Steady-State Operation Scenario and the First Experimental Result on QUEST

Kazuaki HANADA, Kohnosuke SATO, Hideki ZUSHI, Kazuo NAKAMURA, Mizuki SAKAMOTO, Hiroshi IDEI, Makoto HASEGAWA, Yuichi TAKASE[1)], Osamu MITARAI[2)], Takashi MAEKAWA[3)], Yasuaki KISHIMOTO[3)], Masaki ISHIGURO[4)], Tomokazu YOSHINAGA[5)], Hiroe IGAMI[5)], Nobuhiro NISHINO[6)], Hiroshi HONMA[4)], Shoji KAWASAKI, Hisatoshi NAKASHIMA, Aki HIGASHIJIMA, Yuta HIGASHIZONO, Akira ANDO[7)], Nobuyuki ASAKURA[8)], Akira EJIRI[1)], Yoshihiko HIROOKA[5)], Akio ISHIDA[9)], Akio KOMORI[5)], Makoto MATSUKAWA[8)], Osamu MOTOJIMA[5)], Yuichi OGAWA[1)], Noriyasu OHNO[10)], Yasushi ONO[1)], Martin PENG[11)], Shigeru SUDO[5)], Hiroshi YAMADA[5)], Naoaki YOSHIDA and Zensho YOSHIDA[1)]

Research Institute for Applied Mechanics, Kyushu University, Kasuga, Fukuoka 816-8580, Japan

[1)]*University of Tokyo, Kashiwa, Chiba 277-8561, Japan*

[2)]*Tokai University, Kumamoto 862-8652, Japan*

[3)]*Kyoto University, Kyoto 606-8502, Japan*

[4)]*IGSES, Kyushu University, Kasuga, Fukuoka 816-8580, Japan*

[5)]*National Institute for Fusion Science, Toki, Gifu 509-5292, Japan*

[6)]*Hiroshima University, Higashihiroshima, Hiroshima 739-8527, Japan*

[7)]*Tohoku University, Sendai, Miyagi 980-8579, Japan*

[8)]*Japan Atomic Energy Agency, Naka, Ibaragi 311-0193, Japan*

[9)]*Niigata University, Niigata 950-2181, Japan*

[10)]*Nagoya University, Nagoya, Aichi 464-8603, Japan*

[11)]*Oak Ridge National Laboratoty, Oak Ridge, TN 37831, USA*

(Received 9 January 2009 / Accepted 8 July 2009)

QUEST focuses on the steady state operation of the spherical tokamak by controlled PWI and electron Bernstein wave current drive. One of the main purposes of QUEST is an achievement of long duration discharge with MW-class injected power. As the result, QUEST should be operated in the challenging region on heat and particle handling. To do the particle handling, high temperature all metal wall up to 623 K and closed divertors are planned, which is to realize the steady-state operation under recycling ratio, $R = 1$. This is a dispensable check to DEMO, because wall pumping should be avoided as possible in the view of tritium retention. The QUEST project will be developed in increment step such as, I. low β steady state operation in limiter configuration, II. low β steady state operation in divertor configuration, III. relatively high β steady state operation in closed divertor configuration. Phase I in the project corresponds to these two years, and final goal of phase I is to make full current drive plasma up to 20 kA. Closed divertor will be designed and tested in the Phase II. QUEST is running from Oct., 2008 and the first results are introduced.

Keywords: steady state operation, spherical tokamak, electron Bernstein wave, plasma wall interaction

DOI: 10.1585/pfr.5.S1007

1. Introduction

It is important to obtain the academic basics to support high beta and steady state operation approaches. The QUEST (Q-shu University Experiment with Steady State Spherical Tokamak) project focuses on the steady state operation of the spherical tokamak (ST) which has the capability to attain high β rather than conventional tokamaks. A final target of the project is the steady state operation of ST with relatively high β under controlled plasma wall interaction (PWI).

The main difference between a pulsed operation and a steady one is the difficulties of handling of the heat and the particles loads. Although the transient huge heat load comes from plasma in the pulsed operation, the condition of the plasma facing components (PFCs) does not affect the performance of the plasma so much. While in the steady state operation, the erosion and the sputtering of the material make serious effect in the maintenance of the

author's e-mail: hanada@triam.kyushu-u.ac.jp

high performance plasmas via wall saturation and impurity accumulation [1, 2]. The continuous heat load makes large damages to the material of PFCs. The particle handling is more complicate in steady state operation. Because the wall pumping works well even on the divertor configuration, therefore the temperature and the number of absorbed particles of the PFCs should be controlled during discharges. Long duration discharges were sometimes terminated by the wall saturation phenomenon that the particles stored in the wall come back to the plasma abruptly. When the wall saturation phenomenon takes place, the influx of the particles increases and the particle handling could not work well [3]. As the behavior of Hydrogen from the wall significantly depends on the wall temperature [4], the control of the wall temperature and the number of the absorbed particles to the wall should be done.

The QUEST project will be developed in increment step such as, I. low β steady state operation in limiter configuration, II. low β steady state operation in divertor configuration, III. relatively high β steady state operation in closed divertor configuration, where β means the ratio of plasma pressure to magnetic pressure. The specific purpose in phase I is:

(1) To examine the steady state current drive and the generation of closed flux configuration by electron Bernstein wave (EBW) current drive (CD).

The purposes in Phase II are:

(1) To comprehensively establish recycling control based on control of wall temperature, and advanced wall control under high plasma performance.

(2) To improve diverter concepts and to establish the way of controlling particles and heat loads during long duration operation.

(3) To obtain relatively high β (10 %) under high elongated plasma shape and additional heating power in short pulse discharge up to 1 s.

In this paper, the physical design and the specification of QUEST are described in Section 2, and the first experimental results are shown in Section 3 and the summary is described in Section 4.

2. Physical Design and Specification of QUEST

2.1 Machine size decision

Machine size is the most important parameter of the project, because the appropriate size should be naturally derived from the mission of the project. The final goal of the project mission is to provide an academic basis for steady state high β operation at low aspect ratio as contributed to ST-based CTF, which is an important step to develop a fusion power plant. We should select the machine size to attain the final goal of the mission in the view of scientific consideration.

Before the investigation for the machine size, the limitation associated with the machine site should be shown.

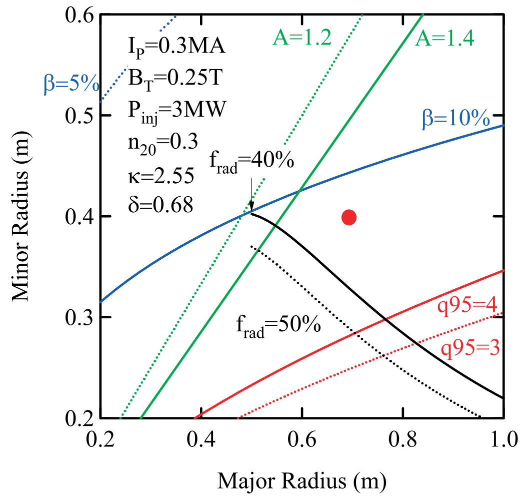

Fig. 1 Operation window in the case of L-mode confinement scaling provided from the data base for conventional tokamaks under the condition of plasma current, I_p = 0.3 MA, toroidal magnetic field, B_T = 0.25 T at the plasma center, injection power, P_{inj} = 3 MW, density normalized by 10^{20} m^{-3}, n_{20} = 0.3, elongation, κ = 2.55, and triagularity, δ = 0.68, where A and q_{95} show aspect ratio, and safety factor, respectively. The heat load is estimated from the width of heat flux on the midplane in low field side [6] and it is not considered about the expansion of magnetic flux surface on the divertor plate. A dot shows the selected parameter. f_{rad} shows the fraction of radiation loss to the input power.

The commercial power source at the site is limited up to 7 MVA and a MG of 125 MJ (effective power is 60 MJ within 1 sec) is available in the short pulse operations. When toroidal magnetic field of 0.25 T at 0.64 m would be applied to the machine, the electric power source of 4 MVA is required continuously and it will be supplied from commercial power source. Available heating power of 3-5 MW will be restricted by the capability of MG up to 1 sec. In particular, the maximum plasma current have the strong relation to the machine size and the capability of available power source and is compelled to be restricted up to 0.3 MA in the case of major radius of 0.64 m. When the larger machine size is selected, the maximum plasma current will be reduced to the lower level because of the increment of the electric power source required to operate poloidal field (PF) coils. When we would like to sustain the plasma in steady state, all of electric power should be supplied from commercial power source. Therefore the maximum plasma current is limited to 100 kA under the heating power is less than 1 MW. RF heating power of 8.2 GHz up to 400 kW, and 2.45 GHz up to 50 kW was prepared for plasma heating. The other power supply can be applied to a heating source up to 500 kW of the other RF heating systems. Finally, we can operate 1 MW power source for plasma heating in Phase II.

Figure 1 shows an operation window on minor and major radii considering under the above-described site limitation for heating power, plasma current and toroidal mag-

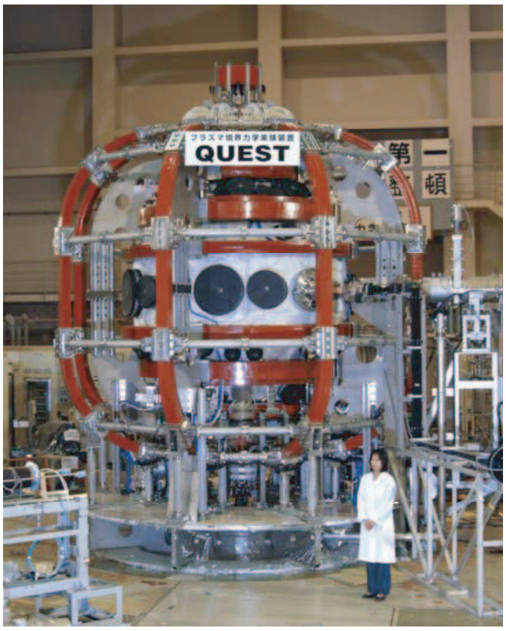

Fig. 2 The vacuum vessel and TF, PF coils of QUEST

netic field. The solid and dotted red lines show the line for $q_{95} = 4$ and 3, respectively. Generally speaking, it is difficult to operate on $q_{95} < 3$ in tokamaks and the solid red line provide a low q limit. The solid black line shows the heat load limitation to a divertor plate under the condition of the fraction of radiation, $f_{\rm rad} = 40\,\%$ (60 % of the injected power comes to the divertor plates as the heat flux) in double null configuration. In steady state, the heat load to PFCs should be kept down on less than 10 MW/m^2. This is also the requirement for ITER [5]. When the major radius is less than the black line, the heat load to divertor plate goes beyond the boundary of the ITER requirement. The solid and dotted green lines show the aspect ratio, $A = 1.4$ and 1.2, respectively. The reduction of aspect ratio leads to the difficulty of the construction of the machine because of the technical limit to construct the machine. The solid and dotted blue lines show the constant β of 10 % and 5 % based on the ITER 89P L-mode scaling [5] under the condition of plasma current, $I_{\rm p} = 0.3$ MA, toroidal magnetic field, $B_{\rm T} = 0.25$ T at the plasma center, injection power, $P_{\rm inj} = 3$ MW, density normalized by 10^{20} m^{-3}, $n_{20} = 0.3$, elongation, $\kappa = 2.55$, and triagurality, $\delta = 0.68$. This indicates the possibility to complete the mission of the project. The machine size window to attain $\beta = 10\,\%$ is not so large and the selected machine size (red point on the figure) has the candidate to attain the mission. The vacuum vessel and TF, PF coils of QUEST are shown in Fig. 2 and the selected parameters are listed in Table 1.

2.2 Non-inductive current drive in phase I and II

In Phase I, the most important issue of the QUEST project is to achieve steady state operation. Non-inductive current drive is indispensable for steady state operation.

Table 1 Specifications and major parameters of QUEST. $P_{\rm RF}$ and $P_{\rm NBI}$ mean the expected injection power of RF and NBI, respectively. *means under consideration.

	Phase I	Phase II		Phase III
	Steady	Pulse	Steady	Steady
R (m)	0.68			
a (m)	0.4			
$B_{\rm T}$ (T)	0.25	0.5	0.25	0.25
$I_{\rm P}$ (kA)	20	300	100	300
$P_{\rm RF}$ (MW)	0.45	1	1	1
$P_{\rm NBI}$ (MW)	—	2	—	4*

The different way to drive the plasma current from that in conventional tokamaks should be developed. Lower hybrid wave (LHW) is one of the promising methods to make current and the long duration discharge more than 5 h was achieved on TRIAM-1 M [7, 8]. However LHW cannot penetrate to the center part of the high density plasmas under the low magnetic field condition in the view of the accessibility of the wave. As for electron cyclotron current drive (ECCD), it is difficult to avoid the cut-off of electron cyclotron wave (ECW) under the low field condition. Instead of LHW current drive (LHCD) and ECCD, electron Bernstein wave current drive (EBWCD) and higher harmonics fast wave current drive (HHFWCD) are candidates to establish the maintenance of plasma current in steady state on STs by use of RF. As we do not have any heating source for HHFWCD in the view of frequency at present, EBWCD is suitable way to apply to QUEST in Phase I.

In Phase II, upgraded heating source such as NBI up to 2 MW will be prepared. At that time, NBCD and bootstrap current are additional candidates to provide sufficient plasma current. The current drive efficiency of NBCD is mainly decided by the fast ion thermal process that is classical slowing down process. In ST, the confinement of fast ion is crucial issue for NBCD because of low toroidal magnetic field. A present fast ion orbit calculation predicted better fast ion confinement than conventional tokamaks. In this paper, it is difficult to discuss about the current drive efficiency including fast ion confinement. We assume that the fast ion confinement in ST is the similar to the conventional tokamaks. Before discuss about the current drive efficiency of NBCD, the beam energy to inject plasma should be decided, because large amount of through power is dangerous as well as inefficient. The absorption of the beam is decided by the cross-section for the processes, which are charge exchange, ionization by ion, and ionization by electron. In QUEST, the machine size is not so large and plasma density is in medium range. Therefore the suitable beam energy may be around 20-30 keV, where the charge exchange process will be dominant. When electron density, n, is 5×10^{19} m^{-3}, cross section multiplied by density, $n\sigma = 5$ m^{-1} at 10 keV of hydrogen beam, $n\sigma = 3$ m^{-1} at 20 keV and $n\sigma = 2$ m^{-1} at 30 keV. As tangent length of QUEST is 1.7 m, 96 % absorption can be expected at 30 keV beam. We assume that 30 keV hydrogen beam will be injected to tangential direction for the estimation of efficiency of NBCD. In this case, the driven current of several 100 kA can be expected.

Investigation of estimation of current drive efficiency of EBWCD is done under the given machine size and heating source in Phase I and II. At first, two excitation scenario of EBW (O-X-B and X-B) should be discussed, because injection way of RF depends on the scenario of wave excitation. The O-X-B scenario is utilized a mode conversion from O-mode to X-mode at the cut-off region for O-mode. The conversion rate from O-mode to X-mode is well-understood and depends on the injection angle to the cut-off layer. When we are willing to adopt this scenario, the adjustment of the injection angle is crucial. The converted X-mode will transfer to the upper hybrid resonance (UHR) from high field side and it will convert to EBW. EBW propagates inwards again and is absorbed bulk electrons effectively. By the effect of the magnetic shear, EBW has the single directed momentum in specified situation and it delivered to plasma. Accordingly plasma is provided single directed momentum from EBW and plasma current can be driven. The X-B scenario is the same in delivery and receipt of the momentum, however the mode conversion process is different. The RF will injected perpendicular to the magnetic field and 3 wave coupling play an essential role in mode conversion process and the mode conversion efficiency significantly depends on the scale length of electron density at UHR. To convert to EBW, we should control the density gradient at UHR.

Experimental observations of EBWCD are obtained in various devices. Driven current of 100 kA at 60 GHz 600 kW was achieved in COMPASS-D on the O-X-B scenario [9]. Full non-inductive plasma current up to 20 kA in a sequence of plasma start-up on center solenoid less configuration was achieved on LATE [10]. The X-B scenario was executed on TST-2 by appropriating RF power source of 8.2 GHz, 200 kW. Plasma heating could be observed in the case of application to ohmic heated plasma. Full non-inductive plasma current of 4 kA for 0.3 sec could be sustained by only-RF on TST-2 [11]. First observation of plasma heating via X-B scenario was also reported from TST-2 experiments [12].

In order to study the wave propagation and absorption of EBW, the wave trajectory has been calculated with some ray tracing codes [13]. The wave trajectory of the incident wave was calculated using the TASK/WR ray-tracing code [14] as shown in Fig. 3. The local wave electric fields were evaluated in the ray trace calculation, and used for the Fokker-Planck (TASK/FP) [14] analysis. The driven-current profile was estimated in the FP analysis. The geometrical coordinates were taken as a simple tokamak configuration with circular poloidal cross-sections. The major and minor radii were $R_0 = 0.64$ m and $a = 0.36$ m in this calculation. The profiles of electron density and temperature, and plasma current were assumed to be parabolic. The central electron density and temperature were $n_{e0} = 0.2 \times 10^{19}$ m^{-3} and $T_{e0} = 100$ eV, respectively.

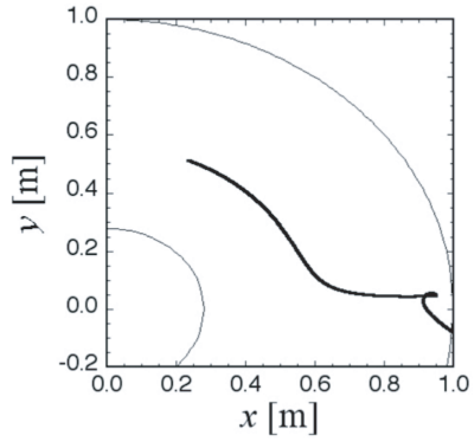

Fig. 3 An example of wave trajectory at the toroidal cross section in the O-X-B conversion scenario at the QUEST [16]. Incident O-wave from low filed side reflects at the plasma cut-off and converts to X-mode. X-wave meets UHR and converts to B-wave.

The total plasma current was 20 kA. The *O*-cutoff was at the plasma radius $r = 0.2$ m. The local wave electric fields were calculated in the ray trace calculation, and used for the Fokker-Planck (TASK/FP) analysis. In the TASK/FP code, a quasi-linear RF operator in the velocity space was evaluated from the wave field, and the bounce averaged velocity distribution function was calculated to evaluate the plasma current driven by the EBWCD. The nonlinear collision operator was used with the trapped electron effect. As the results $I_P/P_{RF} = 0.11$ A/W can be obtained [15, 16].

Bootstrap current will play a significant role in driving non-inductive current at low collisional region. We try to calculate the bootstrap current at some given plasma parameters. Expression provided by S.P. Hirshman [17] is utilized in the calculation. Total value of bootstrap current is 36 kA even in high β (~20 %) plasma. To confirm the result, modification of density and temperature profile were executed, however total bootstrap current did not change so much. This indicates that the bootstrap current may not play an essential role in non-inductive current drive in QUEST.

2.3 Control of plasma wall interaction

Control of plasma wall interaction (PWI) is the key issue to maintain plasmas in steady state. In TRIAM-1 M, strong wall pumping played an essential role in the particle balance in ultra long duration discharge [18]. The estimated value of wall pumping rate depended on plasma parameters such as density, therefore it was difficult to control wall pumping rate during plasma discharge [18]. Moreover the property of the wall pumping was alternatively turned on and off [18] and the wall sometimes modified to source of particles. When the wall works as particle source, the control of plasma density becomes to be difficult and finally plasma termination may take place. To avoid this difficulty of particle handling in long duration discharge, high temperature wall (HTW) is effective, because HTW can hold recycling rate of particles to unity. This property of the wall had been made sure in laboratory experiments [19]. We would like to install HTW in the vacuum vessel on QUEST as shown in Fig. 4. The temperature of HTW will be decided by the reemission property of Hydrogen from the wall. The temperature of HTW may be kept at 623 K at least [19]. The surface of HTW is covered with Tungsten, although the details of the design of HTW are under consideration. At the first year in Phase II, HTW will be installed on the vacuum vessel.

To handle heat and particle load, a divertor is the key component. The plan of the construction of the divertor structure is also shown in Fig. 4. In Phase I, limiter configuration is mainly used. The center stack is surrounded by limiters made of SUS306 coated by Tungsten. The flat divertor plates are installed on the top and bottom side of the vacuum vessel as shown in left half of Fig. 4 and its surface will be also coated by Tungsten. In the first step, diagnostics to monitor heat and particle flux during discharges will be installed and the estimated heat and particle flux will be used for the design of closed divertor structure.

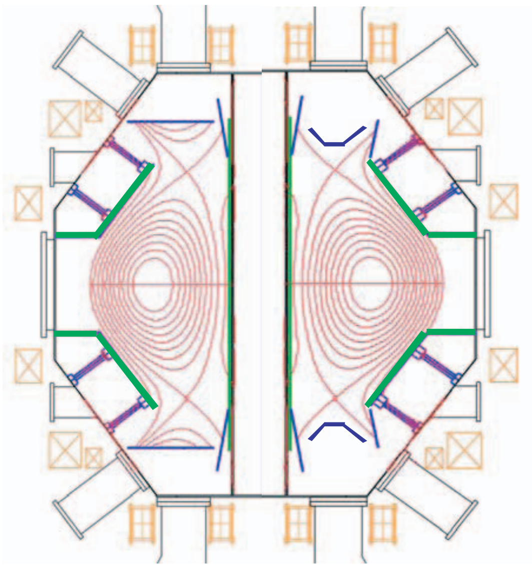

Fig. 4 Schematic view of the plan of the installation of the first wall on QUEST is shown. Left half of the figure illustrates the situation of the first step (the flat divertor plates and HTW). Right half shows the second step (the closed divertor). HTW is shown by green lines. The wall of the center part (high field side) does not heat actively, but passively due to the radiation from the wall.

3. Results of the First Experiments

The first experiments on QUEST started on Oct., 2008. The purpose of this experimental campaign is to make sure the soundness of the machine and to obtain tokamak plasma. In this experimental campaign, it is impossible to operate the machine in steady state and the pulse duration is limited up to a few seconds because of the limitation of power supply for TF coil. The power supply will be improved in next year and steady state operation will be done.

Figure 5 shows the time trace of 2D image of visible light in first formation of tokamak configuration measured with high speed camera. The center solenoid with the cancel coil and a pair of PF coil were used to achieve this discharge. Peak of plasma current is about 10 kA and plasma shifts outwards at 0.50595 s because of poor equilibrium due to weak vertical field. At first plasma was produced by power of RF and a cylindrical plasma around electron cyclotron resonance (ECR) layer can be observed at 0.4545 s. And then the current of center solenoid increases gradually and plasma deformation appears due to the upwelling of return magnetic field from the seam of coils of the center solenoid, which is composed of three independent coils. At 0.486675 s, the plasma was divided into three parts. The center part of plasma forms tokamak configuration afterwards. At 0.4923 s, a bright point appears on the surface of the inner limiter and this bright point expanded as shown the picture at 0.49365 s. Just before the appearance of the bright point, glimmers at the top side of the inner limiter can be observed. At 0.49875 s, tokamak configuration was formed tentatively and the plasma shifts outwards because of weak vertical field. These pictures will be useful to understand the formation of closed flux surface.

After this experiment, we try to obtain the formation of closed flux surface without the assistance of magnetic flux from the center solenoid. This is the crucial point to achieve the steady state operation of ST.

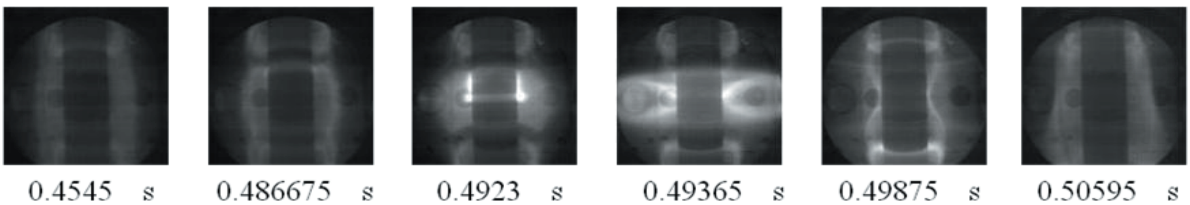

Fig. 5 Time traces of 2D image of visible light on the first formation of tokamak plasma measured with high speed camera. The center stack can be seen at the center part of each picture.

4. Summary

Physical design of the machine size of QUEST is shown. Conventional tokamak scaling indicated the expected plasma parameters, and ray trace and Fokker-Plank calculation for EBWCD were executed. As the result, the specification of QUEST is consistent with the mission. Heat and particle handing is the crucial issue to obtain steady state operation, and it is shown that a combination of closed divertor and high temperature wall is a candidate to resolve the difficulty. In QUEST, the combination will be adopted and the preparation will proceed.

Acknowledgements

This work is performed with the support and under the auspices of the NIFS Collaboration Research Program (NIFS05KUTR014, NIFS08KUTR022). This work is partially supported by a Grant-in-aid for Scientific Research from Ministry of Education, Science and Culture of Japan.

[1] M. Chatelier *et al.*, Nucl. Fusion **47**, S579 (2007).
[2] R. Bhattacharyay *et al.*, J. Nucl. Mater. **363-365**, 938 (2007).
[3] K. Hanada *et al.*, Nucl. Fusion **41**, 1539 (2001).
[4] R. Bhattacharyay *et al.*, Nucl. Fusion **47**, 864 (2007).
[5] ITER Physics Basis, Nucl. Fusion **39**, No12 (1999).
[6] A. Loarte *et al.*, J. Nucl. Mater. **266-269**, 587 (1999).
[7] H. Zushi, *et al.*, Nucl. Fusion **45**, S142 (2005).
[8] K. Hanada *et al.*, Fusion Eng. Des. **81**, 2257 (2006).
[9] V. Shevchenko *et al.*, Phys. Rev. Lett. **89**, 265005 (2002).
[10] T. Yoshinaga *et al.*, Phys. Rev. Lett. **96**, 125005 (2006).
[11] A. Ejiri *et al.*, Nucl. Fusion **46**, 709 (2006).
[12] S. Shiraiwa *et al.*, Phys. Rev. Lett. **92**, 035001 (2004).
[13] T. Maekawa *et al.*, J. Phys. Soc. Jpn. **48**, 247 (1980).
[14] A. Fukuyama, Fusion Eng. Des. **53**, 71 (2001).
[15] H. Idei *et al.*, Proc. 32nd International Conf. on Infrared and Millimetre Waves, Cardiff, UK, 789-790 (2007).
[16] H. Idei *et al.*, J. Plasma Fusion Res. Series **8**, 1104 (2009)
[17] S.P. Hirshman, Phys. Fluids **31**, 3150 (1988).
[18] M.Sakamoto *et al.*, Nucl. Fusion **42**, 165 (2002).
[19] M.Miyamoto *et al.*, J. Nucl. Mater. **337-339**, 436 (2005).