国際化推進共同研究概要

No. 1

タイトル: Joint study of long pulse high beta discharges and related edge turbulence transport in steady state operation (SSO) plasmas on QUEST and EAST

研究代表者: GAO, Xiang

所内世話人: 花田 和明

実施期間: 2016年10月12日～ 10月24日

研究概要: 2016年10月にQUESTを訪問し、花田教授及び九州大学の学生と共に、QUEST実験の解析、及びQUESTとEASTにおける定常運転に関する議論などを行った。EASTでの高ベータ長パルスのための課題におけるの最近の研究成果をQUESTのメンバーに報告した。EASTで定常維持を阻害する要因について検証を行い、QUESTとの違いについて議論した。このテーマで共著論文がNuclear Fusion誌に投稿され、掲載された。

"Key Issues Towards Long Pulse High βN Operation on EAST Tokamak"

Gao, Xiang; Yang, Yao; Zhang, Tao; Liu, Haiqing; Li, Guoqiang; Ming, Tingfeng; Liu, Zixi; Wang, Yumin; Zeng, Long; Han, Xiang; Liu, yukai; Wu, Muquan; Qu, Hao; Shen, Biao; Zang, Qing; Yu, Yaowei; Kong, Defeng; Gao, Wei; Zhang, Ling; Cai, Huishan; Wu, Xue; Hanada, Kazuaki; Zhong, Fubin; Liang, YunFeng; Hu, Chundong; Liu, Fukun; Gong, Xianzu; Xiao, Bingjia; Wan, Baonian; Zhang, Xiaodong; Li, Jiangang

Nuclear Fusion; Article reference: NF-101531.R1

Report of my visit to QUEST laboratory at Kyushu University

Xiang Gao

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1. Background

My experimental proposal of "Study of steady state operation plasmas on QUEST and EAST" was approved by the Boards for Joint Research at the Joint Usage / Research Centers at Kyushu University in 2016. Therefore, I visited the QUEST laboratory to join in the QUEST experimental study for 13 days based on the approved research plan in 2016.

2. Visiting schedule

(1) Arrive in Japan (Hefei – Shanghai – Fukuoka) on Oct. 12, 2016

(2) Departure for China (Fukuoka – Shanghai – Hefei) on Oct. 24, 2016

3. Visiting report

During my visit in QUEST laboratory, I joined in the study and discussion on Foundation of Plasma Physics with Prof. K. Hanada and QUEST laboratory graduated students on Oct.13, 2016. I joined in the QUEST experimental data analysis and discussion with Prof. K.

Hanada and his Ph.D students regarding steady state operation on QUEST and EAST tokamaks during my visit in Kyushu University.

During the 26th IAEA fusion energy conference which was hold at Kyoto in Japan, I visited in Kyoto for my presentation from Oct.17 to Oct.21, 2016.

I presented my recent work of "Key Issues Towards Long Pulse High βN Operation on EAST" for the QUEST team at Kyushu University on Oct.14, 2016. The higher normalized beta discharges (beta-N =1.5 - 2) which was produced by the NBI heating and 4.6 GHz LHW systems in EAST has been studied recently. It is observed that the beta-N value increases with the increased heating power generally on EAST tokamak. The typical results for higher normalized beta discharges such as the heating power limitation, the increase of the impurity radiation, the dynamic and collapse of an internal transport barrier, and MHD events have been studied in detail. The key issues towards the long pulse high beta-N discharges on EAST were studied and summarized as follows: (1) Increasing and sustaining the reliable high heating power. (2) Control of the impurity radiation during the high power experiments. (3) Study of the ITB dynamic and collapse. (4) Study of the MHD limits towards the high beta operation. The research results of my work were partly supported by the Joint Usage / Research Centers at Kyushu University in 2016.

This collaboration is very helpful for the EAST and QUEST steady state operation. I wish that the collaboration of "Study of steady state operation plasmas on QUEST and EAST" could be strongly supported by the Boards for Joint Research at the Joint Usage / Research Centers at Kyushu University in the future.

国際化推進共同研究概要

No. 2

タイトル:　Joint study of calorimetric measurement of heat load and power balance estimation in steady state operation (SSO) plasmas on QUEST and EAST

研究代表者:　LIU, Haiqing

所内世話人:　花田　和明

実施期間:　2017年2月2日　～2月10日

研究概要:　QUESTとEASTにおける定常運転での熱負荷の計測及びパワーバランス見積もりについての共同研究を実施している。本活動で進めているＥＡＳＴでのパワーバランス計測について論文発表を行い、ＰＰＣＦ誌に掲載された。今後のＱＵＥＳＴとＥＡＳＴでの実験計画について議論を行い、今後、博士学生１名をＱＵＥＳＴに長期派遣する計画について合意した。

次回のＥＡＳＴ実験ではすべての水冷システムにフローメーターと温度センサーが設置され、上部と下部の異なる材質のダイバータを含めた全体の熱負荷及びパワーバランスを調べる予定である。

・"Preliminary study on heat load by using calorimetric measurement during long pulse high performance discharges on EAST";
Liu, yukai; Hamada, Natsuhiko; Hanada, Kazuaki; Gao, Xiang; Liu, Haiqing; Yu, Yaowei; Qian, Jinping; Yang, Lei; Xu, Tiejun; Jie, Yinxian; Yao, Yuan; Wang, Shusheng; Xu, Jichan; Yang, Zhengdong; Li, Gongshun
Plasma Physics and Controlled Fusion; Article reference: PPCF-101329.R1

・"Power balance estimation in long duration discharges on QUEST"
K.Hanada, H.Zushi, H.Idei, K.Nakamura, M.Ishiguro, S.Tashima, E.I.Kalinnikova, Y.Nagashima, M.Hasegawa, A.Fujisawa, A.Higashijima, S.Kawasaki, H.Nakashima, O.Mitarai, A.Fukuyama, Y.Takase, X. Gao, H.Liu, J.Qian, M.Ono, R.Raman
Plasma Science and Technology; 2016.5 accepted; 掲載年月日; 2016.11

(様式 11・拠点)

RESEARCH REPORT

Date: Feb. 23, 2017

Visiting scientists: (name) **Haiqing LIU**

(position) **Associate Professor**

(university / institute) **Institute of Plasma Physics,**

Chinese Academy of Sciences

(name) **Yinxian Jie**

(position) **Professor**

(university / institute) **Institute of Plasma Physics,**

Chinese Academy of Sciences

(name) **Yukai Liu**

(position) **Ph.D student**

(university / institute) **Institute of Plasma Physics,**

Chinese Academy of Sciences

Host scientist: (name) **Kazuaki Hanada**

(position) **Professor**

(university / institute) **Kyushu University**

Research period: (from) **Feb. 2, 2017** (to) **Feb.10, 2017**

Research subject: Joint study of calorimetric measurement of heat load and power balance estimation in steady state operation (SSO) plasmas on QUEST and EAST

Introduction

Steady state operation (SSO) of magnetic fusion devices is one of the goals for fusion research. As it is predicted that an enormous heat flux (10MW/m^2) is coming to the diverter (vertical heat target) locally from the plasma in the future fusion reactor, the heat load distrubution (power balance) and its control should be investigated to realize future fusion power plants. In a future steady-state fusion power, huge heat load is expected from main plasmas to plasma facing components (PFCs) and it must be handled in steady-state. Actually, control of contact point of PFCs to plasma has been applied in many long duration discharge devices such as TRIAM-1M, QUEST, EAST and on which long duration discharges can be successfully obtained. The world record of plasma duration on tokamaks for more than 5 hours was achieved in TRIAM-1M, where an accurate power balance of the discharge was investigated and a particle balance was also studied. However, the longest plasma is spontaneously terminated and the reason is still unclear. The mission of QUEST is to develop the scientific basis for achieving a steady state condition at sufficiently high beta (~20%), with high confinement and low collisionality. Although the strong modification of plasma configuration was applied in QUEST, much of the heat load to the outer vessel was still remained. It means that the heat load is mainly supplied from energetic electrons which are generated by injected RF electric field. The mission of the EAST project is to study the physical issues involved in steady state advanced tokamak devices. Long pulse operation is directly related to ITER technology and operation. EAST device have fully actively water cooled plasma facing components (PFCs), so calorimetric measurement can be easily applied to measure heat load of PFCs on EAST. We can derive the heat load distribution and power balance during steady - state operation. Joint study of calorimetric measurement of heat load and power balance estimation in steady state operation (SSO) plasmas on QUEST and EAST will give heat load and power balance database for SSO plasma. Joint study in QUEST and EAST is approachable way to reveal the underlying mechanism and understand the mechanism of termination of long pulse of SSO plasma. The result and database will contribute to ITER and future fusion reactor.

Recently experimental progress in QUEST and EAST

The steady state operation or long pulse operation is one of key roles in burning plasma devices in order to apply fusion energy to commercial electric generation. One of the remaining challenge is handling the heat and particle flux expected in steady state tokamak operation without compromising the plasma performance. Plasma confinement degeneration during long-pulse discharge could be caused by increment of first wall temperature then boundary recycle enhance. So water cooling of PFCs is important to achieve high parameter and long-pulse operation in EAST. The cooling capability strongly depends on the distribution of the heat load to the PFCs, the change of distribution of the heat load strong depends on plasma confinement. Understanding of plasma parameters dependent on distribution of the heat load is in favor of extrapolating result to ITER. It's certainly worth researching distribution of heat load of PFCs in EAST with steady-state or long-pulse operation. Injected power should be equilibrium with heat load brought by cooling water system. To achieve steady-state or long-pulse operation, it's important to investigate the power balance. For SSO plasma, recently research status was shown in Fig. 1.. Measurement of heat load and researching of power balance in EAST and QUEST will provide crucial support for ITER experiments.

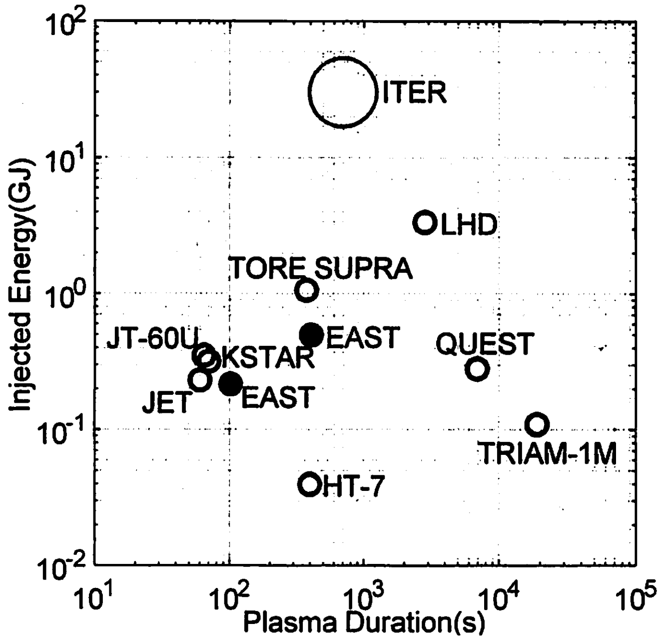

Figure 1. Review of the steady state or long pulse discharges. Maximum injected energy is plotted versus plasma duration. Blue solid circle shows the new high performance discharge obtained in 2015-2016 campaign on EAST. Red solid circle shows the long pulse discharge over 400s entirely driven by lower hybrid current drive in 2012.

Mr. Hamada and Mr. Liu had done calorimetric measurement of heat load on D module on EAST. The Schematic diagram of water cooling modules on EAST is shown in Fig. 2.

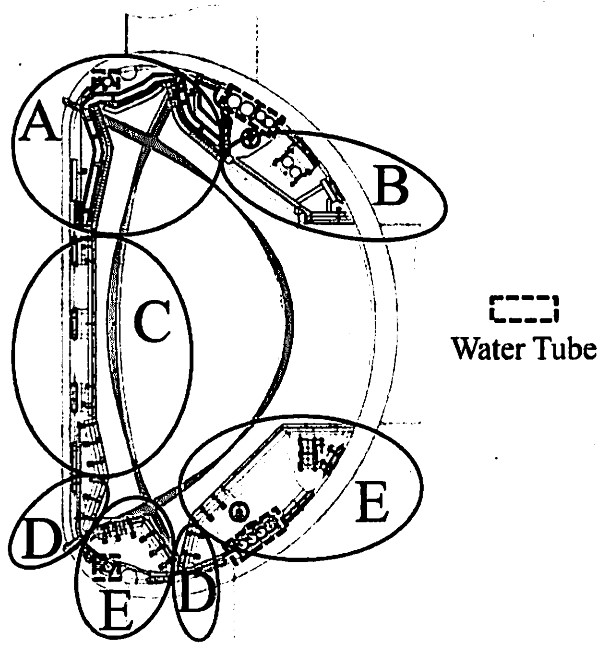

Figure 2. Schematic diagram of water cooling modules on EAST presented from poloidal cross-section. A is integral upper divertor; B is upper passive plate and upper low field side; C is non-strike point region on lower inner divertor, high field side; D is strike point region on lower divertor; E is lower dome, non-strike point region on

lower outer divertor, lower passive plate and lower field side.

Figure 3 illustrates the evolution of flow velocity in a main water tube. The flow velocity is found to be almost constant at 2.02 m/s with two consecutive days. According to calculation formula of Reynolds number, it is equal to 92828 much larger than 4000, so the state of water in main tube is turbulence. Figure 4 shows the evolution of temperature difference (ΔT) of two main tubes during all day. Clear increment of the temperature difference caused by plasma operation was observed. And two water cooling curves are almost same. It is indicated that the heat load on D module transported by cooling water is equational toroidally.

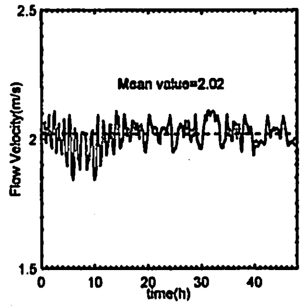

Figure 3. Evolution of flow velocity in one main water tube during two consecutive days.

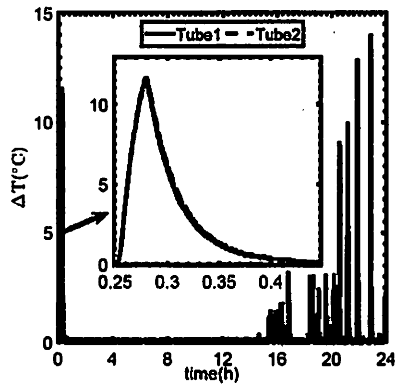

Figure 4. Evolution of temperature difference (ΔT) of water in two main tubes during all day. There are some ohmic cleaning discharges and disrupted discharges which were disrupted when plasma current ramp up. Because the injected energy of these discharges is very small, the temperature difference remain approximately constant from 00:15 to 15:25.

The comparison of measurement results of heat load on strike point region of lower divertor measured by divertor Langmuir probes and calorimetry diagnostic is carried out (Figure 5). Assuming that the divertor energy flux is approximately toroidally symmetric, the time integrated heat load on the divertor targets measured by Langmuir probes is

$$\Delta W_{div} = \int_{t}^{t+\Delta t} \int_{s_a}^{s_b} 2\pi R_{div} q_t \left(R_{div}, s, t \right) ds dt \tag{1}$$

Where s is the poloidal coordinate along the target plate. R_{div} is the major radius at the corresponding probe tip. q_t is target heat flux. This indicates the most of heat load was delivered on the divertor panels as plasma, not radiation and charge exchange neutrals due to limit area of strike point region on lower divertor compared with all of the first wall.

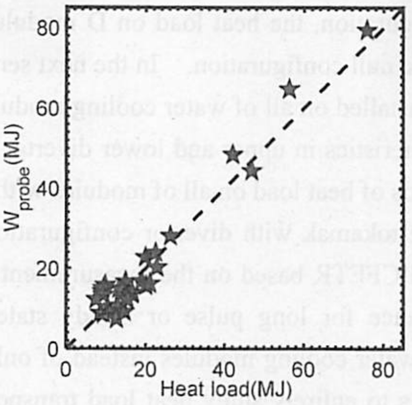

Figure 5. Comparison of measurement results of heat load measured by divertor Langmuir probes and calorimetry diagnostic.

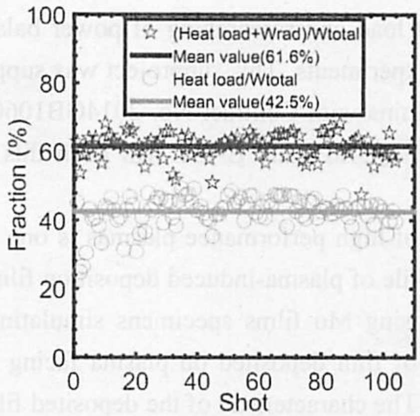

Figure 6. Ratios with 104 shots in lower single null plasma configuration.

Figure 6 illustrates statistically two kinds of ratio with 104 shots in lower single null plasma configuration. It indicated from Figure 6 that the ratio of heat load to total injected energy is 42.5%. The rest of injected energy is exhausted by other water cooling modules. If the radiated energy loss measured by fast bolometer diagnostic is taken into consideration, it will find the ratio to be 61.6%.

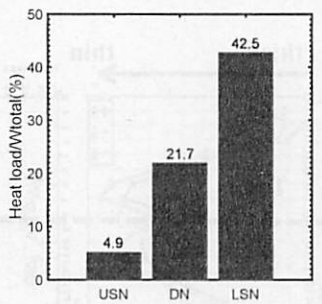

Figure 7. The ratio of heat load on D module to total injected energy including ohmic energy and auxiliary heating energy with upper single null(USN), double null(DN) and lower single null (LSN) plasma configuration, respectively.

Typically, double null configuration and upper or lower single null configurations in EAST can be produced and stably controlled. The ratios of heat load on D module to total injected energy including ohmic energy and auxiliary heating energy with three kinds of plasma configurations are shown in Figure 7. In upper single null configuration, the heat load on D module is only 4.9% of total injected energy. And it's 21.7% for the double null configuration. In the next series of experiments, flow meters and temperature sensors would be installed on all of water cooling modules instead of only D module to investigate the water cooling characteristics in upper and lower divertor made by tungsten and graphite respectively, and to compare the ratios of heat load on all of modules with three plasma configurations.

As a first full superconducting tokamak with divertor configuration, the base of experiment was provided for operation of ITER and CFETR based on the measurement of heat load on D module and partial investigation of power balance for long pulse or steady state operation on EAST. In next campaign, it is expected that all of water cooling modules instead of only D module would be installed flow meters and temperature sensors to entirely study heat load transport capacity with upper divertor and lower divertor made by different material (tungsten and graphite respectively), heat load distribution and power balance on EAST.

Because measurement of heat load and researching of power balance in EAST and QUEST will provide crucial support for ITER experiments. This subproject was supported by the National Magnetic Confinement Fusion Program of China with Contract No. 2014GB106002 (Associate Prof. Liu) in the next two years. The joint study of QUEST and EAST will push this subproject forward in the next collaboration.

Steady-state operation (SSO) of high performance plasmas is one of the key issues to accomplish fusion power plants. Thickness profile of plasma-induced deposition film on the plasma-facing wall also was investigated in QUEST. By using Mo films specimens simulating the deposits, a clear positive correlation between the thickness of film deposited on plasma facing wall (PFW) and the amount of retained hydrogen has been found. The characteristic of the deposited film on PFW is very important for SSO. The thickness profile of deposited film on wide-area PFWs have been investigated by measuring the reflectivity of light (figure 8). The thickness of deposited film is non-uniform on PFW. It could be assumed that the ions attacking on PFW play an important role on forming asymmetric thickness profile of deposited film in poloidal direction on PFW. In typical QUEST discharges, because of the direction of toroidal magnetic field the ions drift upward and make the sputtering on the wall. On upper wall the erosion plays an important role. The film is very thin. On lower wall the deposition plays an important role. The film is thick.

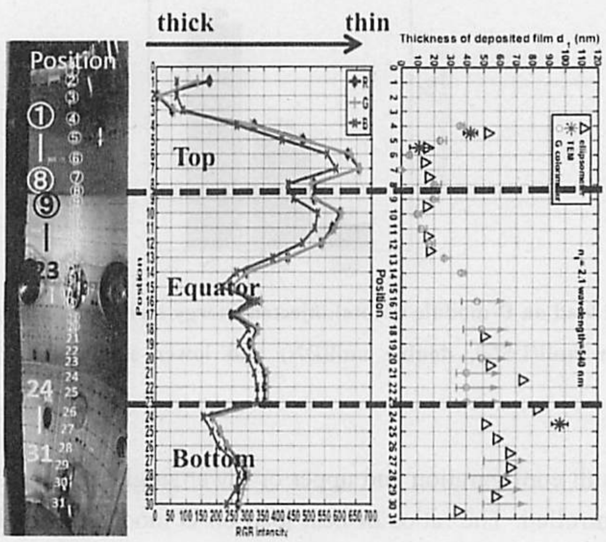

Figure 8. Left: The illustration of the location of the numbered positions. Middle: The distribution of the reflectivity measured with the colorimetry of red light (red line) of 615 nm, green one (green line) of 540 nm, and blue one (green line) of 465 nm. Higher reflectivity means thinner film. Right: the thickness of deposited film measured by ellipsometer, colorimeter and TEM.

Plasma-facing wall interaction (PWI) is a major problem for SSO. Tritium retention in ITER wall occlusion is limited to 700g. Properties of vessel wall outgassing of hydrogen after long duration discharge in QUEST also was investigated. The emission flux of the facing wall is shown as follow

$$\Gamma_{out} = s\gamma_{out} = \frac{k}{sd^2} H_{wall}^2 \tag{2}$$

Where k is recombination coefficient, s is surface area of redeposition layer, d is thickness of redeposition layer, H_{wall} is amount of hydrogen occluded in the wall. Figure 9 indicates that k changes by changing the wall temperature. Experimental results are consistent with hydrogen barrier model.

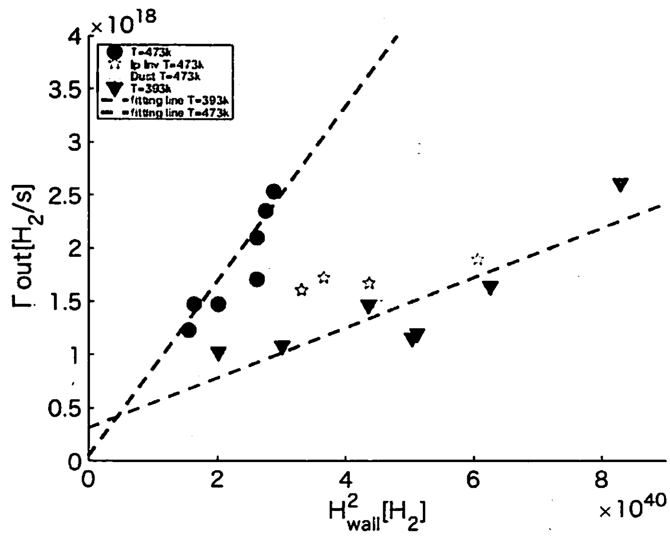

Figure 9. The emission flux of the facing wall as the function of the square of the amount of hydrogen occluded in the wall.

Discussions

QUEST and EAST are both to develop the scientific basis for achieving a steady state condition. Now here has a new start point for the comparative and joint study on QUEST and EAST, especially in study of calorimetric measurement of heat load and power balance estimation during high beta discharges, high performance (confinement and transport) SSO operation. QUEST has installed the ITER like hot wall. EAST has installed the ITER like tungsten upper divertor and has 26MW heating power totally. The joint study results now and in future may shed light on the ITER SSO scenario. Also the new joint subproject, study of power balance and particle balance in EAST and QUEST, has been supported by two part and will be started as a new collaboration research.

During this visit, several interesting topics are also involved in discussions. Those are "Investigation of thickness profile of plasma-induced deposited film on the plasma-facing wall in QUEST" by Mr. Wang, "Properties of vessel wall outgassing of Hydrogen after long duration discharge in QUEST" by Mr. Long, "preliminary study on heat load by using calorimetric measurement during long pulse high performance discharges on EAST" by Mr. Liu.

Acknowledgement and comments:

Work supported by the international joint research at the Joint Usage of Research Centers for Applied Mechanics for 2017. We would like to thank our host, Professor K. Hanada, who helps a lot during our staying at QUEST and very appreciate the useful discussions and comments. It is a good chance for us to join in study in the QUEST. Also Mr. Hamada, Mr. Long and Mr. Wang are thanked for their helpful discussions and thank Kawasaki san for giving a detailed introduction of baking system on QUEST. Ms. Yamakuchi , Ms. Kono and Ms. Funaki are thanked for their helps for this visit. We hope that the international joint research at the Joint Usage of Research Centers for Applied Mechanics could continue to enhance China-Japan cooperation on fusion plasma research in the future.

(Signature) H.Q.LIU

(Name in print) Haiqing Liu

国際化推進共同研究概要

No. 3

タ イ ト ル:　Develop and improve EFIT code of the plasma equilibrium reconstruction for SSO operation and advanced physical study on QUEST

研究代表者:　QIAN, Jinping

所内世話人:　花田　和明

実 施 期 間:　2017 年 2 月 2 日　~2 月 10 日

研 究 概 要:　世界標準のEFITコードによるQUESTデータの解析を行っている。28GHzの第二高調波電流駆動プラズマにおいて、EFITで計算されたプラズマ圧力とトムソン散乱で計測されたバルク電子圧力に20倍の違いがあることがわかり、高速電子成分が平衡にも大きな寄与を持っていることが明らかとなった。この結果は現在共著論文として投稿中である。また、本課題で行ったEASTでのEFITを用いた実験結果についてNuclear Fusion 誌に投稿され、掲載された。

・"EAST equilibrium current profile reconstruction using polarimeter-interferometer internal measurement constraints"
J.P. Qian, L.L. Lao, H.Q. Liu, W.X. Ding, L. Zeng, Z.P. Luo, Q.L. Ren, Y. Huang, J. Huang, D.L. Brower, K. Hanada, D.L. Chen, Y.W. Sun,
B. Shen, X.Z. Gong, B.J. Xiao and B.N. Wan
Nuclear Fusion ; Article reference: NF-101759,掲載年月日 ; 2016.12

・"An efficient technique for magnetic equilibrium reconstruction with q profile constraints and its application on the EAST tokamak"
Qian, Jinping; Lao, L; Holcomb, Chris; Wan, Baonian; Sun, Youwen; Moreau, Didier; Li, Erzhong; Zeng, Long; Hanada, Kazuaki; Garofalo, Andrea; Gong, Xianzu; Shen, Biao; Xiao, Bingjia
Nuclear Fusion ; Article reference: NF-101759

（様式 11・拠点）

RESEARCH REPORT

Date Feb. 28 2017

Visiting scientist: (name) Jinping Qian

(position) Associate Professor

(university / institute) Institute of Plasma Physics,

Chinese Academy of Sciences

Host scientist: (name) K. Hanada

(position) Professor

(university / institute) Kyushu University

Research period: (from) Feb. 3, 2017 (to) Feb. 9, 2017

Research subject: **Develop and improve EFIT code of the plasma equilibrium reconstruction for SSO operation and advanced physical study on QUEST**

The visiting started on Friday 3th Feb. 2017, when our host, Professor K. Hanada, welcomed us on site and led us to the meeting room.

During our visit, QUEST did not start the experiment of this campaign, Hanada and I talked the EFIT reconstruction for the QUEST RF discharge based on the last campaign. Then the PhD student, Hatem Elserafy and I discussed the equilibrium reconstruction on QUEST. Firstly, Hatem helped download all magnetic data which were required by EFIT. Then, at the very beginning, we worked together, benchmarked all flux measurements with the vacuum shots, i.e., PF35, PF26 & PF17. Most of the calculated flux loop can match the measurement very well. Next, the statistic error bars were used as the inputs for EFIT shown as the fitting weights. Finally, we optimized the flux loop data by switching on and off signal in the input file. All of those RF discharges, the discharges without using PF4 coils, can be well reconstructed. For those discharges of using ohmic coils cannot be reconstructed. The reason is, we have no vacuum shots for the benchmarking of flux loop data and no data in the support of patch panel of PF4.

In addition, our collaboration of EFIT paper EAST equilibrium current profile reconstruction using polarimeter-interferometer internal measurement constraints was published on Nuclear Fusion 57 (2017) 036008. Another paper related was submitted to Nuclear Fusion.

As a conclusion to this visit, we of the EAST experiment team warmly thank professor Hanada for welcoming us and showing us QUEST activities and very appreciate the useful discussion and comments between EAST and QUEST.

Note that since my compute was broken during my stay, so no figure of the analysis is shown in this report.

（様式 11・拠点）

(Signature) J. Qian

(Name in print) Jinping Qian

国際化推進共同研究概要

No. 4

タ イ ト ル:　Numerical calculations for the PDP using PdCu membrane

研究代表者:　SHARMA, SANJEEV, KUMAR

所内世話人:　図子 秀樹

実 施 期 間:　2017 年 3 月 19 日 ～3 月 26 日

研 究 概 要:　QUESTにおける透過測定をNi薄膜を用いて行った。異なるタイプの長時間放電実験において、原子およびイオン(照射)フラックスの空間分布を、透過プローブおよびラングミュアプローブを用いて初めて直接測定した。透過型プローブを信頼性の高い診断ツールとして確立させてプラズマ実験に広く利用されるために 、今後も研究を継続していきたい。

Plasma driven permeation measurement using PdCu membrane

Sharma Sanjeev Kumar (Institute of Plasma Research, India)

1. Introduction

Retention and recycling of hydrogen isotope fluxes from the plasma facing components are required to be estimated to control fuel particles inside the fusion grade plasma devices. The hydrogen retention (permeation) depends on incident particle (ions and neutrals) fluxes, temperatures and surface/bulk properties of the plasma facing materials e.g. the surface recombination, diffusion coefficients. Permeation of hydrogen through a metallic membrane for incident atomic (or ionic) hydrogen isotopes is several orders of magnitude higher than that for the molecular flux. This makes the permeation probes suitable for measuring incident atomic or ionic hydrogen fluxes on the walls. In past, the permeation probes (Fe and Ni membranes) have been used in tokamak like TEXTOR and spherical tokamak QUEST to estimate incident atomic/ionic fluxes on the first walls.

Initially, the permeation measurements in the spherical tokamak QUEST were carried out using Ni membrane. The atomic hydrogen fluxes were investigated as a function of various operational parameters like RF power, gas pressure and magnetic configuration. Using the static particle balance and permeation measurements, the progress of wall conditioning was also investigated.

Recently, a new permeation membrane based on $Pd_{60}Cu_{40}$ membrane was installed in QUEST. The PdCu membrane (out of other materials e.g. Ni and F82H) was selected for the permeation probe system for its faster response time, wider permeation regime, and higher sensitivity & independence for incident hydrogen flux. The permeation of hydrogen atoms in these probes is determined by the surface processes instead of diffusion in the bulk of the materials. Obtained results suggests that the modified permeation probe is also capable of measuring dynamic variation of the incident atomic/ionic hydrogen flux on the walls. An array of the five membrane probes has been installed inside the QUEST vessel as shown in figure 1.

2. Experimental

Various newly designed permeation probes (PdCu), mounted on the plasma facing walls of the spherical tokamak QUEST, are used to measure atomic (or ionic) hydrogen flux distribution (fig. 1). The measurement positions are corresponding to various poloidal locations in the spherical tokamak QUEST. Distribution of atomic hydrogen flux was measured in different kinds of plasmas for estimating its influence on the fuel recycling during these experiments. Mainly two kinds of plasmas were used for permeation measurements 1) non-confined annular electron cyclotron resonance (ECR) plasma and 2) properly confined & long duration plasmas. During these experiments, the temperature of QUEST vessel was kept at ~ 100 °C and the temperature of plasma facing walls was ~200 °C. Gas fuelling was done with mass-flow controller. Measurement of the particle fluxes was done with permeation probes distributed poloidally from θ = -95° to 85°, where the poloidal angle $\theta = 0$ corresponds to the mid plane. As the permeation flux through metals is indistinguishable for ionic or atomic fluxes (H^+ or H^o), Langmuir probes were also used for estimating the contribution of the ions.

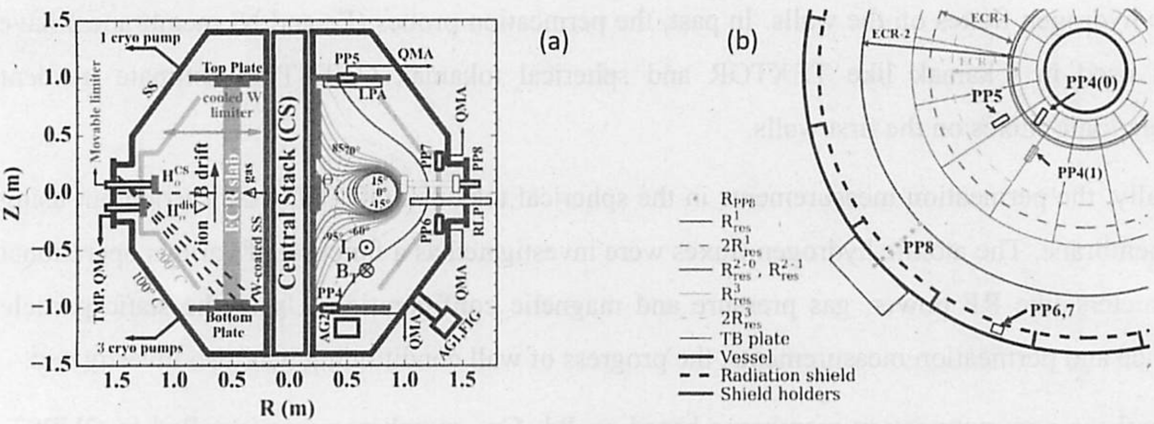

Fig.1. Positions of permeation probes in plasma facing walls of QUEST showing in (a) vertical section view (b) horizontal section view. The other important parameters are also shown in the figures.

3. Results of permeation measurements

The poloidal distribution of the incident fluxes consisting of both protons and atomic hydrogens is measured directly with the permeation probe. The results are compared with the ion flux measured with a Langmuir probe array or neutral hydrogen flux measured by hydrogen Balmer line intensity (H_α) as shown in figure 2. The figure depicts the poloidal distribution of average value of atomic fluxes in various experiments. Here Φ_{PP}, Φ_{LPA} and $\Phi_{H\alpha}$ are time integrated fluxes (Γ_{PP}, Γ_{LPA} and $\Gamma_{H\alpha}$) from permeation probe, Langmuir probe array and H_α respectively., The permeated flux is converted to the incident flux by solving a diffusion equation of hydrogen atoms in the membrane metal. The total fluence (time integrated flux over the discharge period) is compared with the amount of retained hydrogen. The retained hydrogen is derived by a global gas balance method based on a difference between external pumping and fueling amounts.The amount of retained hydrogen is determined as the difference between injected and pumped out amount of the hydrogen atoms.

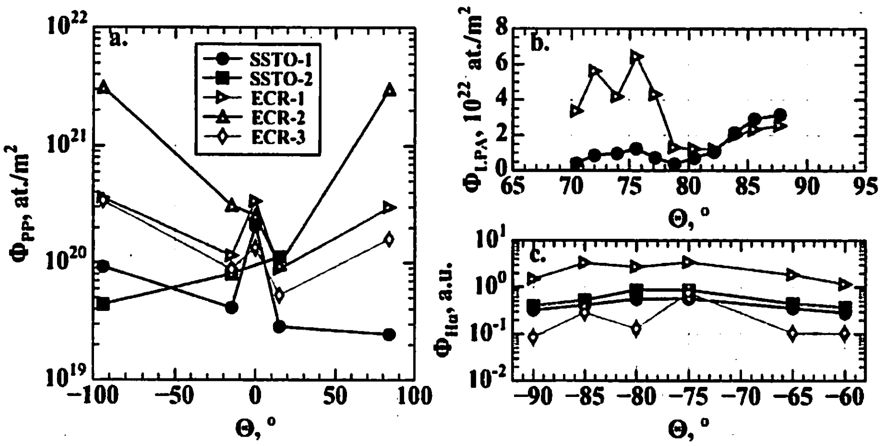

Fig. 2 (a) poloidal distribution of average value of atomic fluxes in various experiments. Φ_{PP}, Φ_{LPA} and $\Phi_{H\alpha}$ are time integrated fluxes (Γ_{PP}, Γ_{LPA} and $\Gamma_{H\alpha}$) from permeation probe, Langmuir probe array and H_α respectively. Here SSTO-1, SSTO-2, ECR-1, ECR-2 and ECR-3 are various types of plasmas.

Spatial distribution of the hydrogen wall irradiation fluxes has been directly measured in different types of long duration discharges with permeation probes and the Langmuir probes. The fluxes are also compared with that of hydrogen spectroscopy. The poloidal distribution of incident atomic hydrogen flux (Γ_{inc}) determined by the permeation probes shows that for ECR plasma, the atomic hydrogen flux peaks near the main PWI regions, corresponding to the intersecting region of the ECR plasma on the top and bottom plasma facing plates, and the profile is almost up-down symmetric. For SSTO, the incident atomic flux to the top & bottom plasma facing plate is reduced compared with those near the outer horizontal wall. Γ_{inc} at $\theta \sim 0°$ does not depend on the plasma configuration strongly. It was deduced that the high-field side walls in SSTO are mainly irradiated by hydrogen atoms.

4. Radial Distribution of atomic/ionic fluxes

Radial scan of atomic and ionic flux was carried out using the reciprocating permeation (RPP) and Langmuir probes (RLP) during ECR-3 type discharges. Radial scan of two types of probes in the same types of discharges was carried out to understand differences in atomic and ion fluxes far from the main PWI region both in the plasma side and outside. Two regions are separated by the radiation shield at R=1.3m. The radial profile of the irradiation flux by reciprocating permeation and Langmuir probes are shown in figure in 3.

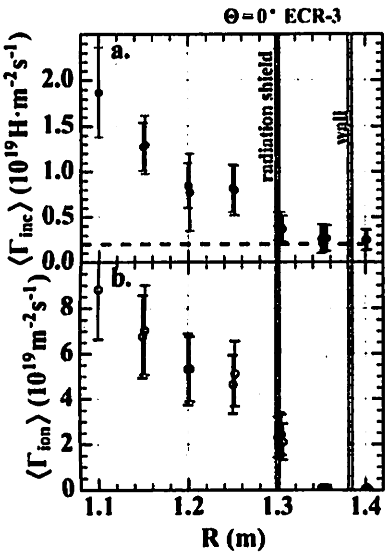

Fig.3. Radial profile of the irradiation flux by reciprocate probes RPP (a) and RLP (b). Time average values of the fluxes are shown

Radial distributions of incident atomic and ionic fluxes (Γ_{inc} and Γ_{ion} respectively) are measured along the major radius inside the plasma side and behind the radiation shield on the mid-plane. In the former region, their relative profiles are consistent with each other, suggesting that the main irradiation flux to the PFCs originates from the radially diffusing plasma ions. On the contrary, beyond the PFC, Γ_{ion} inside the closed port section can be completely neglected. Here Γ_{inc} is mainly dominated by the atomic hydrogen flux. The Γ_{inc} profile behind the PFC is almost constant at ~2×10^{18} H/s/m^2, and this net Γ_{inc} to the hydrogen retention is 2-7% of the slow hydrogen molecule flux evaluated by the pressure. It can be concluded that the ion flux is dominant inside the PFCs but the atomic hydrogen plays an important role on the irradiation flux behind the radiation shield or inside the closed port section.

5. Conclusion and future work

Spatial distribution of the atomic and ionic (irradiating) fluxes has been directly measured for the first time in different types of long duration discharges with permeation and Langmuir probes. It is encouraging to continue the R&D work to establish the permeation probes as an important and rugged diagnostic tool to be used in all the experimental and fusion grade plasma machines.

国際化推進共同研究概要

No. 5

タイトル: Measurement of plasma flow, edge turbulence and coherent modes and their effect on plasma confinement in QUEST

研究代表者: BANERJEE, SANTANU

所内世話人: 図子 秀樹

実施期間: 年月日 ～ 月 日

研究概要: QUEST特有のInboatd Poloidal field Null (IPN) 閉じ込めシナリオでは、0.85～0.15 kHzの低周波コヒーレントエッジ揺動が、長距離ポロイダル相関を伴って放射状に局在している。この低周波揺動は、オーミックプラズマにおけるECW注入の10～20ms後およびInboard Limiter (IL)配位からIPN配位へ遷移後に励起される。しばらくすると、ECW注入を続けても低周波揺動は消える(存続時間はショットごとに異なる)。この揺動は一般的なGAMの周波数範囲にあるが、現在のデータセットからは正確な特徴付けが不可能であるため、今後検討したい。

Report on the international joint research for 2016-17.

Topic: Measurement of plasma flow, edge turbulence and coherent modes and their effect on plasma confinement in QUEST

Santanu Banerjee
Scientist-SE,
Institute for Plasma Research,
Bhat, Gandhinagar 382 428, Gujarat, INDIA

During the tenure of this collaborative research, edge turbulence and plasma flow at the SOL are analyzed in the electron cyclotron wave induced high β_p plasma in QUEST. Electron cyclotron wave (ECW) injection in the Ohmic plasma features excitation of low frequency coherent fluctuations near the separatrix and enhanced cross-field transport. Plasma shifts from initial high field side (HFS) limiter bound (IL) towards inboard poloidal null (IPN) configuration with steepening of the density profile at the edge. This may have facilitated the increased edge and SOL fluctuation activities. Observation of the coherent mode, associated plasma flow and particle out-flux, for the first time in the IPN plasma configuration in a spherical tokamak may provide further impetus to the edge and SOL turbulence studies in tokamaks.

Fast visible imaging system on QUEST

The fast visible imaging system is now equipped with a new fiber bundle and modified fixtures and front end lens holder assemblies. This give more maneuverability to the optics and it is now easier setting it up. Fig. 1(a) shows the back end of the fiber bundle connected to the fast camera though the image intensifier. Also, the front end can be seen in Fig. 1(b). The front end unit enables us to rotate the fiber view in alt-azimuth directions.

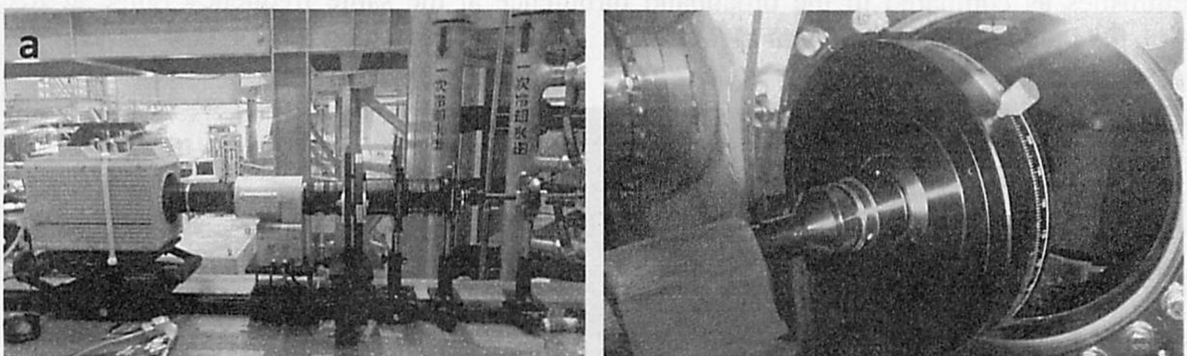

Fig. 1: (a)-(b), Back and front end of the fiber bundle on the fast visible imaging system.

Low frequency coherent mode (LFCM) and SOL flow in the IPN plasma in QUEST

Data acquired using the fast visible imaging system and SOL probes during the ECW induced OH discharges. This analysis primarily deals in the observation and characterization of the LFCM. Also, an attempt is made to characterize the broadband background fluctuations as well. Reconstructed flux contours superimposed on a raw image is shown in Fig. 2(a). Contours are numbered for further reference in the analysis. Spectral characteristics at the SOL are determined by the magnitude squared coherence estimate (C) of intensity fluctuations ($\widetilde{I}$) along the flux contours shown in Fig. 2(a). A distinct mode (LFCM) appears at ~0.8 kHz, much lower than the ion cyclotron frequency ($2\pi\Omega_i$~4.4 MHz) and remains along the flux contours suggesting long range poloidal structure of the LFCM. Coherence length of the LFCM is calculated to be ~40 cm along and ~10 cm across the flux surfaces. The mode frequency varies as 0.85 ± 0.15 kHz for all the shots in this dataset.

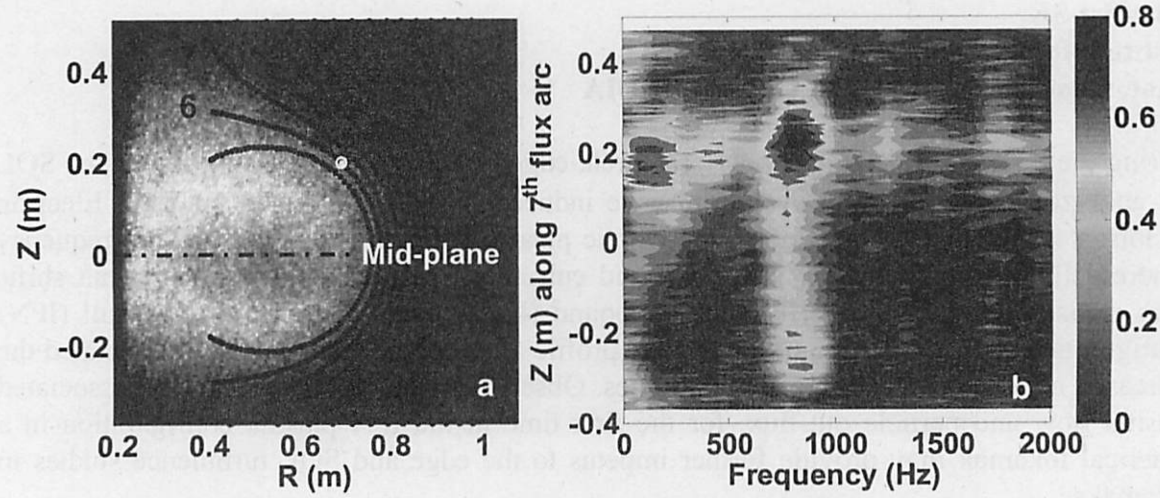

Fig.2: (a) Raw image superimposed with flux contours in the SOL. Reference pixel for coherence (C) estimate is highlighted in white (R=0.69m and Z=0.2m); (b) C of all the pixels along the 7[th] flux contour.

Time dependence of the LFCM is illustrated by the spectrogram of the intensity $\tilde{I}$ in Fig. 3(a)-(c) for R_{edge}, R_{sep} and R_{sol} for the representative shot #17633. Spectrograms show a fairly clear spectral peak in the R_{edge} and R_{sep} regions at 0.75-0.9 kHz, but the peak amplitude diminishes as compared to the broadband turbulence at the same frequency range at R_{sol}. Also, the mode is most prominent at R_{sep}.

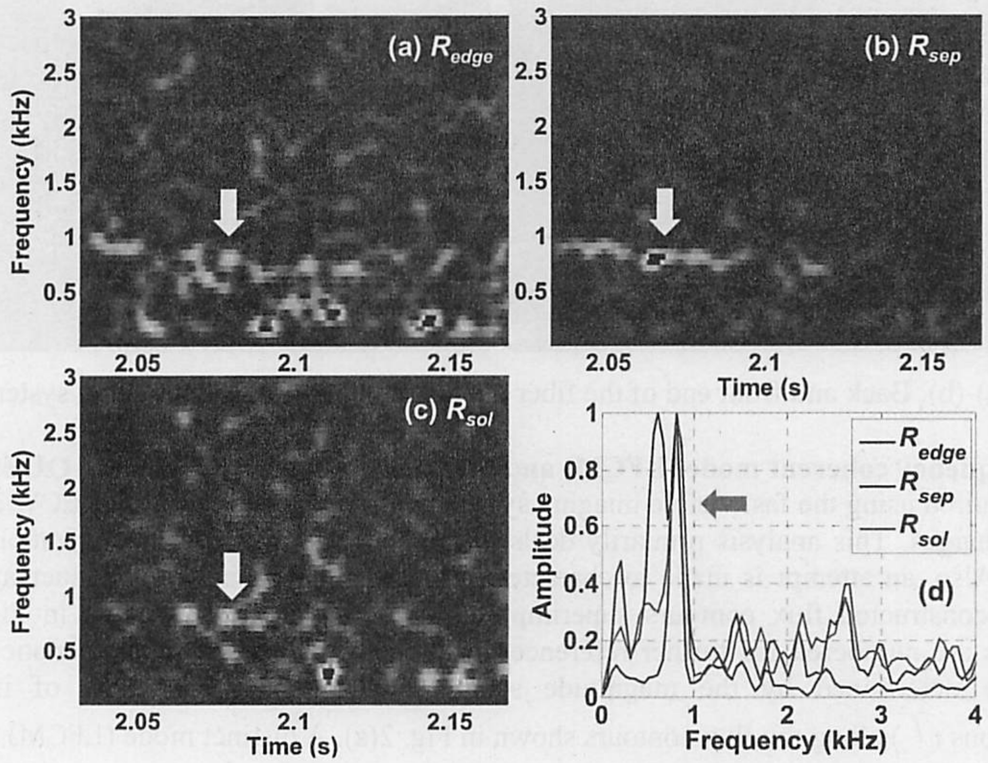

Fig. 3(a)-(c): spectrograms for 150 ms at R_{edge}, R_{sep} and R_{sol} respectively; (d): Fourier amplitude of the coherent mode. The mode is shown with bold arrows in all the panels.

Wavenumber (along Z direction)-frequency (k_z-f) spectra for the two radial locations, one around the separatrix location (R_{sep}) and the other at the SOL (R_{sol}), are shown for shot #17633 in Fig. 4. At both R_{sep} and R_{sol}, k_z-f spectra shows a single lobe with $k_z > 0$ fluctuations, which means fluctuation (both broadband and LFCM) propagation is vertically upward in the ion diamagnetic drift direction (IDD). Hence the propagation direction is QUEST is similar to the features observed during L-mode in Alcator C-Mod discharges featuring L-H mode transitions and pure Ohmic L-mode plasmas in NSTX

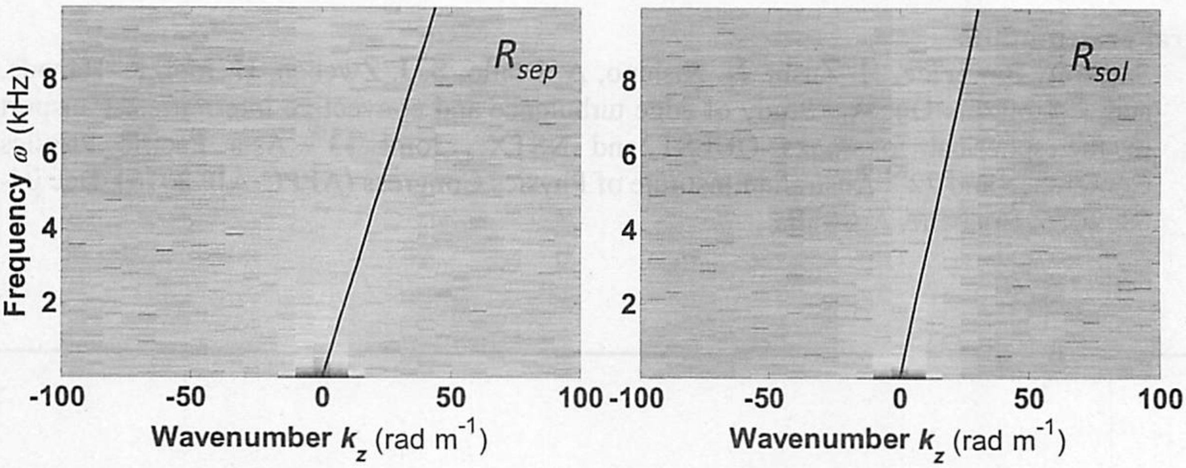

Fig. 4. Shot #17633; Turbulence characteristics at the separatrix and the SOL in poloidal wavenumber and frequency (k_z-f) space. $k_z > 0$ corresponds to propagation vertically up, in the IDD.

The measured k_z-f spectra can be fitted by a single linear dispersion relation ($f = v(k)k / 2\pi$) up to moderate range of frequencies (~6 kHz) such that poloidal $v_{ph} = v_g$ for a broad range, as shown in Fig. 4. The black solid lines at the edge and SOL panels represent the linear fit and hence phase velocity along Z can be calculated. For the representative shot #17633, velocity along Z at the separatrix and SOL are 1.4 km s^{-1} and 2 km s^{-1} respectively.

Summary of the analysis done

Low frequency coherent edge fluctuations at 0.85 ± 0.15 kHz, localized radially with long range poloidal correlation, are observed in the unique confinement scenario of IPN in QUEST. This low frequency mode is excited after 10~20 ms of ECW injection in Ohmic plasmas and the change in confinement regime from IL to IPN. After a while the mode is extinguished even though ECW injection is continued. The lifetime of the mode varies from shot to shot. This mode is in the frequency range of generic GAM, however, the exact characterization is not possible from the present dataset and will be attempted in future. In the IPN configuration, the n_e profile steepens at the edge. Enhanced cross-field transport, poloidal flow and increase in fluctuation level are also observed.

Acknowledgements

I gratefully acknowledge the support of my PhD supervisor Prof. H. Zushi, my collaborator Dr. T. Onchi, Prof. H. Idei and all other AFRC staff during the tenure of this collaborative effort. This work was supported by the International Joint Research Program of Research Institute for Applied Mechanics (RIAM), Kyushu University.

Publications and presentations regarding this collaboration during 2016-17.

Publications:

1. **Santanu Banerjee,** H Zushi, N Nishino, K Mishra, Y Mahira, S Tashima, A Ejiri, T Yamaguchi, T Onchi, Y Nagashima, K Hanada, K Nakamura, H Idei, M Hasegawa, A Fujisawa, A Kuzmin, K Matsuoka, "Observation of an edge coherent mode and poloidal flow in the electron cyclotron wave induced high βp plasma in QUEST", Phys. Plasmas **23**, 082507 (2016).

Oral presentation:

1. **Santanu Banerjee,** H. Zushi, N. Nishino, A. Diallo, S. J. Zweben, H. Idei, K. Hanada, and T. Stoltzfus-Dueck, "Study of edge turbulence and convective intermittent transport in the spherical tokamaks QUEST and NSTX", Joint 13th Asia Pacific Physics Conference and 22nd Australian Institute of Physics Congress (APPC-AIP 2016), Dec 04-08, 2016, Brisbane, Australia

国際化推進共同研究概要

No. 6

タ イ ト ル:　Plasma start-up and sustainment in spherical tokamak configuration by RF

研究代表者:　SHEVCHENKO, Vladimir

所内世話人:　出射 浩

実 施 期 間:　2017 年 2 月 2 日 ～ 2 月 3 日

研 究 概 要:　平成29年2月2−3日の 2 日間に国際 WS を開催した。欧州から 1 名、米国から 3 名、国内の学外者が 3 名と応研関係者の参加があった。主に QUEST の電子バーンシュタイン波加熱・電流駆動に関する課題を議論した。QUEST 実験の最近の進展に加え、国内外実験の進展、新たな運動論的理論・シミュレーション解析などが議論された。特に今年度は、NSTX 実験での電子サイクロトロン加熱・電流駆動による非誘導プラズマ電流立ち上げの発表があり、QUEST での非誘導プラズマ電流立ち上げに関する共同研究が広く議論されてた。また、両装置で展開されている CHI 実験についても議論された。

RF-only ST plasma confinement, sustainment, and interactions with wall materials

Hiroshi Idei

Vladimir Shevchenko

A program of the Workshop which was held on 2-3 February and was as following:

2-Febrary

AM

9:30-

Vladimir Shevchenko / Hanada

WS purpose and agenda

9:40-10:30

Vladimir Shevchenko

ST40: Status and Prospects

10:30-11:20

Masayuki Ono

An overview and motivation for the NSTX-U start-up research program

11:20-11:30

Coffee Break

11:30-12:10

Kazuaki Hanada

Progress and Plans of QUEST Experiments

12:10-12:40

Kengoh Kuroda

CHI Experiments in QUEST.

PM

14:00-14:45

Hitoshi Tanaka

Recent results on EBW experiments in LATE

14:45-15:35

Yuichi Takase

Plasma Current Start-up and Ramp-up by LHW on TST-2

15:35-16:05

B. Roidl

Equilibrium reconstruction using a MHD-particle hybrid code to study LHW driven ST plasmas

QUEST Machine Tour

3-Febrary

AM

9:30-10:20

Gary Taylor

Non-Inductive Plasma Current Start-up, Ramp-up and Sustainment in NSTX-U

10:20-10:50

Nicola Bertelli

PPPL Code Capabilities for ECH and EBW

10:50-11:00

Coffee Break

11:00-11:30

Ryota Yoneda

GENRAY-code analysis for ECH/EBW experiments in QUEST

11:30-12:00

Atsushi Fukuyama

Modeling of EC heating and current drive in QUEST plasmas

PM (Drafting of proposals for experiments, diagnosis, and analysis)

14:00- All Suggested focus and output for this joint drafting session

16:30- Vladimir Shevchenko

Summary

The presentation summaries are as following:

Vladimir Shevchenko, **S40: Status and Prospects**

Recent advances in the development of high temperature superconductors (HTS) and encouraging recent results on a strong favourable dependence of electron transport on higher toroidal field (TF) in Spherical Tokamaks (ST) open new prospects for a high field ST as a compact fusion reactor. A compact ST is also a promising candidate for an intense and efficient neutron source.

Overview of the ST40 project was presented. Main objectives of the project, parameters of the tokamak, physics program issues are described and physics and engineering challenges of this device are discussed. The present status of the construction is reported. Discussions of physics issues included plasma formation using merging-compression. A set of ST40 diagnostics for machine engineering parameters and specific for plasma were discussed in details. In particular, fast visible and infrared, visible and UV spectroscopy, soft X-ray multi-foil tomography, magnetics and dual color interferometry are the diagnostics to be installed on the machine for the first plasma operation.

The demonstration of reliable steady state operations in a compact ST even at the level of a few MW Fusion output as a first step will significantly advance not only the mainstream Fusion for the Energy research, but also the commercial exploitation of Fusion and will be an important step on development of the Fusion Energy.

Masayuki Ono, **An overview and motivation for the NSTX-U start-up research program**

National Spherical Torus Experiment Upgrade is a low-aspect-ratio (A ~ 1.6 – 1.8) spherical tokamak facility at PPPL with B_T ~ 1 T, I_p ~ 2 MA, P_{NBI} ~ 14 MW, and P_{RF} ~ 4 MW. The unique operating regimes of NSTX-U can contribute to important physics issues for ITER operations and beyond. With doubling of magnetic fields and heating power compared to NSTX, NSTX-U is designed to attain fully non-inductive operations at high beta needed for future ST reactors. Highlights from the first NSTX-U campaign in 2016 and longer-term goals were presented. The NSTX-U initial operation produced H-mode during the first two weeks of operations. The plasma stored energy of up to ~ 330 kJ was obtained matching the previous record of NSTX at ~ 1 MA. The NSTX-U team has also performed preliminary design work of the ST-based fusion nuclear science facility (FNSF) and the pilot plant. A relatively compact FNSF design with R = 1.7 m was developed which can be tritium self-sufficient. A pilot plant design using high temperature superconductor TF coil was developed where it is possible to produced net electric and tritium generation with R = 3 m. Perhaps the most crucial research program is the plasma start-up and current ramp-up. The NSTX-U will be testing the coaxial helicity injection start-up as well as the electron cyclotron heating and current drive heating. The ECH system plans to use the 1.5 – 2 MW 28 GHz gyrotron being developed by the Tsukuba University. Once the plasma temperature is sufficiently high well over 100 eV, the HHFW heating and current drive can be applied for up to 4 MW to ramp-up the plasma current to ~ 400 kA at sufficiently high temperature (~ few keV) and high density (~ few x $10^{19}m^{-3}$) for the NBI target. The 14 MW NBI can be then applied to further ramp-up the current toward 1 MA. The ultimate goal of NSTX-U is to start-up and sustain high [illegible]plasma fully non-inductively without use of central solenoid.

Kazuaki Hanada, **Progress and Plans of QUEST Experiments**

Introduction of the QUEST project is presented. QUEST (Q-shu Univ. Experiment with Steady state spherical tokamak) is a middle sized spherical tokamak (ST), which is R~0.64m, a~0.4m, BT (R=0.6m) <= 0.25T. Main plasma heating has been done by RF of 2.45GHz (<50kW), 8.2GHz (<250kW), and 28 GHz (<350k). In 2017, we can successfully demonstrate a 6 hour discharge of 3kA with last closed flux surface using the hot wall set at 473K. Previously the plasma duration was limited by uncontrollability of plasma density because of accumulation of fuel hydrogen surrounding a main plasma. The accumulation is caused by reduction of wall pumping rate (wall saturation) and transition of the wall property from fuel sink to source. We tried to decrease the hot wall temperature from 473K to 393K to enhance the wall pumping capability, but 2 hour14min met the limitation on plasma duration. We try to invert toroidal field direction, which means ion drift direction is also inverted from top to bottom, therefore fuel hydrogen is likely to accumulate in the lower area of the vacuum vessel, and three cryo-pumps located behand a CHI electrode is expected to work more efficient. As the result, H_2 pressure in the plasma chamber could keep constant during the 6 hours discharge in spite of growing of H_2 pressure behind the CHI electrode. It clearly indicates that fuel hydrogen is gradually compressed to a space behind the CHI electrode during the discharge. It should be investigated the reason, because the neutral particle compression may play an essential role in steady state operation of tokamak type devices.

Kengoh Kuroda, **CHI Experiments in QUEST.**

We installed CHI electrodes and other systems on QUEST.

Result of the first experiment:

Reliable gas breakdown was observed in the injector configuration, PF35-2; -1.5kA, PF26; 0.015kA and PF17; 0.3kA with 8.2GHz ECH.

Plasma images imply that the plasma is expanded after breakdown until about 30kA toroidal plasma current drives.

Future work:

Research of expanding and maintaining conditions of the CHI plasma

To do it, PF coil system should be modified for the vertically asymmetrical configuration and the measurement system have to be improved.

Hitoshi Tanaka, **Recent results on EBW experiments in LATE**

H. Tanaka, M. Uchida, Y. Nozawa, T. Kawaharada, K. Takamatsu, A. Hoshino, D. Honda, D. Watanabe, Y. Omura, R. Kajita, Y. Sakai, H. Shirai, T. Maekawa

In the Low Aspect ratio Torus Experiment (LATE) device, highly overdense ST plasmas (5 ~ 7 times the cutoff density) are produced non-inductively with EBW mode-converted via O-X-B scheme both at 2.45 GHz and 5 GHz. When the fundamental ECR layer is located at the plasma core and the 2nd harmonic ECR layer is near the outboard vessel wall and the UHR layer is located between them ($R_\Omega < R_{UHR} < R_{2\Omega}$), that is the condition for excitation of EBW in the 1st propagation band, the electron density increases significantly and exceeds 5 ~ 7 times the plasma cutoff density. The central temperature is estimated to be ~100 eV by X-ray absorption method. High density may be attributed to high electron temperature due to bulk electron heating at the fundamental ECR at the center of the plasma. On the other hand, high energy electrons in the low field side of the last closed flux surface with energy of ~50 keV decreases significantly. This result may be explained by calculations of EBW absorption which show that the absorption by high energy electrons decreases at high electron density. When the 2nd harmonic ECR layer is located at the inboard side of the UHR layer, the density increases to not more than twice the cutoff density. This may be attributed to the fact that in ST plasma due to low aspect ratio, the 2nd harmonic ECR layer is difficult to be located in the core region and efficient bulk heating is not expected. These phenomena are observed at two different microwave frequencies (2.45 GHz & 5 GHz) by adjusting the position of the ECR layer nearly the same.

Yuichi Takase, Plasma Current Start-up and Ramp-up by LHW on TST-2

ST plasma initiation and I_p ramp-up by LHW were demonstrated. ICC, grill, and CCC (outboard-/top-launch) antennas were used. Similar I_p was obtained for similar B_v and P_{RF}, but highest η_{CD} was obtained with the CCC antenna. Using the grill antenna (with variable $n_{||}$ spectrum), the highest I_p was obtained for $1.5 < n_{||} < 4.5$. A new type of antenna, capacitively-coupled combline (CCC) antenna was developed for LHCD. I_p ramp-up to 25 kA was achieved at higher B_t and with the installation of top/bottom limiters. The observed HX co/ctr asymmetry indicated higher directionality than grill antenna, and HX/SX profiles indicated deterioration of accessibility as $I_p \uparrow$, $n_e \uparrow$, and $B_t \downarrow$. A more reliable start-up was obtained with the outboard-launch CCC antenna, while it is expected that the top-launch CCC antenna would perform better for I_p ramp-up. A comparison of bulk electron pressure measured by Thomson scattering and the total pressure obtained by equilibrium reconstruction indicate that the pressure is dominated by the energetic electron contribution. An attempt is being made to characterize this situation using the MHD-particle hybrid code MEGA. The observed density limit appears to be lower than the accessibility limit, and may be caused by SOL losses. Under the present condition, energetic electron orbit losses are important because of low I_p and low n_e. Up to an order of magnitude improvement in $\eta_{CD} \equiv \bar{n}_e I_p R / P_{RF}$ may be possible at higher I_p. In order to achieve higher I_p, a top-launch CCC antenna was installed to improve wave propagation and single-pass absorption. Since the maximum I_p depends strongly on B_t, a TF coil power supply upgrade, aiming for $B_t = 0.3$ T and $I_p \cong 0.1$ MA for > 0.1 s is planned. In addition, a microwave scattering system is being developed to measure the LHW directly.

B. Roidl, Equilibrium reconstruction using a MHD-particle hybrid code to study LHW driven ST plasmas

MEGA is a hybrid fluid-particle code which is used to study TST-2 LH-driven plasmas, where energetic electrons play dominant roles in equilibrium, current, and losses. The current primary goals are, among others, reproduction of a self-consistent equilibrium, consistent with experimental and theoretical results and investigating the role of choice of the distribution function. Right now, bi-Maxwellian, semi-empirical distribution function and profiles obtained from CQL3D are applied. MEGA yields a self-consistent equilibrium for peaked density profile which are similar to those obtained from reference results of Ishida et al.. To investigate arbitrary distribution functions being mapped on the energetic electrons, a reduced version of MEGA was developed to efficiently solve induction, high-frequency Ampere's law to obtain the matching magnetic field configurations. First results suggest that hollow current distribution profiles are relaxed and fast electrons migrate to the plasma center. Particle orbits have to be studied carefully to link the impact of initial distribution function on equilibrium, current drive and losses.

Gary Taylor, **Non-Inductive Plasma Current Start-up, Ramp-up and Sustainment in NSTX-U**

- NSTX-U research has a goal to develop a fully non-inductive plasma.
- Initially NSTX-U will develop non-inductive start-up, ramp-up and plasma sustainment separately, using a combination of coaxial helicity injection, fast wave heating and neutral beam injection.
- There is a long-term plan to add 28 GHz electron heating system to support non-inductive start-up.
- Simulations predict 28 GHz electron heating can rapidly heat coaxial helicity injection plasmas.
- Simulations also predict a positive synergy between 28 GHz electron heating and 30 MHz fast wave heating during start-up and plasma current ramp-up.

Nicola Bertelli, **PPPL Code Capabilities for ECH and EBW**

The main EC/EBW codes used in PPPL have been presented:

1) GENRAY: a ray tracing code used for several different RF wave regimes (EC, EBW, LH, IC).

2) TORBEAM: a beam tracing code for EC waves. It takes into account refraction and diffraction effects.

3) CQL3D: solves a Bounce-Averaged Fokker-Planck equation. Used for several different RF wave regimes (EC, LH, IC).

4) TRANSP: time-dependent transport solver for interpretive or predictive analysis in tokamak plasmas. It evolves self-consistently the equilibrium, the heating and current drive sources and the pressure profiles, within the limits of the physics model used.

Ryota Yoneda, **GENRAY-code analysis for ECH/EBW experiments in QUEST**

Summary:

- Scenario for high-density plasma via EBW was shown
- X-mode propagation to UHR was calculated by changing density profile
- O-X-B optimized launch was also investigated
- Waveguide test bench is planned to prevent ECR breakdown inside

Future Work:

- Open issue remains on EBW propagation (HFS X-B) because of numerical error in the code - with N. BERTELLI
- Evaluation of current drive efficiency with Fokker-Plank code - with N. BERTELLI
- Preparation for HFS X-mode injection

Atsushi Fukuyama, **Modeling of EC heating and current drive in QUEST plasmas**

We are developing 2D kinetic full wave modelling of EC heating of plasmas employing an integral form of dielectric tensor.

1D kinetic full wave analysis of O-X-B is now available using integral form of DE, and extension to 2D is under way.

2D analysis with conventional collisional cold plasma models describe the O-X mode conversion of EC waves in toroidal plasmas, though electron Bernstein waves and cyclotron resonance absorption are not included.

For self-consistent analysis including the modification of velocity distribution function, formulation of integral form of quasi-linear velocity diffusion and dielectric tensor for arbitrary velocity distribution function is required.

PIC simulation of ECW heating and plasma production is also under way.

Several Proposals were submitted:

Gary Taylor:

- NSTX-U research team proposes collaborating with the QUEST team on plasma start-up experiments that use 8.5 and 28 GHz heating and/or CHI that will help to guide planning for the NSTX-U non-inductive start-up experiments.

Nicola Bertelli:

All the codes above can be used for EC/EBW studies on QUEST plasmas under the collaboration between NSTX-U and QUEST teams.

Tasks:

1) Try to fix GENRAY for X-B high field launcher in QUEST plasmas

a. Cold dispersion relation vs. warm dispersion relation

2) Include high energy electron population in GENRAY simulation for EC studies on QUEST plasmas

国際化推進共同研究概要

No.7

タ イ ト ル:　Towards high mode purity in ECRH transmission lines for ITER

研究代表者:　KASPAREK, Walter, Hermann

所内世話人:　出射 浩

実 施 期 間:　2017 年 3 月 12 日 ～ 3 月 14 日

研 究 概 要:　平成29年3月12−14日、Dr. Kasparek, 平成29年3月21−29日、Dr. Lechte を迎えて、共同研究を実施。Kasparek 氏とは、準光学高速伝送路スイッチの新規セットアップを行い、初期動作不良の解決を図った。また、動作詳細について議論した。また、2周波数動作の矩形導波管高速伝送路スイッチ、パワーモニータシステム動作を議論した。Lechte 氏とは、準光学高速伝送路スイッチの実際の動作に向けて、テーパー管の設置、スイッチ導入部の設置などを行い、さらにスイッチ実動作に向けた作業を進めた。

Towards high mode purity in ECRH transmission lines for ITER

Applicant: Walter Kasparek (male, 64)

Institute of Interfacial Process Engineering and Plasma Technology (IGVP)

Electron Cyclotron Heating (ECH) using high power millimetre waves is an attractive method for plasma production, auxiliary heating, and current drive in a nuclear fusion research. Accordingly, the ECH system at the International Thermonuclear Experimental Reactor (ITER) will have a total injected power of 20 MW at an operating frequency of 170 GHz. For 20 MW injections to the plasma, 24 high-power gyrotron oscillators of 14 MW each will be used. The output beam from each gyrotron oscillator is led to a Circular Corrugated (CC) waveguide line, and transmitted as an HE_{11} mode of a main eigen-mode in the waveguide. 24 CC waveguide lines will be prepared to transmit the total 20 MW power. Excitation of unwanted higher-order modes in the oversized waveguide causes many problems such as excessive transmission loss, arcing, thermal overload of components due to stray radiation, and finally deviations of the launched beams from the nominal direction. In earlier works, transmission losses in the ITER ECH system due to higher-order mode excitation in misaligned components have been estimated, and the impact on the launched beams was studied.

This collaboration has been established to excite and transmit the high-purity main HE_{11} mode under monitoring and controlling of the transmitted wrong modes for the CW high power application.

The International Joint Research team consisted (besides the applicant) of Hiroshi Idei (RIAM, Kyushu), Keishi Sakamoto (JAEA Naka), Takashi Shimozuma (NIFS, Toki), Richard Temkin (MIT PSFC Cambridge), Michael Shapiro (MIT PSFC Cambridge), Alexander Zach (IGVP Stuttgart), Carsten Lechte (IGVP Stuttgart), and Burkhard Plaum (IGVP Stuttgart).

In-situ mode analyzer

For reliable operation of high-power cw transmission systems, detection and mitigation of higher-order modes is of major importance. Therefore, the development of in-situ mode analyzers, which are integrated into the surface of mitre-bend mirrors continued. All developments are made in view of application in cooled mirrors, which can tolerate heat loads and thermal stress typical for high-power CW operation.

An advanced 6-mode analyzer was built at RIAM, confirming the detectability of various wrong modes in HE_{11} waveguides.

A new prototype mitre bend mirror with 7 ports is being manufactured featuring two kinds of mode-matched hole array couplers. One suppresses the coupling of LP_{0n} and asymmetric higher-order modes with high rejection. The other kind isolates the LP_{02} mode by 1st-order hole coupling towards a second channel. Calculations on the suppression of wrong modes other than the modes to be detected have been performed; especially for HE_{11}, high discrimination against higher-order modes is obtained. As the manufacturing of the device is performed in several steps, fabrication is finished only in March 2017; therefore, experimental results are expected only later.

For the resonator control of 2-frequency diplexers (see below), measurement of the output power in the channels is required. These couplers employ leaky wave antennas basically designed for the lower of the two frequencies; in addition, an amplitude grating overlaid on the hole coupling structure produces a receiver lobe which is sensitive of the higher frequency. Thus the couplers provide power monitor signals at two frequencies with a relatively high suppression of interference from stray radiation. Fig. 1 shows a design of the coupler for the frequency combination of 105/140 GHz with the calculated receiver pattern, and the measured patterns at 140 GHz for both polarization planes taken from a prototype. The design based on the present result can be taken with slight improvements for the final design of a 110/138 GHz power monitor.

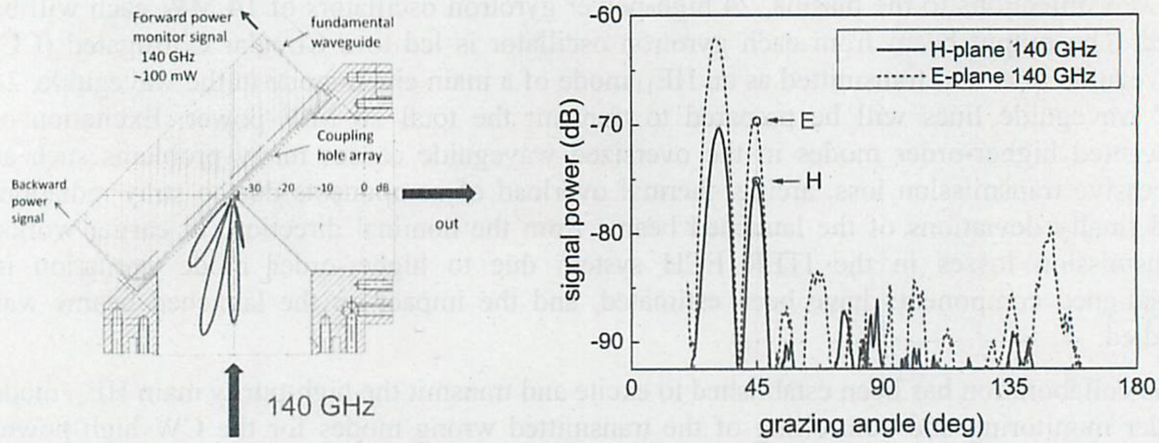

Fig. 1: Left: Design of a 2-frequency power monitor in a mitre bend, with the 140-GHz antenna pattern of the coupling hole array. For the lower frequency, (105 GHz), the main lobe points into the direction of the incoming HE11 mode. Right: Measured antenna patterns of the coupler at 140 GHz, H-plane (solid) and E-plane (dashed).

HE_{11} Mode Generator

The generation of high-purity HE_{11} modes, as well as the generation of modes, which are typically excited in misaligned waveguides is required for comprehensive studies on mode excitation, and effects of misalignment in the Circular Corrugated (CC) waveguide transmission. Therefore, work has continued aiming at the generation of a pure HE_{11} mode on one hand, and the generation of pure high-order modes in a folded 170 GHz resonator system on the other. This resonator is excited via a 63.5 mm waveguide; depending on the alignment, the main mode (HE_{11} or LP_{01}) or spurious modes are excited.

Several modes, especially LP_{01}, and higher-order modes, LP_{11}^{even} and LP_{11}^{odd} modes could be excited with good purity. Also the mode LP_{21}^{even}, which is a marker for an elliptical beam coupled to the corrugated waveguide, could be found. However, instead of the marker mode for a beam of wrong diameter at the entrance of the waveguide, only modes like the LP_{12} was excited.

To get a better performance, the resonator tuning system was upgraded, and the coupling grating for the resonator was optimized. Further tests are planned at the facilities at RIAM, to take advantage from the good test equipment at 170 GHz. For this purpose, the ring resonator was recently sent to RIAM. Fig. 2 shows the ring resonator consisting of two phase-inverting mirrors, two plane coupling gratings, and the matching mirrors to the waveguide feeds.

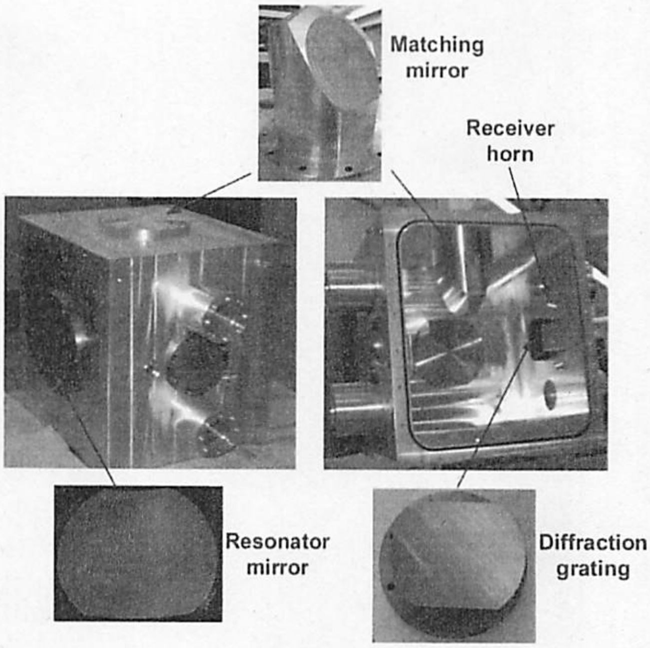

Fig. 2: Photographs of the ring resonator and its components. The top plate of the box carrying the upper resonator mirror with tuning system is taken away.

High power diplexer for ECH applications

In support to the RIAM development of a two-frequency diplexer based on Talbot splitters, especially the frequency and angular dependence of the characteristics of the corrugations for the square corrugated waveguides (SCW) in the splitters have been investigated further. These investigations are performed mainly at the frequency of 140 GHz, as for this frequency, a remote-steering launcher with 1 MW CW capability was developed for the stellarator W7-X.

Results found before were mostly confirmed: With the electro-forming technique, corrugation profiles can be machined with sufficient precision, such that for the nominal frequency and propagation angle practically no cross-polarisation occurs. However, the phase shift of the corrugation depends on frequency, and - to a less extent – on the splitter angle. Concerning the loss of the SCWs, this has practically no effect. However, for arbitrary polarisation in the waveguide, phase shifts between the polarisation components parallel and perpendicular to the propagation plane occur. For the design of a frequency-independent diplexer, this fact requires a general design of the corrugation profile for the high frequency, and the optimization of the geometry of the Talbot splitters such that phase shifts between the polarisation components at the lower frequency are multiples of 360°. For this condition, dual-frequency operation (138 / 110 GHz) under arbitrary polarization is possible.

Another feature of the corrugations, which was found during the investigations, is an absolute phase shift of the corrugated wall, especially for low-loss corrugations with broad grooves, as the TE polarisation component penetrates somewhat into the corrugations. Therefore, the effective width of the waveguide appears larger. This has to be taken into account when optimizing the length of the splitter waveguides (which depends on the cross-section of the waveguide). Calculations for the investigated corrugations with depth of 0.612 mm yield an effective broadening of the waveguide of $\Delta a \geq 200$ µm; measurements gave ≈ 140 µm. These facts require an integrated optimization of the parameters of the Talbot splitters for the two-frequency diplexer.

Note that high-power tests of the 140 GHz SCWs are planned within the next operation campaign at W7-X, which is going to start in autumn 2017.

国際化推進共同研究概要

No.8

タ イ ト ル:　Investigation of rotation reversal near resonance layer in a EC heated/driven plasma

研究代表者:　MISHRA, KISHORE, KANTI

所内世話人:　出射 浩

実 施 期 間:　年 月 日 ～ 月 日

研 究 概 要:　QUESTでは、Electron Cyclotron (EC)波の注入を伴うオーミック加熱プラズマの固有回転方向反転が観測されている。トロイダル方向およびポロイダル方向に沿って流れる多数の不純物イオンのドップラーシフトを測定するために、25視線の可視分光計を使用し、予備実験を行った。プラズマ配位はEC注入前後で異なる平衡状態にあり、オーミック状態ではInboard Limiter(IL)配位にあり、EC注入直後では高磁場側に自然発展したダイバータ構造が現れるインボードポロイダルヌル(IPN)平衡になる。

A short report on, “Study of EC induced intrinsic rotation in QUEST tokamak”

By Kishore Mishra
Institute for Plasma Research, Gandhinagar, India, 382 428

Intrinsic rotation reversal in Ohmic plasma with injection of EC waves is studied in QUEST tokamak. A 25 viewing chord based visible spectrometer is used to measure Doppler shift of majority and impurity ions flowing along toroidal and poloidal directions. Details of the measurement system and some preliminary experimental findings are described in the following sections.

Rotation measurement diagnostics on QUEST

A 25 channel optic fiber bundle is placed at the outboard side viewport on a 3 dimensional stage with a precision rotary stage to align the fiber in horizontal (toroidal rotation measurement) or vertical (poloidal rotation measurement) direction. A schematic of the setup is shown in figure 1. A one meter focal length Czerny-Turner scanning spectrometer (AM-510, Princeton Instruments) with 1800 grooves/mm plane grating is used to disperse wavelength and measure the required emission spectrum. The spectrometer along with the thermoelectric cooled 2048 × 512 pixel CCD is placed away from the vessel to minimize the effect of Hard X-ray emissions from the plasma. Additional 10 cm thick lead blocks are used to cover the CCD, which completely eliminated the Hard X-ray bursts effect on the CCD. The spectrometer system along with a 45 μ slit offered an excellent spectral resolution of better than 0.02 nm in the range of 185 to 800 nm.

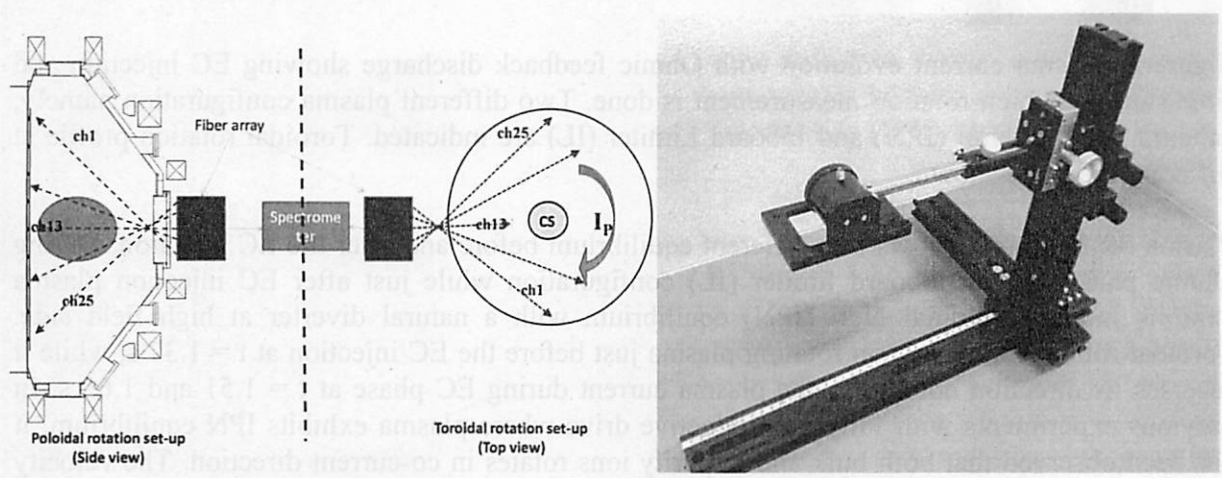

Figure-1: A sketch schematic of 25 channel optic fiber system with lines of sight. A new 3-axis positioning system with a toroidal/poloidal circular flip holder for positioning fiber system.

A new 3 axis positioning system with a toroidal and poloidal flip facility for the fiber is integrated to the system, where the fiber can be positioned in horizontal/vertical position to measure toroidal/poloidal rotation respectively. This new holder helps to reduce time and enhance accuracy in repositioning of fiber array for different set of measurements across different experimental campaign.

Spontaneous rotation reversal with EC injection

Ohmic target plasma with feedback to OT transformer is formed at a fixed plasma current Ip = 30 kA. EC power of ~ 100 kW is injected to this plasma with suitable support for vertical magnetic field. Spectral data from CIII and HeII emission are captured before and after EC injection with the help of 25 channel fiber array and spectrometer at 25 ms exposure time. The time integrated toroidal rotation profile is evaluated during these times. An example of time trace of Ip with rotation measurement timings and corresponding rotation profile are shown in figure-2.

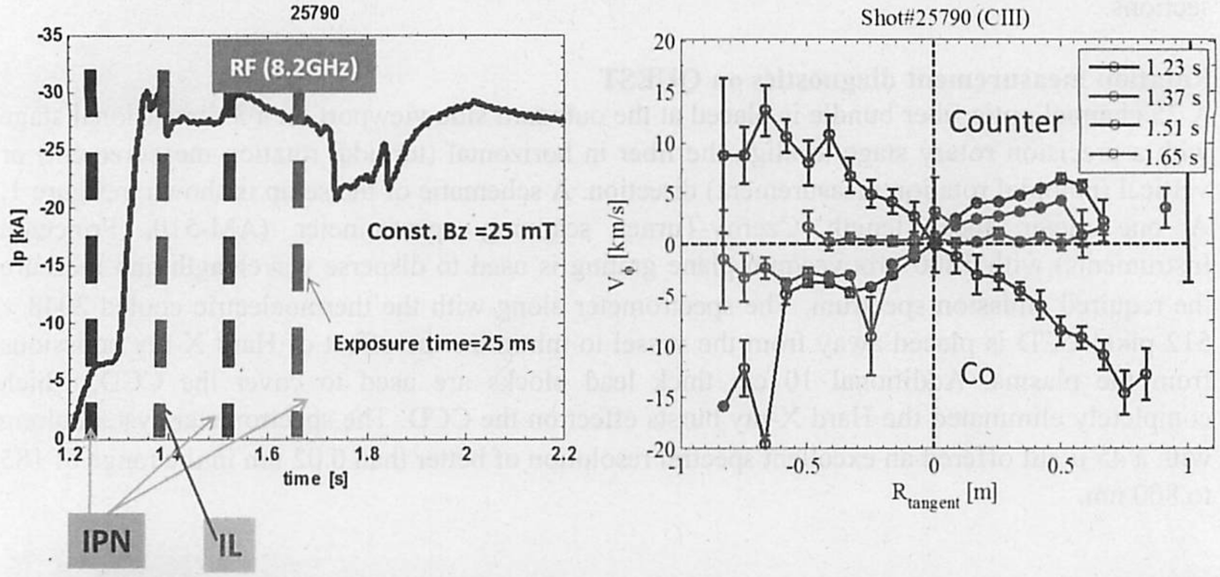

Figure-2: Plasma current evolution with Ohmic feedback discharge showing EC injection and time slices at which rotation measurement is done. Two different plasma configuration namely, Inboard Poloidal Null (IPN) and Inboard Limiter (IL) are indicated. Toroidal rotation profile at different times are shown.

Plasma configuration shows two different equilibrium before and after the EC injection. During Ohmic phase it is in inboard limiter (IL) configuration while just after EC injection plasma exhibits Inboard Poloidal Null (IPN) equilibrium with a natural diverter at high field side. Toroidal rotation shows a non rotating plasma just before the EC injection at t = 1.37 s, while it reverses its direction counter to the plasma current during EC phase at t = 1.51 and 1.65 s. In previous experiments with fully non-inductive drive where plasma exhibits IPN equilibrium, it has been observed that both bulk and impurity ions rotates in co-current direction. The velocity scales almost linearly with applied Bz and Ip. However, IPN equilibrium with EC injection in this case shows a counter driven rotation. This spontaneous rotation reversal remains to be investigated.

New 14 channel spectrometer system

A new 14 channel viewing chord optic fiber based spectrometer system (schematic shown in figure 3) is integrated with QUEST by a joint collaborative work with Kyoto University (Prof

Shikama). This is a 1m grating spectrometer with 1800 grooves/mm gives a spectral wavelength resolution of < 50 pm and a wavelength measurement accuracy better than 0.5 pm corresponding to 0.3 km/s velocity. This added system will complement rotation measurement diagnostics on QUEST and help us to investigate the plasma better.

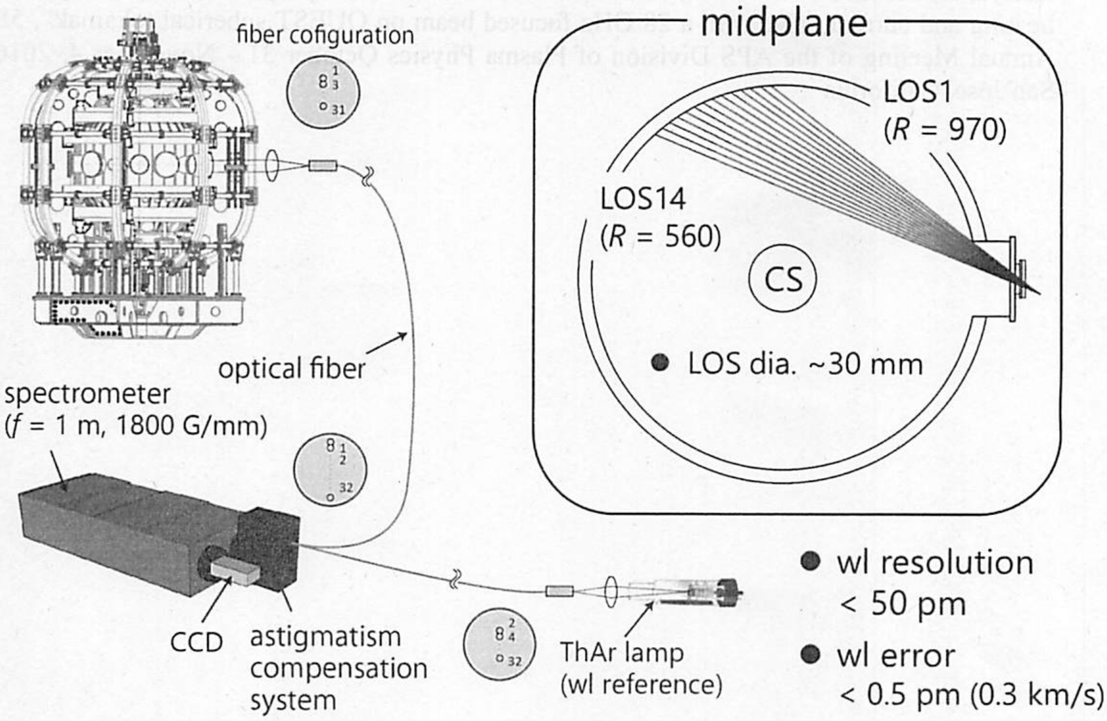

Figure 3: A schematic of 14 fiber based new spectrometer system for rotation measurement from one toroidal line of sight.

Acknowledgement:
This research is carried out under international joint research grant from Research Institute for Applied Mechanics, Kyushu University. The author gratefully acknowledges active support and guidance of RIAM collaborators Prof Hiroshi Idei, Prof. Hideki Zushi, Dr. Takumi Onchi and fruitful discussions with Dr. Santanu Banerjee. Part of this research is with a collaboration with Prof. Shikama of Kyoto University, Kyoto.

Publications and Presentations:

1. Santanu Banerjee, H. Zushi, N. Nishino, K. Mishra, Y. Mahira, S. Tashima, A. Ejiri, T. Yamaguchi, T. Onchi, Y. Nagashima, K. Hanada, K. Nakamura, H. Idei, M. Hasegawa, A. Fujisawa, A. Kuzmin and K. Matsuoka, “Observation of an edge coherent mode and poloidal flow in the electron cyclotron wave induced high plasma in QUEST”, Phys. of Plasmas 23, 082507 (2016)
2. Kishore Mishra, Hideki Zushi, Hiroshi Idei, Takumi Onchi, Makoto Hasegawa, Kazuki Hanada, and QUEST Team, " Origin and Evolution of Spontaneous Rotation in Plasma Under Different Magnetic Field Geometries in Tokamak QUEST", IEEE TRANSACTIONS ON PLASMA SCIENCE, VOL. 44, NO. 4, APRIL 2016 pp 441

3. Kishore Mishra, H. Zushi, H. Idei, T. Onchi and QUEST Team, "Self Organization of Plasma Equilibrium at Poloidal Beta limit", Oral Presentations at "Joint 13th Asia Pacific Physics Conference and 22nd Australia Institute of Physics Congress" (APPC 2016) December 4-8, 2016, Brisbane, Australia.
4. T. Onchi, H. Idei, M. Hasegawa, H. Ohwada, H. Zushi, K. Hanada and QUEST team, T. Kariya, K. Mishra, T. Shikama, "Non-inductive current built-up by local electron cyclotron heating and current drive with a 28 GHz focused beam on QUEST spherical tokamak", 58th Annual Meeting of the APS Division of Plasma Physics October 31 - November 4, 2016 • San Jose, California

国際化推進共同研究概要

No.9

タ イ ト ル:　Plasma Start-up Using CHI on QUEST

研究代表者:　NELSON, Brian, A

所内世話人:　花田　和明

実 施 期 間:　2016 年 12 月 3 日　~12 月 17 日

研究概要:　2016 年に QUEST で初めて同軸ヘリシティ入射(CHI)の実験が実施された。電源、電極、ガス入射システムすべてが予定通りに運転され、最大で 29kA のトロイダル電流が観測された。また、プラズマ着火の様子が高速カメラの映像で観測された。ECH の入射により再現性良くプラズマ点火が起こることが分かった。次年度は CHI による ST プラズマ形成に向けたポロイダル磁場の設定を行い、ST プラズマを形成することを目的とする、また、この CHI プラズマに28GHzの ECH を入射してプラズマ加熱が起こるかどうかを確認する。

Plasma Start-up Using CHI on QUEST *
2 February 2017

R. Raman[1], K. Kuroda[2], K. Hanada[2], T. Onchi[2], M. Hasegawa[2], M. Ono[3], T.R. Jarboe[1], B.A. Nelson[1], M. Nagata[4], O. Mitarai[5]

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Introduction

Methods for starting a plasma discharge in a spherical tokamak (ST) without reliance of the center solenoid are essential for the validity of the ST concept. These methods could also simplify and reduce the cost of tokamak-based systems and make them more economical by eliminating components that are not needed during steady-state operation. Coaxial Helicity Injection (CHI) for an ST, first developed on HIT-II at the Univ. of Washington, is the leading method adopted by NSTX-U to genrate the initial current during a planned full non-inductive current start-up and current ramp-up scenario. On QUEST, this method would be further developed using the unique all metal capability of QUEST, which is predicted to reduce low-z impurities. In addition, CHI on QUEST will develop a new configuration that is much more suited to ST-FNSF. CHI start-up on QUEST could be used to provide an alternate, and when combined with induction, higher current targets for RF current drive studies. There are a number of new and important studies that would be possible on QUEST. These are: (a) Benefits of high-power ECH for CHI discharge initiation and heating of CHI plasmas, (b) impact of an all metal wall configuration for reducing low-z impurities, and (c) development of a simpler electrode configuration that is much more suitable for a fusion reactor as described in the Reference [Raman, et al., Fusion Science & Technol., **68** (2015) Pg. 674]. All these objectives are well aligned with the long-term mission of QUEST to develop steady-state fusion reactor technologies. This document discusses the near-term plans for CHI studies on QUEST.

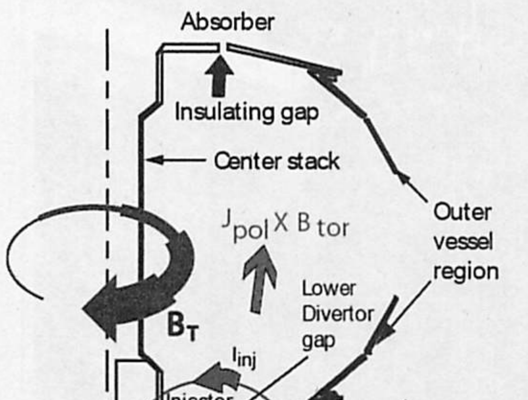

Figure 1: Layout of the transient CHI startup systems in NSTX. The blue circle is the poloidal injector flux produced by the lower divertor coils. This connects the two lower divertor plates, which are insulated. Gas is injected in the region below the divertor gap. On NSTX typically a 5 to 15mF capacitor bank charged up to 1.7kV is used to produce the injector current.

* We acknowledge helpful discussions with Prof. Zushi, Mr. Noda (V-Tech Limited) and Mr. Rogers (Univ. of Washington) and with other members of the QUEST Team.

Basic concept for CHI operation on QUEST

On QUEST, as briefly described in Fig. 2, a toroidal ring electrode is mounted on top of the existing lower divertor plate, and the electrode separated from the divertor plate using a toroidal alumina insulator. Magnetic flux generated by the lower divertor coils connect this electrode plate (the cathode) to the outer vessel (the anode). Gas is injected in the gap between these electrodes and a 20-30mF capacitor bank, charged up to 2kV is discharged across these electrodes to generate the CHI plasma.

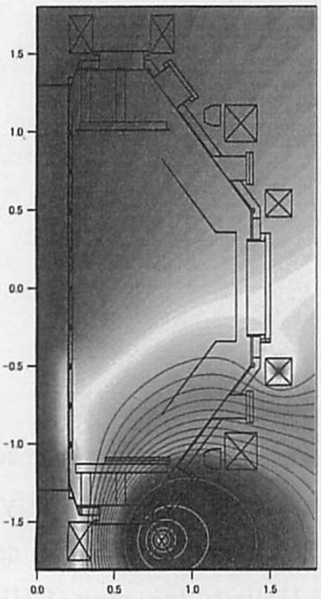

Figure 2: Typical magnetic surface at plasma production using CHI electrode. A continuous ring insulator electrically separates the cathode electrode from the lower divertor plate.

Progress with CHI on QUEST

During 2016, we were able to fully operate the newly installed transient CHI system on QUEST. This was completed in two parts. First, during March 2016, University of Washington personnel traveled to Kyushu University to test and commission the CHI capacitor bank, CHI gas injection systems and the hi-voltage snubber systems in a laboratory located near the QUEST ST. Several members of the QUEST Team helped with this initial commissioning work. The control system to operate these from the QUEST Control Room was also tested, and some hardware improvements for the capacitor bank and control system were identified. Between April and November the additional needed hardware was built, and some of these were shipped from the Universality of Washington to Kyushu University. QUEST personnel also relocated the capacitor bank to the QUEST Test Cell, built a protective cage around the capacitor bank [Fig. 3], and connected the capacitor bank to the three CHI current feed locations on QUEST. QUEST personnel also installed the two gas injection systems and attached the three high voltage snubber assemblies to the current feed locations on QUEST.

Figure 3: QUEST ST will test ECH heating of CHI in an all-metal configuration using a new CHI electrode configuration.

Then during December 2016, University of Washington personnel traveled to QUEST to fully commission the CHI system. The goals were to establish reliable gas breakdown and measure toroidal current generated by CHI on QUEST. Both these goals were successful.

The first week was spent commissioning the CHI capacitor bank and the two fast CHI gas injection systems, both of which were now attached to the QUEST machine. During these non-plasma tests remote control operation of both systems was tested, and the QUEST vessel was subjected to 600V for the first time. During the second week, reliable plasma breakdown was demonstrated for the first time in the new ST-FNSF relevant CHI electrode configuration used on QUEST. During these tests, the QUEST vessel was subjected to the full 2kV CHI voltage potential. This was followed by operation with increased capacitor bank energy using two switching ignitrons to generate measurable toroidal current.

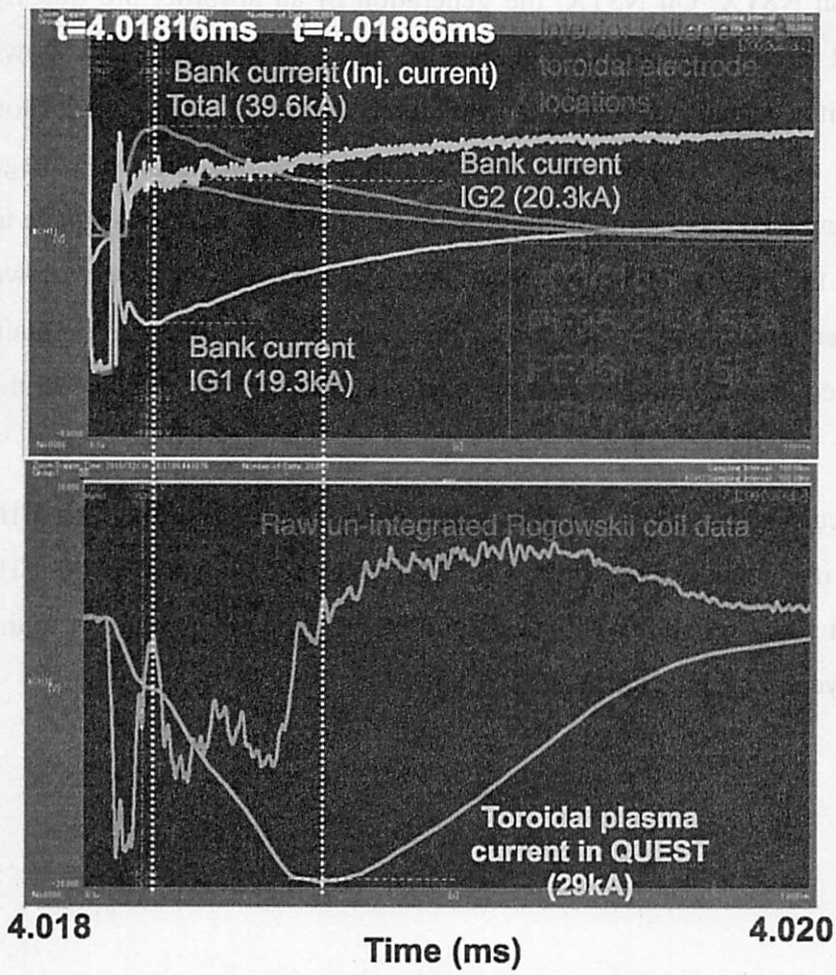

Figure 4: The top frame shows the individual current measured by the current sensors on the two ignitrons inside the CHI capacitor bank. These are about 19 and 20 kA each. The total current is measured by a third sensor, and it shows an injector current of 39kA. The green trace shows the voltage across the three injection electrodes, which initially reach the full charging voltage of 1800V, but rapidly drop to a lower value as breakdown occurs across the electrodes. This is accompanied by an increase in the injector current as the capacitor bank drives up to 39kA through the plasma load. The injected current begins to decay quite rapidly, but on a slower rate the toroidal current begins to grow, and reaches 29kA. The plasma current is numerically integrated using the raw un-integrated rogowski coil signal.

Fig. 4 shows an example of a discharge in which 29kA of toroidal current was generated. The top frame shows that a peak of 39kA of injector current is produced, and as this current begins to decay the toroidal CHI generated current increases and reaches 29kA (lower frame). In general, mid-plane camera images show the plasma detaching from the injector electrode and moving up towards the upper divertor. This is expected because the upper PF3-2 coil was connected in series to the PF5-2 injector coil and had a current polarity that attracted the CHI generated plasma. Plasma equilibrium control was not an objective for the 2016 CHI campaign, so this coil connection configuration was adequate for the 2016 experiments. It was also not possible to determine from the camera images if undesirable arcing (known as absorber arcing on NSTX) was occurring. However, a positive feature was that even if such arcing occurred, it did not stop the CHI plasma from forming. This was not the case on NSTX. On NSTX, the generation of an absorber arc quickly diverted all of the injected current through the undesirable absorber arc. The top frame also shows three overlaid fast voltage monitor signals measured at the three electrode current injection locations. The fact that they all overlay is good, as it means that noticeable toroidal voltage asymmetries do not exist in this new CHI configuration. In these experiments no effort was made to control the plasma equilibrium, as the primary objective was to demonstrate reliable gas breakdown and to make a useful plasma current measurement. As part of the work in 2017, we plan to obtain improved control of the other PF coils on QUEST to trap the CHI plasma near the midplane of the vessel and better diagnose it.

The magnitude of the toroidal current is similar to that achieved on NSTX during first CHI commissioning tests on NSTX in 1999. During over 150 CHI pulses on QUEST; the CHI engineering systems operated with 100% reliability with no issues either to the CHI system or to the QUEST system. This was a great year for CHI studies on QUEST.

国際化推進共同研究概要

No.10

タイトル: Electron Bernstein wave heating with XB mode conversion from low field side launch

研究代表者: Hwang , Yong-Seok

所内世話人: 花田 和明

実施期間: 年月日 ～月日

研究概要: 弱磁場側入射によるXBモード変換を経た電子バーンシュタイン加熱1次元のfull waveシミュレーションを用いてQUESTでの8.2GHzマイクロ波加熱におけるXBモードの変換効率についての解析を行った。電子バーンシュタイン波へのモード変換が生じれば第2高調波、もしくは基本波共鳴により効率的な加熱が行われる。しかし現状のQUEST実験のピーク電子密度 $2\times10^{18}/m^{3}$ ではモード変換効率は1%であり、ピーク電子密度 $5\times10^{18}/m^{3}$ では35%に上昇する。トロイダル磁場を増加させることは出来ないが、ポロイダル磁場配位のミラー比を上げることで密度は上昇させられると推測する。
今後の予定では適切の磁場配位での2.45GHz, 8.2GHz, 28GHzのマイクロ波加熱を組み合わせた場合における解析を行う予定である。

Th

Electron Bernstein wave heating with XB mode conversion from low field side launch

Hwang, Yong-Seok (Seoul National Univ., Korea)

Abstract

EBW is considered as one of the most important heating and current drive methods in Spherical Torus (ST). Mode conversion and tunneling efficiencies are critical to produce over-dense plasmas beyond density cutoffs that are relatively low in ST with low magnetic field strength. Over-dense plasma production via XB mode conversion with low field side (LFS) X mode injection has been successfully demonstrated beyond L cutoff in Versatile Experiment Spherical Torus (VEST) with low microwave frequency of 2.45 GHz. This simple scheme may be useful if demonstrated in QUEST with higher magnetic field and higher heating frequency. Mode conversion efficiency is estimated to be very low from one-dimensional full wave code with measured plasma density profiles in the present experimental setup. For sufficient mode conversion efficiency of >35%, electron Bernstein wave (EBW) system with 8.2 GHz from LFS X mode injection requires much higher plasma densities of $5 \times 10^{18}\ \#/m^3$ while plasma density of only $2 \times 10^{18}\ \#/m^3$ is generated in the present QUEST operation. Scenarios for efficient XB mode conversion with 8.2 GHz system are proposed by utilizing all available RF sources with appropriate poloidal field configurations. Experimental demonstration will be attempted in the next year's international joint research.

Introduction

We have been working on EBW heating in VEST at Seoul National University. Over-dense plasmas are generated for pre-ionization by applying 2.45 GHz microwave in X mode from LFS via direct XB mode conversion. Efficient penetration and mode conversion with LFS X mode injection are successfully explained with both Budden parameter analysis and numerical simulation with one-dimensional full wave code. [1,2] Even for the frequency of 8.2 GHz equipped in QUEST, significant XB mode conversion efficiency with LFS injection requires much steeper density gradient near UHR than that of the VEST at the low frequency of 2.45Hz. Sufficient density gradient for efficient XB mode conversion will be estimated with numerical analyses, and operation scenario will be developed to meet the requirement with additional high power heating system at the frequency of 28Ghz and appropriate magnetic field configuration.

Method

Considering different machine and plasma parameters of QUEST, mode conversion efficiencies for available RF systems are estimated with one-dimensional full wave simulation as a first step. For higher XB mode conversion efficiency in QUEST with the 8.2 GHz heating system, higher plasma density may be pursued for steep density gradient near UHR by providing trapped particle configuration (TPC) with high mirror ratio for better particle confinement.[3] Various combinations of PF coils will be compared for better mirror trapping. Superposition of all available heating powers of 2.45 GHz, 8.2 GHz and 28 GHz with an appropriate magnetic field configuration will be searched for the required steep density gradient.

Results

One-dimensional full wave simulation has been performed for the QUEST heating system at the frequency of 8.2 GHz. Experimental density profiles are utilized as a reference profile with various peak density values as shown in Fig. 1. Once it is mode converted, converted Bernstein wave can meet either second harmonic or fundamental electron cyclotron resonances, providing efficient heating without density cut-off. However, XB Mode conversion efficiency for the present experimental density value of $2 \times 10^{18}\ \#/m^3$ is estimated to be only 1% even with high edge pedestal as shown in Fig. 1. If the peak density can be increased to $5 \times 10^{18}\ \#/m^3$, this low efficiency can be increased up to 35%. Relatively low plasma density in QUEST may be due to the low absorption efficiency of the second harmonic resonance of the 28GHz heating system. Since toroidal field cannot be increased for the relocation of resonance layer, poloidal field configuration can be changed for particle confinement as well as heating. Trapped particle configurations with high mirror ratios from different sets of poloidal field coils as shown in Fig. 2 may provide possibility of increasing plasma density.

In steady state plasmas, XB mode conversion takes place mostly in the LFS edge region, where the magnetic field scale length is so long that its effect is not important in the mode conversion. However, in a low density pre-ionization or ramp-up phase, the UHR layer can exist near the high field side (HFS), and the magnetic field scale length becomes shorter, especially in ST, and can lead easier mode conversion as reported in VEST. [1] This may provide another possibility of XB mode conversion in QUEST by utilizing short magnetic field connection length at HFS, where the condition of the density scale length for optimal mode conversion can be relaxed significantly.

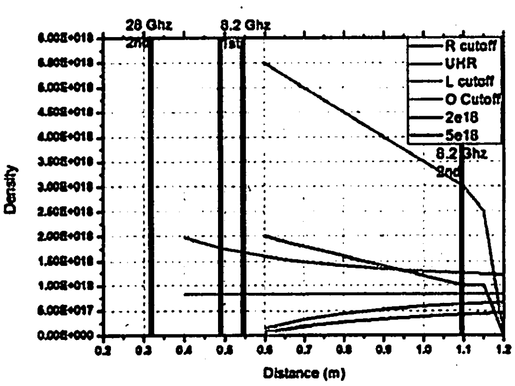

Fig. 1 Various profiles of QUEST

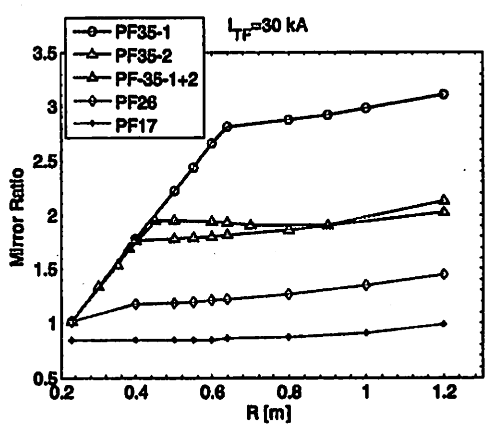

Fig. 2 Mirror ratios with different PF coil sets [4]

References

[1] S. H. Kim, H. Y. Lee, J. G. Jo and Y. S. Hwang, "One-dimensional full wave simulation on XB mode conversion in electron cyclotron heating", Phys. Plasmas 21, 062108 (2014)

[2] JongGab Jo, H. Y. Lee, S. C. Kim, S. H. Kim, Y. H. An, and Y. S. Hwang, "Efficient pre-ionization by direct X-B mode conversion in VEST", Phys. Plasmas 24, 012103 (2017)

[3] YoungHwa An, et. als, "Efficient ECH-assisted plasma start-up using trapped particle configuration in the versatile experiment spherical torus", Nucl. Fusion 57, 016001 (2017)

[4] Kishore Mishra, H. Zushi, H. Idei, M. Hasegawa, K. Hanada and QUEST team, "High βp plasma formation using off-axis ECCD in Ohmic heated plasma in the spherical tokamak QUEST", EPJ Web of Conferences 87, 02012(2015)

国際化推進共同研究概要

No.11

タ イ ト ル:　Model inter-comparison study of long-range chemical transport model to have a better understanding of PM2.5 issue over East Asia

研究代表者:　WANG, Zifa

所内世話人:　鵜野　伊津志

実 施 期 間:　2017 年 2 月 19 日　～2 月 23 日

研 究 概 要:　中国華北平原から北京にかけて観測される高濃度の PM2.5 汚染とその越境影響について、野外観測結果の解析と複数の化学輸送モデルを用いた相互比較実験の取りまとめを進めた。 今回は、特に、2015 年 3 月末から 4 月初めに北京市内で観測された黄砂と大気汚染粒子の混合状態の観測と、同期間を対象としたモデル解析結果の論文化を最優先に進めた。

No. 11

タイトル：　Model inter-comparison study of long-range chemical transport model to have a better understanding of PM2.5 issue over East Asia

研究代表者：　Zifa WANG　(中国科学院大気物理研究所)

共同研究の目的

本共同研究では、中国華北平原から北京にかけて観測される高濃度の PM2.5 汚染とその韓国・日本域への越境影響について、野外観測結果の解析と複数の化学輸送モデル(NAQPMS, CMAQ, GEOS-CHEM など)を用いた相互比較実験を進めている。

中国と福岡での最新のエアロゾルの観測装置、ライダーなどを駆使したデータの蓄積を独自に行い、同時に、中国・韓国・台湾・日本・アメリカ合衆国の研究者が進めているアジア域の化学輸送モデル相互比較実験(MICS-Asia)への参画を通じて、PM2.5 大気汚染のモデルの問題点とその改良を進め、化学輸送モデルの精緻化を目指す。

共同研究の成果

今年度の PM2.5 の観測は、中国大気物研究の 325mの観測鉄塔周辺と九州大学応用力学研究所で同一の観測測値（偏光式光学粒子計測器 POCP、化学成分連続自動分析装置ＡＣＳＡ）を進めた。この観測データは、中国大気物理研究所の化学輸送モデル NAQPMS と、九大で進めている CMAQ 及び GEOS CHEM 化学輸送モデルの相互比較実験に用いることと、それを用いた中国からの研究代表者の訪日時にモデル相互比較実験の仕様の詳細を議論した。

モデルの相互比較研究の対象として、これ以外に、（１）２０１４年５月末から６月初めに１週間の長期にわたり継続した高濃度黄砂と大気汚染現象、（２）２０１５年１月の高濃度越境輸送エピソード、の２つの事例解析も行っている。

（1）についての研究成果は研究論文「Modeling the Long-Range Transport of Particulate Matters During Winter in East Asia using NAQPMS and CMAQ」として学術雑誌に投稿中である。（2）の事例解析についても２編の学術論文としてとりまとめることを相互に同意した。

<u>セミナー講演</u>

2017年2月19日から2月23日に国際化推進共同研究経費でZifa Wang教授が来日した。これにあわせて、同時に、大気物研究所の研究費でLi Jie教授、Pan Xiaole准教授も来日した。これを機会に、以下の３件のセミナーを行い、相互の意見交換を行った。

2017年2月20日

Li Jie教授： An Integrated Study of AIR Pollution PROcesses in Beijing (AIRPRO) between China IAP and UK $PM_{2.5}$.

2017年2月22日

Zifa Wang教授：Towards the Next Generation Air Quality Forecast Model for the analysis and modeling of Chinese PM2.5 pollution

Pan Xiaole 准教授: Long-term analysis of aerosol compositions and polarization OPC observation in Beijing, China