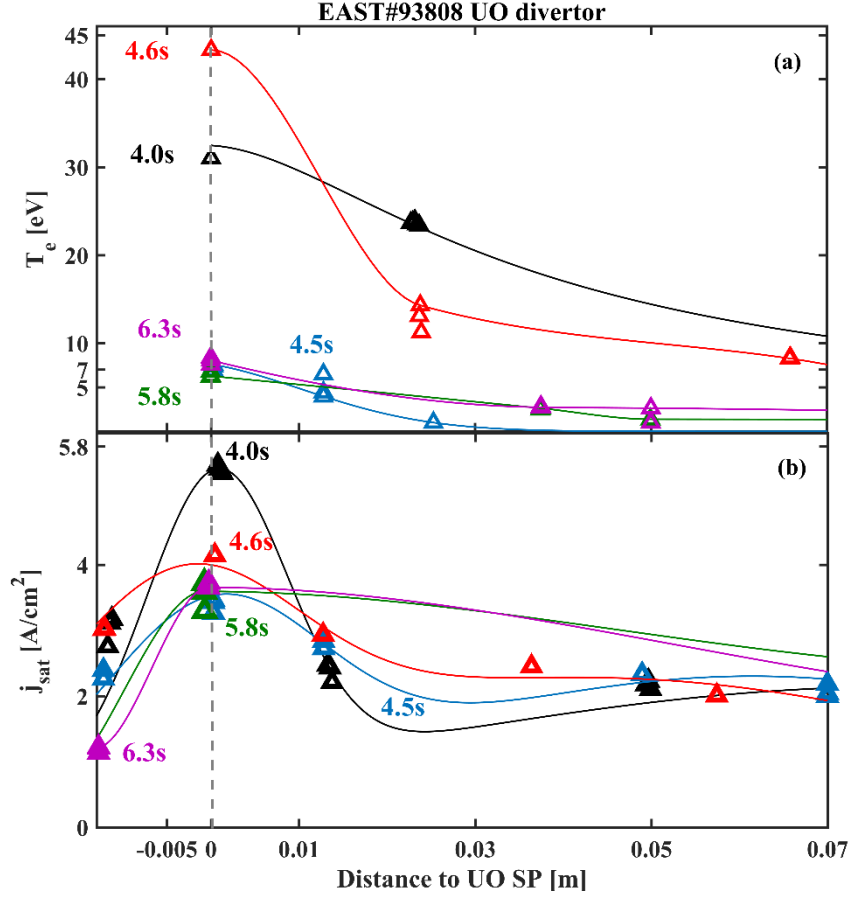


Figure 4.7 (a)The divertor electron temperature $T_{e,div}$ profile outside the strike point on the UO divertor measured by the Langmuir Probes and the relative fitting curves at the different time points. (b)The divertor saturated ion flow j_{sat} profile outside the strike point on the UO divertor measured by the Langmuir Probes and the relative fitting curves at the different time points. The time points in figure 6(a)&6(b) are consistent with the time points in figure 4.6.

Figure 4.7(a) also proves that both in feedforward and feedback control phase, the $T_{e,div}$ were kept below 8eV, the plasma was into partial detachment phase. It should be mentioned that the essential proof of the detachment phase is the momentum loss in the downstream SOL. For now, we still have no a proper diagnostic system to measure or estimate the momentum loss for EAST SOL. The

primary goal is building a useable feedback control system to enter the detachment phase actively, so $T_{e, \text{div}}$ is an appropriate controlled variable to judge whether the plasma is detached.

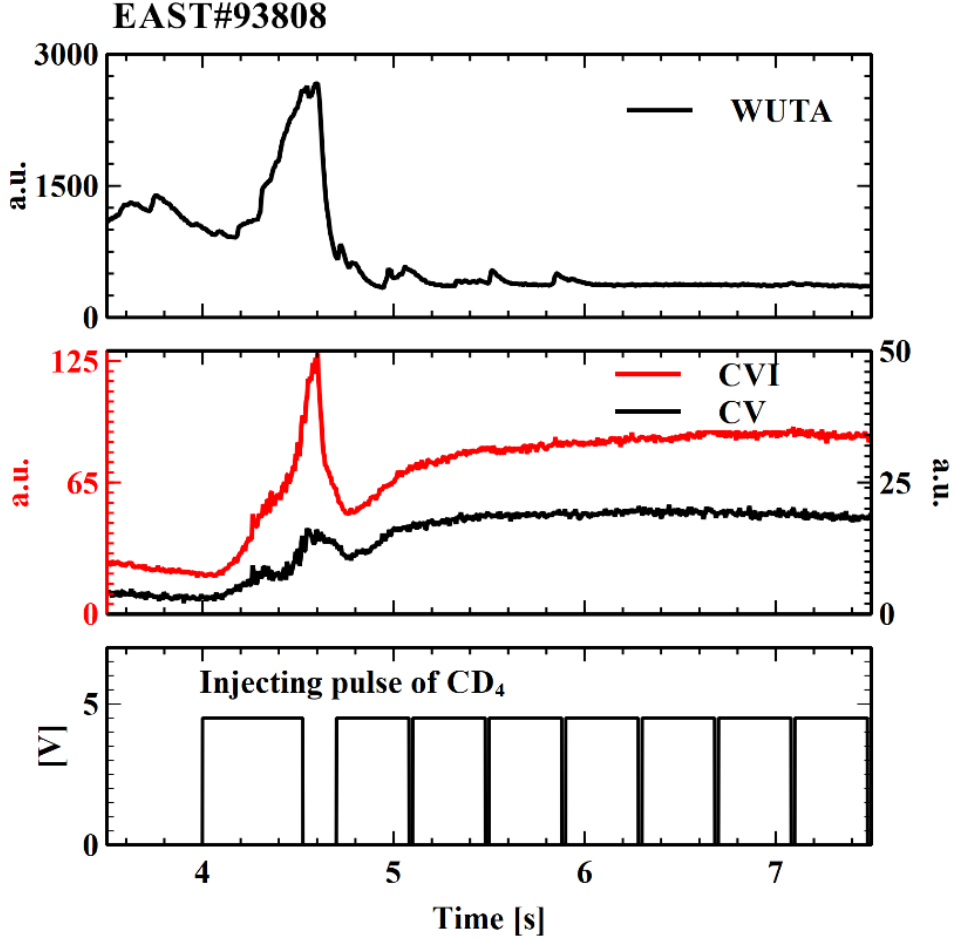


Figure 4.8 The time evolutions of the W-UTA, CVI and CV line emissions and the UO $T_{e, \text{div}}$ during the control period. The y-axis of the CD₄ waveform is the working voltage of the gas puffing valve, unit: V.

Figure 4.8 shows the time evolutions of the W-UTA intensity and carbon line emissions (CV, CVI). The intensity of the CVI (C5+ at 33.73Å) is much higher than that of the CV (C4+ at 40.27Å) during the control period. CVI represents the higher ionized state of carbon than CV. The intensity of CVI is far beyond that of CV indicated that more carbon particles are stripped to the high ionized state. Considering that the carbon radiation mainly happens in the SOL, so most of the C atoms were in the high ionized state in the SOL. The W-UTA has a different tendency. At the control beginning, the W-UTA bursts a lot and was up to the peak value at 4.6s. During the

feedback control phase, it has a monotonous reduction and kept a low level until the end of the control period. Whether the feedforward CD₄ injection caused the increment of the W-UTA starting at about 4.2s still needs to be proven by more experimental results, but the control phase lowered the $T_{e,div}$, which is beneficial to reduce the W sputtering.

4.3 $T_{surface}$ feedback control

4.3.1 Control system building

The divertor surface temperature is measured by the IR camera. EAST develops the infrared camera system with large viewing range and high spatial resolution. The monitoring area includes the first wall and the divertor target plate. The detecting temperature range is about 20~600°C^[85-86]. Compare to the $T_{e,div}$ and radiation feedback control system, the data proceeding to the infrared camera is the easiest. The EAST PCS receives the surface temperature data of different positions, which cover the whole upper and outer target plate. Different to the electron temperature, the surface temperature measured by IR camera directly represents the “hot” and “cold” state of the target, which represents the cumulative effect of the parallel heat flux over the discharge time, and its reduction indicates that the heat load of the target plate is effectively relieved.

The basic control logic is same to the $T_{e,div}$ one in the section 4.2.1, the diagnostic data is changed to the surface temperature.

4.3.2 The control results

The feedback control of the surface temperature of the divertor target plate also achieved a good control effect. Figure. 4.9 shows the control experiment results of #93411 and #94003. The infrared signal channel selected for these two experiments is based on the results of the pre-experiment, which is most close to the strike point. The experiment was performed by mixing argon-deuterium and neon-deuterium gases respectively. The upper outer divertor valve was used as feedback control actuator to inject the impurity gas. The experimental results show that the divertor surface

temperature drops to near the set target value and maintains until the end of the control cycle.

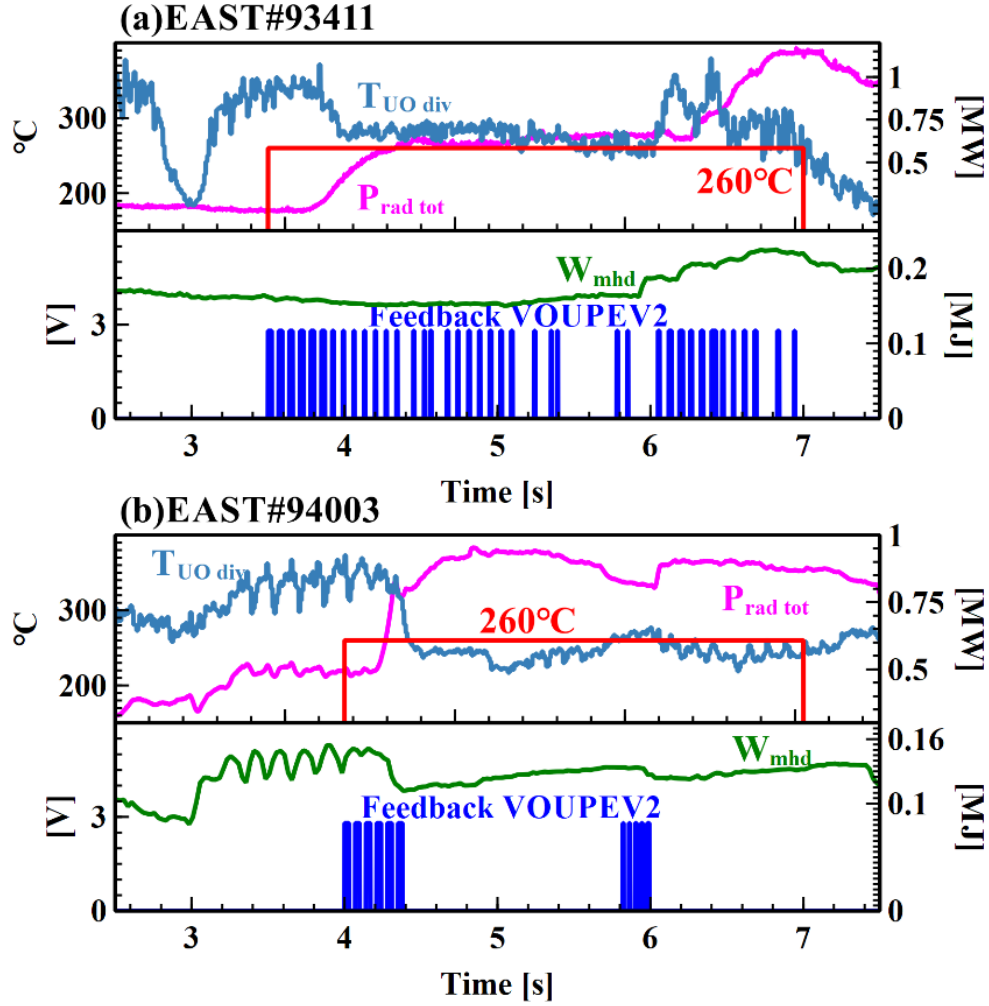


Figure 4.9 The T_{surface} feedback control results in shot #93411 and #94003.

At the beginning of the control cycle, the plasma storage energy decreases due to the large amount of impurity injection, but gradually recovers. In #93411, the plasma input power was increased in the second half of the control cycle (after 6.0s), resulting in a significant increase in plasma energy storage, radiation power and target surface temperature at about 6.4s. The feedback control system still reduces the surface temperature to the set value at the end of the control cycle, which proves that the control system has a reliable control ability in mitigating the divertor heat load, even when the input power increases suddenly.

It is the first time to implement the target surface temperature feedback control on EAST, so there is still a large space for improvement in the optimization of the

control parameters and control accuracy.

4.4 Summary

The EAST radiative divertor feedback control now owns multiple control functions to lower the divertor heat load.

By injecting impurity gas, the electronic temperature of the target plate can be greatly reduced according to the set target value, so that the plasma can enter the partial detachment state, and the heat load of the divertor thus can be significantly reduced.

It is the first time to use the low-Z impurity gas in the $T_{e,div}$ feedback control and get an expected result on EAST. The $T_{e,div}$ can be controlled close to 5eV by injecting a giant amount of the CD₄. The CD₄ injection caused two physical processes in the SOL: the radiated power exhaust and the lifting of the upstream plasma density. The $T_{e,div}$ and $T_{surface,div}$ are both reduced in the feedback control period, which demonstrated that the CD₄ injection is effective to mitigate the power load on the divertor target. The CD₄ injection brings a lot of D atoms and thus changes the SOL recycling state. The PID parameters and set level of the $T_{e,div}$ still need to optimize to get a better control result.

The CD₄ shows a completely different characters on the radiative divertor compared with neon and argon. The latter two impurities main generate the high radiation power loss, while the increment of the radiated power from CD₄ actually is not so strong. The big change of the plasma density also lead the negative influence on the plasma confinement in the meantime.

The method of direct feedback control of saturated ion flow and the feedback control system based on the diagnostic data of infrared camera also achieved good control effect, and the heat load of the divertor target decreased significantly.

Chapter 5 Summary and outlook

5.1 Summary

One of the most challenging problems in the development of modern tokamak and future fusion reactors is the long-term excessive heat load of the first wall materials, especially the divertor target plate. In order to solve this issue, we need study the SOL transport, to understand the physical mechanism and build the control method based on the physical principle.

The “two-point” model simplify the real SOL structure, help us to master the transport mechanism from the core plasma to the divertor target plate. This theoretical analysis provides us two main ways to mitigate the divertor heat load: increase the radiation power loss to reduce the divertor heat flux, or lift the plasma density to change the SOL recycling state. The final aim of the both technical ways is to realize the stable complete/partial detachment in the long-term high-performance plasma discharge.

Most of the 2-D edge plasma transport modeling codes are based on solving Braginskii equations with the Bohm boundary conditions, and SOLEDGE2D-EIRENE is a typical one. We firstly use this code to analyze the radiative divertor and the QSF experiments on EAST tokamak. Under the current plasma parameters, neon particles increase the radiation of both the core plasma and the divertor region, and the core plasma radiation takes the lead. Meanwhile, the divertor heat load mitigation from the impurity seeding still have a big improving space. The EAST QSF configuration is a snowflake-like divertor shape whose aim is also to reduce the divertor heat load and easy to produce the detachment plasma. The modeling result shows that the second X point of QSF discharge is too far to increase the flux expansion on the divertor like a normal snowflake configuration. Thus, for the QSF configuration, the most important optimization is to reduce the distance between the two X points

The radiative divertor principle has been implemented on many tokamaks to achieve the detached plasma. EAST also building the comprehensive feedback control system. We used the CD4 gas as the working medium of the $T_{e,div}$ feedback control system, to lower the divertor electron temperature to 5eV, which is reaching the detachment condition. Unlike the medium-Z impurity main increase the power loss, the CD4 also changes the SOL recycling state because bring a lot of deuterium particles. The divertor surface temperature feedback control was also realized and showed a powerful control capability.

5.2 Outlook

The SOLEDGE2D-EIRENE will be used more often on EAST tokamak. Besides the analysis of the experiments. Many new physical phenomena have also been found in the feedback control experiment of radiative divertor. By running SOLEDGE2D-EIRENE, the experimental data can be fitted and analyzed to understand the transport process of impurity particles in the feedback control, and experimental methods also can be improved according to the modeling results, and improve the effectiveness of the control system.

SOLEDGE2D-EIRENE code will also be upgraded and modified to strengthen the modeling capability, like higher simulation efficiency, adding more simulation features and be suitable for more modeling tasks.

EAST radiative divertor feedback control system still has great room for improvement and upgrade. With the EAST lower divertor modification, new divertor puffing system and diagnostic data will be added to the control system. A new development direction is developing the multiple variable control. Using several physical diagnostic data as the controlled variables together in the control system, the judgment to the divertor heat load and the detachment state will be more precise, which greatly improve the stability of the control system. Another direction is to develop the radiation profile control capability, to improve the radiation ratio in SOL and the divertor region, so that the radiation process mainly occurs in the plasma boundary region, reduce the influence of impurity injection on the main plasma.

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