Overview of the KSTAR Research in Support of ITER and DEMO

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Abstract. Korea Superconducting Tokamak Advanced Research (KSTAR) program is strongly focused on solving the scientific and technological issues in steady-state high performance plasma operation in preparation for ITER operation as well as DEMO design basis. In this regards, KSTAR has made significant advances in developing long pulse and high performance plasma scenarios utilizing the advantage of fully superconducting tokamak. Nine-year of operation showed that KSTAR is the best engineered superconducting tokamak having extremely low intrinsic error field and reliable operation without any serious failure. According to the effort in plasma control, H-mode plasma discharge in KSTAR has extended up to 1 MA in plasma current and up to 55sec in flat top duration at 0.6 MA. The high normalized beta scenario (β_N >3) surpassing no-wall limit was sustained up to 3sec without any external field correction. A good candidate for advanced steady-state operation has been developed with high poloidal beta (β_N >3) and fully non-inductive current drive at 0.4 MA. However, the sustained pulse length is limited by 12secdue to excessive heat-load on the ploidal limiters. The analysis and the results of fast ion loss showed good agreement in increased orbit loss of NBI fast ions at low Ip operation. In addition, several advanced scenarios have been explored for the study in transport and MHD instability under extreme conditions; i) Hybrid scenario with large fusion gain (G ~ 0.38) close to ITER baseline (G=0.4), ii) internal transport barrier (ITB) scenario sustaining 3.6sec stably, and iii) extremely low q95 (q95 ~ 2.25). Due to adoption of tungsten divertor, ITER requires robust and fully suppressed edge localized mode (ELM)-crash. In 2016, very robust and reliable ELM-crash suppression at n=1 RMP has been achieved and ELM-crashed status sustained over 10 sec. The L-H-mode transition threshold power depends on applied error field. Plasma surface interaction and high heat flux effect on tungsten divertor has been investigated very effectively by installing tungsten castellation tile with difference shape and elevation. In near terrn, improved plasma discharge are expected with upgraded heating systems including NBI-2 system (~6 MW, available from 2018), active water-cooling on plasma facing components and divertor. After several years of operation under carbon wall, KSTAR will be upgraded in around 2021 for the DEMO relevant research in plasma discharge and in technical aspects also. Especially metal divertor, advanced current drive like HFS LHCD, Helicon CD.

1. Introduction

As magnetic confinement fusion move to reactor scale, development of the steady-state high performance plasma scenarios that are applicable to ITER and DEMO and verifying those scenarios in the present superconducting tokamak are very essential. Korea Superconducting Tokamak Advanced Research (KSTAR) program is strongly focused on solving the scientific and technological issues in steady-state high performance plasma operation in preparation for ITER operation as well as DEMO design basis [1-3]. In this regards,

The key features of KSTAR, which enable exploring the steady-state high performance plasma are as follows;

- Superconducting magnet system : same superconductor and operated very reliable without any failure. an extremely low intrinsic error field ($\delta B/B_0 \sim 10^{-5}$) [4], as well as with unprecedentedly low level of field ripples (δ_{TF}=0.05%) [5].
- In-vessel control coils : The KSTAR has three rows (Top/Middle/Bottom) of versatile in-vessel control coils (IVCC) (capable of up to n=2) in low field side, whose configuration is similar to that of the planned ITER RMP coils (capable of up to n=4). Unlike other conventional tokamaks equipped with two (Top/Bottom) rows of RMP coils, the KSTAR is the only major tokamak that has in-vessel mid-plane RMP coils. Taking advantage of broadband switching power amplifiers (SPA) capable of up to 5 kA/turn in DC to 1 kHz, various 3-D configurations with n=1 and/or n=2 fields can be applied in conjunction with three dozens of assorted patch panels in KSTAR.
- Heating and current driving system : The major heating and current drive power in KSTAR is come from NBI and ECH. One tangential neutral beam system having three ion sources provides up to 5.5 MW with deuterium beam energy of 100 keV in co-current direction [6]. Another neutral beam system which is capable to deliver 6 MW, 100 keV deuterium neutral beam in co-current direction during 300 sec is being constructed targeting operation from 2018. The three new beams will be arranged vertically with tangency radius of 1.56 m to provide balanced core heating and off-axis current. Until 2015, ECH systems having frequencies of 110 and 170 GHz had operational, From 2016, single dual frequency system (105/140 GHz) replaced old systems. The new dual frequency gyrotron and antenna system provides 1 MW during 300 sec.
- Diagnostic system: Besides, along with an upgraded magnetic diagnostic (i.e. magnetic probes installed on the passive plates), the plasma response for 3-D field physics study is directly measured by the state-of-the-art diagnostics, such as electron cyclotron emission imaging (ECEI) and microwave interferometry (MIR). Improvement basic profile diagnostics such as Thomson scattering, ECE. New commissioning a motional stark effect (MSE) diagnostics

In this regards, KSTAR has made significant advances in developing long pulse and high performance plasma scenarios utilizing the advantage of fully superconducting tokamak. Firstly, according to the improved plasma control, extension of H-mode plasma discharge up to 1 MA operation and flattop extension up to 55s at 0.6 MA (see section 2). Secondly, development of advanced high performance scenarios such as stationary high normalized beta (β_N >3), high normalized ($\beta_P \sim 3.0$), *and* low q_{95}, ($q_{95} \sim 2.25$) (see section 3). At third, research on the plasma response using 3D in-vessel control and very stable ELM-crash suppression was achieved and power threshold dependence on field perturbation, (see section 4). Finally, research on tungsten PFC against high heat flux (see section 5)

2. Plasma discharge control improvements for long pulse operation

2.1. Improvement of plasma startup and MIMO control in superconducting tokamak

In future devices such as ITER and DEMO, various constrains given on super-conducting (SC) coils in addition to their large inductances make the conventional inductive plasma startup hardly applicable. To improve the reliability and robustness of plasma startup under these constraints, the ECH assisted plasma startup scenario has been investigated in many devices. Recently in KSTAR, an innovative plasma startup scenario, called as Trapped Particle Configuration (TPC) scenario [7], has been applied and investigated.

Stable and robust plasma position and shape control is a fundamental but essential element for tokamak plasma operations and advanced physics studies. In this regard, KSTAR has some constraints such as a limited number of plasma shaping coils and their long distances to plasma (i.e. poor proximity of coils). Therefore, the shape responses to control coil currents are strongly coupled each other. To resolve this control issue, decoupled multi-input, multi-output (MIMO) shape controller has been developed. Consequently, the obtained MIMO isoflux shape controllers have shown a great improvement of plasma control and stability.

2.2. Improvement of vertical /radial stability control and achieving stable discharge in large plasma current (over 1 MA)

In recent years, the base-level controllability of real-time feedback plasma discharge control has been established and experimentally verified including optimization of fast vertical and radial position control [8]. Also, various machine protection system and algorithms have been successfully implemented and tested. Particularly, a forced landing control scheme, which has

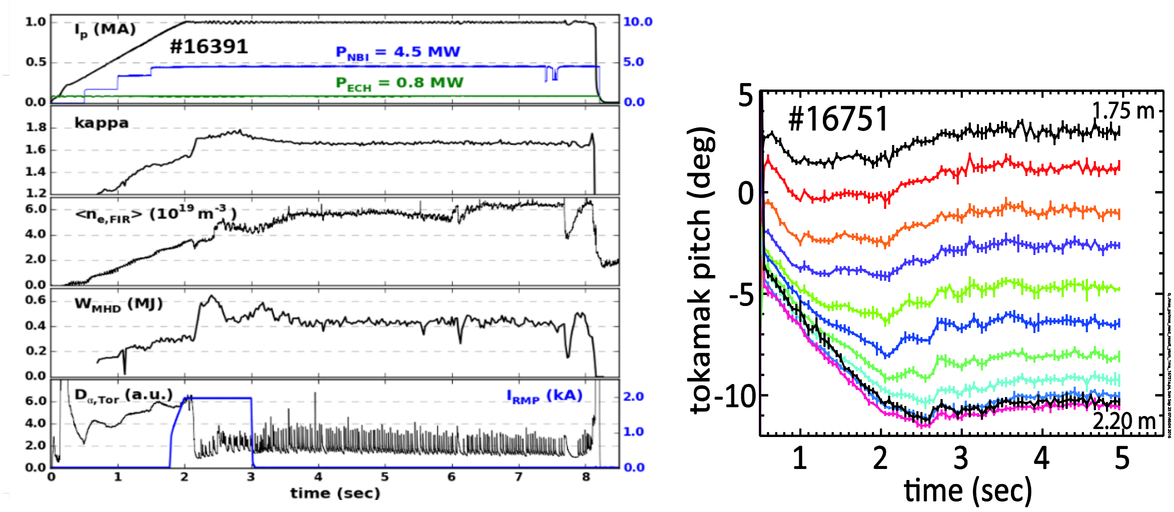

FIG. 1. The representative 1.0 MA h-mode discharge (a) and MSE pitch angle evolutions during the 1-MA plasma current ramp-up showing the on-axis current increasing due to neutral beam during current ramp (b).

been designed to force a plasma discharge landed down when an off-normal event is occurred, greatly reduces the possibility of major disruptions. Thanks to this improvement, the development of Mega-Ampere (MA) operation scenario has been able to be conducted much safely than before. Recently we have also succeeded in 1.0MA h-mode discharge, where 4.5MW NBI and 0.8MW ECH heating were used at B_T=2.5T. Recently Motional Stark Effect (MSE) diagnostics has been installed and commissioned for the real-time control of q profile [9]. Figure 2 (b) shows the time evolution of the pitch angles in MSE signals during the 1-MA ramp-up phase. The on-axis current drive by the neutral beams is clearly observed from the sudden change of the pitch angle evolution in the core channels at 1 and 1.5 sec.

2.3. **Extending the H-mode into long pulse and steady-state over 50s**

A long pulse plasma operation with high performance is a key goal of KSTAR as a super-conducting tokamak device. Since the base-level controllability of real-time feedback plasma discharge control has been obtained in 2014, a large effort has been made on this long pulse discharge achievement. In the first phase (2014), a stable 0.5MA h-mode plasma (#11664) was well controlled and sustained in a long pulse. In the second phase (2015), the long pulse operation was further extended even with higher plasma currents (0.6MA in #14326) as shown in figure 2. In 2016, the attempt was dedicated on utilizing the fully non-inductive high betap scenario in to a long pulse operation. Various attempts were made to reduce the PFC heat-up, such as in-and-out sweeping of striking point and increasing density to reduce fast ion losses etc. As a result, one successful long pulse h-mode discharge with low I_P=0.4MA (#17078). To achieve steady state long pulse operation, in principle the plasma performance should be improved as like the fully non-inductive high betap discharge. However, also there will appear various issues linked to engineering problems such as the stable operation of heating system in a long pulse, the particle and heat fluxes into PFC and long pulse operation in the control coil system.

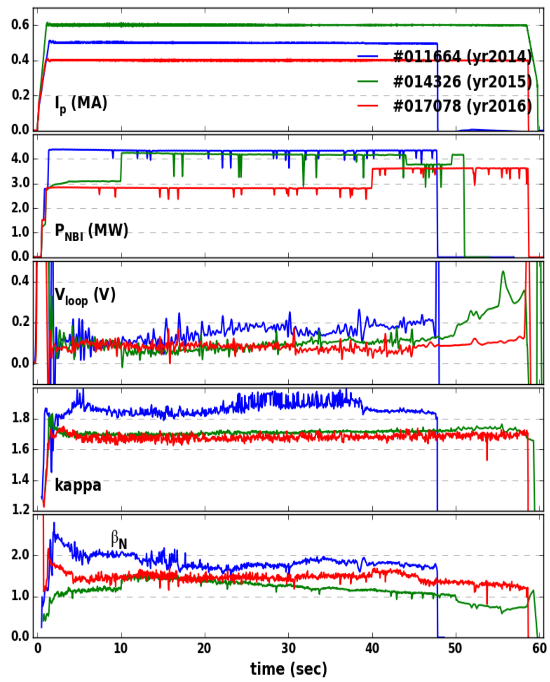

FIG. 2. Yearly achieved long pulse discharges of KSTAR

The ITER-similar shape (ISS) development has been initialized to investigate the feasibility of planned RMP-ELM suppression in a scaled shape and under basic parameters that can be more easily interpreted for ITER. In order to make eligible discharges, a genuine lower single null (LSN) shape, a variety of control implementations was done and verified both under the control simulator environment [10] and the reality, including 1) VS feedback decoupled from slow shape control, 2) adaptive coil trajectory calculations for integral gain reductions, and 3) MIMO-style relay feedback algorithm which was designed for simultaneous tuning of multiple points PID.

3. Exploring high beta operation scenarios in preparing ITER and DEMO

In preparing the fusion reactor realization, solving the scientific and technical issues in the steady-state high beta scenarios is one of the essential research topics in the superconducting tokamak. In this regards, KSTAR has been focused on developing the scenarios to extend operation regime into high beta and long pulse. As a results of the recent efforts, there have been a remarkable improvement in developing the high performance plasma discharges such as i) stationary high β_N (~ 3.0) operation sustaining more than 3s without tearing mode onset, ii) very high β_P (~ 3) advanced scenario with fully non-inductive current drive, iii) hybrid scenario, internal transport barrier (ITB), and reversed profile.

3.1. **Development of stationary high β_N plasma surpassing no-wall limit**

KSTAR plasmas have reached the device record high β_N in the recent operations [11]. In the earlier experiment, the highest β_N greater than 4 was transiently reached and sustained for longer than a few τ_E. The ratio of β_N/l_i has reached 6.3 which is a high value for advanced tokamaks. Total pulse lengths of this high β_N discharges were initially limited to approximately 1.5 s.

In subsequent device operation, high β_N operation was significantly extended to longer pulse by utilizing fast plasma radial control [12]. The evolution of the reconstructed plasma parameters for the shot that sustained the high β_N > 3 for the longest duration of 3 s up to date are shown in Figure 3. The plasma current was initially maintained at higher level of 470 kA then was lowered to 430 kA at B_T = 1.2 T for better equilibrium control at the earlier phase of the discharge The total neutral beam heating power (P_{NBI}) from three beam sources is 4.5 MW. The l_i values at β_N > 3 ranging from 0.8 to 1.4 indicating that many of the equilibria stay above $\beta_N^{\text{no-wall}}$. The high β_N phase was limited by the onset of 2/1 tearing mode around t = 4.3 s in the discharge.

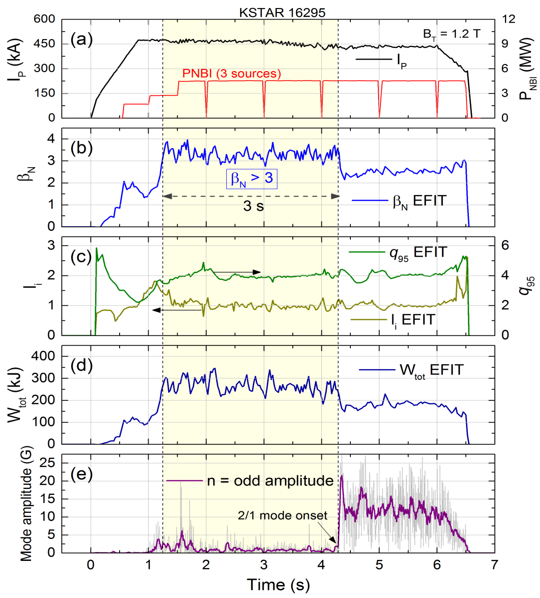

FIG. 3. KSTAR high β_N discharge evolutions showing (a) plasma current and neutral beam heating powers, (b) reconstructed β_N, (c) edge safety factor (q_{95}) and l_i, (d) plasma stored energy and (e) mode amplitude.

3.2. **Optimization of high β_p plasma**

The high-β_p scenario with a certain reduced the plasma current is an very promising scenario for the high performance long pulse discharge in KSTAR near term due to the limited heating and current drive power. However, the main disadvantage of this scenario in KSTAR is an increase of the bad orbit loss of the fast ion by the larger drift. It causes the overheating of plasma facing component by the fast ion loss striking the edge surface of the outer poloidal limiter, hence the early termination of the plasma discharge by the protection interlock.

With regard to the high $\boldsymbol{\beta_p}$ experiments, conditions of the maximum $\boldsymbol{\beta_p}$ are investigated mainly by parametric scans of I_p=0.4-0.7 MA and also P_{NBI} =3-5MW. The achieved maximum $\boldsymbol{\beta_p}$ is above 3 with lowest I_p of 0.4 MA, B_T=2.9T and P_{ext} ~ 6MW including 0.7 MW of ECH and up to now, maximum $\boldsymbol{\beta_p}$ is found to be limited by heating power and without indication of MHD activities ($\boldsymbol{\beta_N}$ ~2) due to elevated q_{95} ~ 11. In addition, the high $\boldsymbol{\beta_p}$ discharge is, due to higher bootstrap fraction compared with the high $\boldsymbol{\beta_N}$ discharge, closed to the state of fully non-inductive current drive. However, though the developed scenario is a good candidate for advanced steady-state operation, the sustained pulse length is limited by 12 seconds due to an excessive heat-load on the protection limiters. Further optimization will be investigated with active cooling of PFC and improving fast ion confinement by plasma density and gap control.

3.3. **Exploring various advanced scenarios**

The hybrid regime[13, 14] has been continuously explored in KSTAR with the goal of achieving high fusion gain G = H89* β_N /q95^2 in stationary conditions. In recent KSTAR experiments, G up to 0.38, close to ITER baseline scenario (G = 0.4) is achieved with H_{89} < 2.3, β_N < 2.7 at q_{95} = 3.8-4.5. This regime is obtained by applying the third neutral beam (NB) when the fusion performance is linearly increasing with the sawtooth period after applying the second NB. Internal transport barrier (ITB) [15] formation condition is also valuable. Because KSTAR is not fully equipped with heating and current drive systems to see the ITB with H-mode, alternative approach to get ITB was investigated by an early injection of the full NBI power (~ 4.5 MW) during the current ramp-up if the plasma could stay in the L-mode. In the recent experiment, ITB formation during L-mode has been observed and sustained for maximum 3.6 s. As shown in figure 4, the plasma performance such as temperatures, the stored energy and the β_N are comparable to the H-mode in the discharge. Ion and electron temperature profiles show the barrier clearly in the temperature.

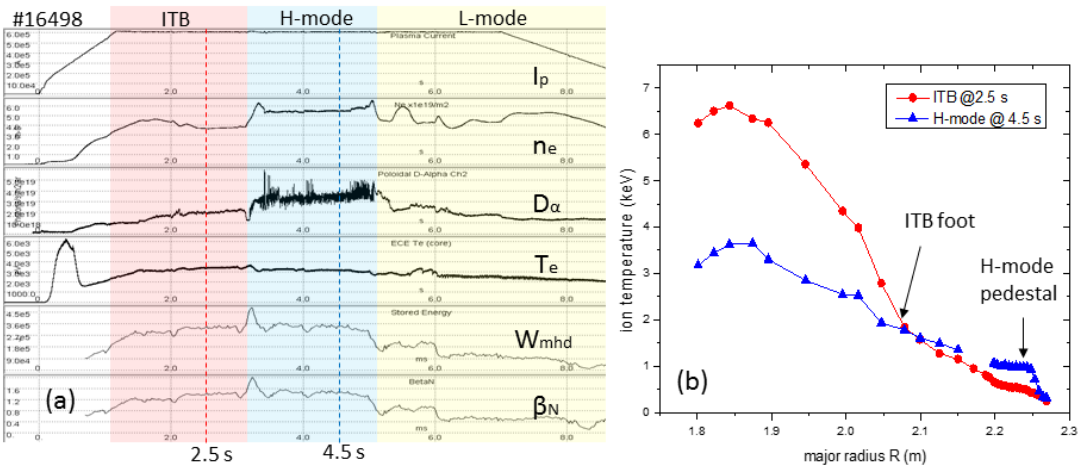

FIG. 4. The first ITB discharge in KSTAR (B_t = 2.7 T). A total of 4.5 MW neutral beam injected in the middle of the current ramp-up (dI_p/dt = 0.5 MA/s). Time trace parameters of the discharge (a) and ion temperature profiles at 2.5 s and 4.5 s (b) show the plasma performance during the ITB.

One of expected benefits in extremely low q_{95} operation is inherent removal of performance limiting MHD modes by pushing dangerous rational surfaces (*e.g.*, 2/1 and 3/2 surfaces) toward safe region (*i.e.*, the region with small pressure gradient) [16]. KSTAR had reached up to q_a=2 limit in limited discharges in 2015 and the operation regime was extended to q_{95}~2.25 in diverted H-mode discharges in 2016. In future, we will systematically explore the confinement of extremely low q_{95} regime with controlling performance limiting MHD modes both in active and inherent ways for investigating the feasibility of the regime in reactors.

4. Exploring the 3D field effect on plasma performance and ELM control

Recently, the KSTAR has been identified with an extremely low intrinsic error field ($\delta B/B_0 \sim 10^{-5}$) [4], as well as with unprecedentedly low level of field ripples (δ_{TF}=0.05%) [5]. As a result, such an order of magnitude lower non-axisymmetric in KSTAR than in conventional devices provides an ideal test bed to accurately address the plasma responses against externally controlled non-axisymmetric fields without being influenced by any insignificant residual non-axisymmetric.

The KSTAR has three rows (Top/Middle/Bottom) of versatile in-vessel control coils (IVCC) (capable of up to n=2) in low field side, whose configuration is similar to that of the planned ITER RMP coils (capable of up to n=4). Unlike other conventional tokamaks equipped with two (Top/Bottom) rows of RMP coils, the KSTAR is the only major tokamak that has in-vessel mid-plane RMP coils. Hence, any uncertainties involved in ITER in-vessel mid-RMP coils can be directly addressed in KSTAR. Taking advantage of broadband switching power amplifiers (SPA) capable of up to 5 kA/turn in DC to 1 kHz, various 3-D configurations with n=1 and/or n=2 fields can be applied in conjunction with three dozens of assorted patch panels in KSTAR [17]. Besides, along with an upgraded magnetic diagnostic (i.e. magnetic probes installed on the passive plates), the plasma response for 3-D field physics study is directly measured by the state-of-the-art diagnostics, such as electron cyclotron emission imaging (ECEI) [18] and microwave interferometry (MIR) [19].

4.1. **Stationary ELM crash suppression achievement using low-n RMPs**

In ITER and future reactors, the presence of ELMs poses substantial risks, in that the out-bursting particle and heat fluxes could damage the divertor and plasma facing components in an uncontrolled manner [20]. Among several schemes to control such ELMs, the application of RMP, which creates stochastic magnetic field to divert the ELM onset condition, has been proven to be quite effective to either suppress or mitigate ELMs.

As one of important milestones, KSTAR has demonstrated a record-long (> 10 sec, more than 90 τ_E) sustainment of RMP-driven ELM-crash-suppressed H-modes, as shown in Figure 5.

Unlike a typical ELM-suppressed discharge in the past in KSTAR (near q_{95} ~ 6) [21], a very wide ELM suppression window of safety factor (q_{95}) is identified near 5.0 ± 0.25 at a level of RMP field $\delta B_{RMP}/B_T$ ~ (5 - 7) x 10^{-4} (with magnetic field B_T=1.8 T, plasma current I_P=0.5 MA, and external heating power P_{NBI} ~ 2.8 MW). Moreover, such superb performance of RMP-driven ELM-crash-suppression has been ascertained to be very reproducible (with more than 100 discharges). Similarly, we have confirmed the validity of the approach even on the n=2 (near q_{95} ~ 4) RMP ELM control, but so far observed the marginal suppression or strong mitigation frequently, rather than full suppression.

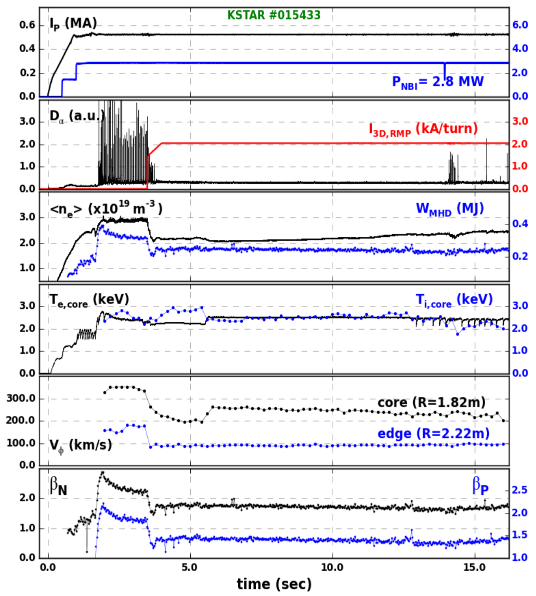

FIG.5. Demonstration of quasi-stationary (> 90τ_E) ELM suppression with n=1 RMP (whose 3-row configuration is similar to what is planned for ITER)

Besides, the use of supersonic molecular beam injection (SMBI) or impurity injection (e.g. Krypton) were tested to see if the ELM characteristics would be altered [22, 23]. On the other hand, the Kr-impurity injection was found to be quite influential on ELMy behavior, leading to strong mitigation or pseudo-suppression [23]. However, since the injected Kr impurity affects the plasma deep in the core, an optimal set of conditions, including dose and injection frequency, is being investigated.

4.2. **Neoclassical toroidal viscosity research using 3D field**

Due to no or little intrinsic non-axisymmetry, KSTAR is capable of providing the most stringent control of neoclassical toroidal viscosity (NTV) impact using its 3-row IVCCs. In fact, according to recent NTV calculations [24], the routine presence of rotation pedestal uniquely observed in H-mode plasmas in KSTAR is a strong evidence of momentum transport barrier, attributable to low non-axisymmetry. As a result, the goal of this multi-year NTV study is to improve the understanding of 3D neoclassical transport, focusing on the toroidal momentum transport in NTV transport that would dominate other classical or anomalous transport mechanisms. Hence, a systematic scan has been conducted to identify the most resonant and non-resonant configurations for both n=1 and n=2. As had been expected, given the limited 3-row configuration of n=2 in KSTAR, the dominantly non-resonant configuration was found to be 0 degree phasing, while the least non-resonant configuration was 90 degree phasing [25]. Besides, the NTV offset profiles are being studied in various Ohmic, and ECH-heated plasmas, whose rotation speed and direction are being accurately diagnosed using both X-ray imaging crystal spectrometer (XICS) and charge exchange spectrometer (CES). Considering that ITER and future reactors are expected to rotate slowly, a detailed study of the relevant mechanism to rotate plasmas based on NTV offset is of great interest in KSTAR.

4.3. **Non-axisymmetric field dependence of power threshold for L-H transition**

Since the introduction of non-axisymmetric field (δB) for RMP-driven ELM control affects the power thresholds (P_{th}), a very systematic study was conducted in DIII-D, showing that unless the RMP strength gets high, there would be minimal impact on P_{th} for L-H transition [26]. Ever since, many other conventional devices reported a similar trend of P_{th}, which appears insensitive in low δB, but linearly increasing in high δB. Considering that KSTAR has a much lower level of n=1 intrinsic error field than conventional tokamaks, we investigated how such δB dependence would appear in an extremely low level of 'uncorrected' intrinsic error fields.

Figure 6 shows the measurement results based on three sets of non-axisymmetric fields. Compared with an empirical scaling (drawn in dashed line) that has no δB dependence [27], the n=1 δB shows a weakly linear dependence at low dB, while showing a strongly linear dependence at high δB. However, the n=2 δB shows a strongly linear dependence even at a low δB, an insensitivity at intermediate dB, but a strongly linear dependence at high δB. Interestingly, when a fixed level of n=1

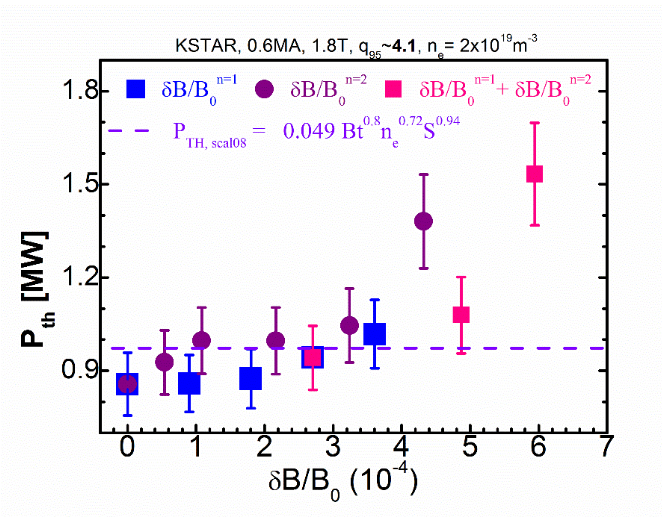

FIG. 6. Power thresholds (P_{th}) dependence on non-axisymmetric fields. Shown are the dependences of n=1 (blue square), n=2 (purple circle), and n=2 on top of a fixed n=1 field (magenta square) respectively.

field ($\delta B/B_0$~$2.7\text{x}10^{-4}$ at edge, where pedestal top could be formed) is overlaid with varying n=2 field, the combined dependence (magenta square) is shown with a steep slope [28].

5. Research on divertor and impurity accumulation in preparing the metallic plasma facing elements operation

One of important design issues for the future fusion machine like ITER and DEMO is the use of metallic plasma-facing elements (or plasma-facing components; PFCs) which are exposed to high parallel power fluxes. PFCs have to tolerate large amount of heat flux during their lifetime without loss of their thermal and mechanical properties. The use of metallic PFCs leads to the high-Z impurity influx to the plasma core, especially in type I ELMy H-mode discharges resulting in the increase of radiated power, thus, the degradation of core performance. Furthermore, fuel retention and removal after steady state long pulse operation are a critical issue for a fusion reactor, since it is directly linked to the tritium amount inside the vacuum vessel. Therefore, it is very important to investigate 1) ELM/inter-ELM heat load on PFCs, 2) SOL characteristics, 3) fuel retention and removal, 4) impurity transport and core accumulation, its control.

5.1. Research on plasma surface interaction (PSI) on Tungsten divertor

One major milestone of KSTAR is to develop technology for long pulse steady state operation relevant to ITER and DEMO. Long pulse steady state operation in next generation of tokamaks results in high heat and particle flux towards PFCs, especially limiter and divertor structure, and PFCs should tolerate such high heat and particle flux. KSTAR has developed successfully ITER-like tungsten bonding technology based on W, copper (Cu), and copper-chrome-zirconium (CuCrZr) alloy [29] as shown in figure 7a)

Furthermore, parallel to the development of tungsten material for ITER divertor, efforts have been taken to find a way to control and to reduce heat load on PFCs by shaping & tilting of the divertor and its castellation structure [30-33]. KSTAR has studied systematically the effect of leading edges since 2012. As shown in figure 7b), various different designs of the tungsten blocks were studied, tested, and reported. Recently, tungsten blocks with different leading edge heights of 0.0 mm (perfectly aligned case), 0.3 mm, 0.6 mm, 1.0 mm, and 2.0 mm were measured the ELM and inter-ELM heat load by high resolution IR system (0.4 mm/pixel).

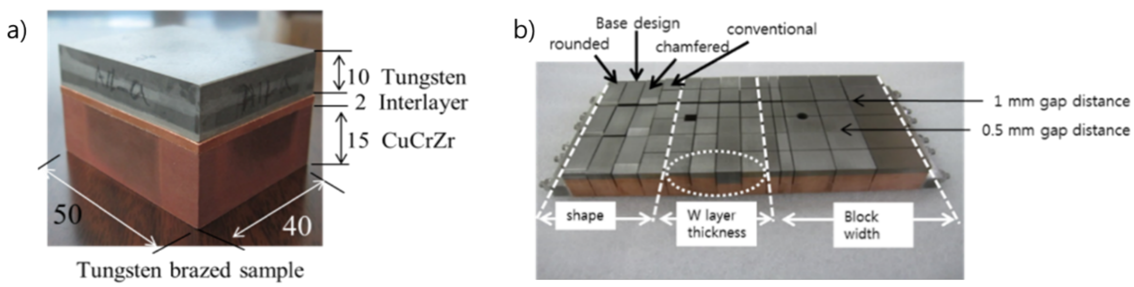

FIG. 7. a) Developed proto-type of tungsten brazed block and assembly of tungsten blocks for dedicated PSI study.

5.2. **Research on heat flux control on Divertor**

Other than the shaping of the castellated blocks, active control of divertor heat load by radiative divertor (including detached divertor) and RMP have been be utilized. Figure 8 shows the results of heat flux measurements by applying from techniques. Figure 8a) shows the heat flux pattern on the divertor during the divertor puffing. As the density increases, the heat load decreases and finally almost vanished indicating that detachment has occurred. In figure 8b), the strike point splitting is clearly shown by the application of RMP. KSTAR RMP coils has ability to rotate the RMP field, which is very important for ITER and future tokamaks. Divertor heat flux obtained from the IR camera has been compared with the SOLPS-ITER modeling results, showing a reasonably agreement.

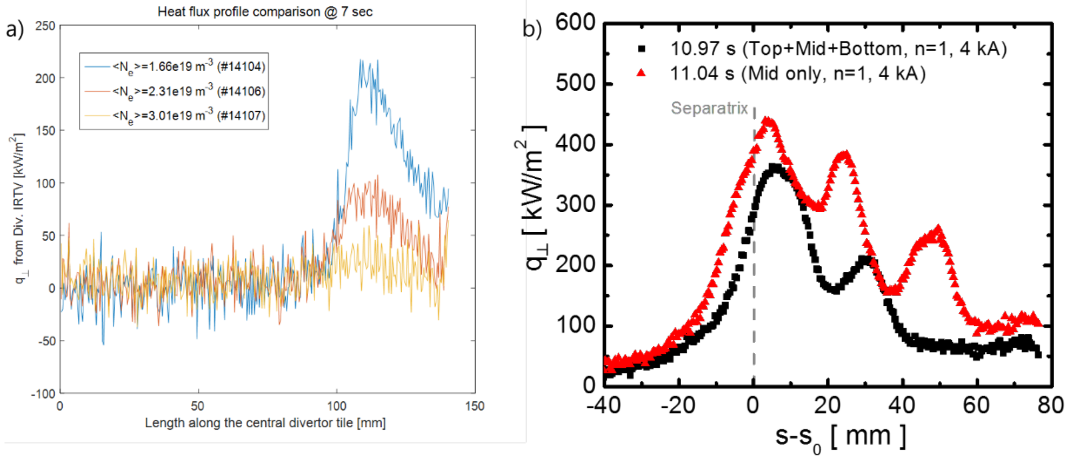

Fig. 8. Heat flux pattern a) on divertor as a function of density, b) during the application of RMP.

The issue of limiter scrape-off layer (SOL) heat flux widths has recently become of much greater importance following the realization that the previously simple approximation of a single exponential fall-off from the last closed flux surface (LCFS) assumed in the design of the ITER first wall panel toroidal shaping was invalid. Double exponential heat flux profiles in the SOL region, a narrow feature (few mm) near the LCFS followed by a broader e-folding length, have now been observed in inner wall-limited (IWL) discharges using fast reciprocating Langmuir probe assemblies (FRLPA) and infra-red (IR) measurements in several tokamaks including KSTAR [33-37].

5.3. **Impurity injection and accumulation control**

Impurity transport experiments with trace argon (Ar) gas injection have been conducted to investigate effects of ECH and RMP on the transport of Ar, of which emission was measured by a soft X-ray array diagnostic (mainly Ar^{16+} and Ar^{17+}) and a VUV spectrometer (Ar^{15+}) [40]. It was demonstrated that ECH and RMP can reduce Ar accumulation at the plasma core and thereby control the impurity distribution. Tungsten (W) powder injection to the plasma was successful using a recently developed powder injector. Significant evidences of tungsten penetration into the plasma were detected by the VUV and EUV spectrometers and visible camera along with changes in density and stored energy by the W injection. Analyses for core impurity transport with the gas electron multiplier detector X-ray pinhole camera and radiation distribution with imaging bolometer are in good progress using tangential tomography validated by various phantom tomographic reconstruction tests.

6. Summary and future plan

There has been a significant progress in the recent KSTAR experiments, especially in developing robust high beta plasma in steady-state operation that are essential for ITER and DEMO.

- Improved plasma shape control and access to 1 MA plasma discharge in H-mode.
- Extension the H-mode flattop up to 55 sec at 0.6 MA utilizing utilizing long pulse operation of NBI and 170 GHz ECCD.
- Achieving stationary high β_N (> 3) plasma operation exceeding no-wall limits and sustained up to 3 sec.
- Achieving advanced high β_P (> 3) plasma operation up to 12 sec with almost fully non-inductive current drive.
- Developed various advanced scenarios such as hybrid scenario, ITB, low q95 (~2.25 in H-mode).

Recent experiments in KSTAR revealed that KSTAR is the best device for the research on 3D field or error field effect on plasma confinement due to extremely low intrinsic error field, in-vessel control coil, and advanced imaging diagnostics in KSTAR.

- Robust ELM-crash suppression at n=1 and sustaining over 10 s
- Finding optimal phasing window in RMP for ELM-crash suppression
- L-H transition threshold power dependence on error field

The plasma surface interaction research and high heat flux effect analysis on tungsten divertor has been conducted very effectively by installing tungsten tile on divertor with castellation.

In the near future, KSTAR research will focus on exploring the key physics and technology by extending the operation window to steady-state, high beta operation under low Z carbon wall environment. To support this research, there will be upgrades in heating and current drive systems.

- Heating & CD system upgrade : NBI system (from 5.5 MW to 12 MW), 105/140 ECH/CD (> 2 MW), Helicon CD (~ 1 MW, 0.5 GHz)
- Improvement in density control : Active water cooling on PFC, divertor cryopump, pellet injector, IC wall conditioning

After several years of operation under carbon wall, KSTAR will be upgraded in around 2021 for the DEMO relevant research in plasma discharge and in technical aspects also. Especially metal divertor, advanced current drive like HFS LHCD, Helicon CD.

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