ISSN 0915-633X

NATIONAL INSTITUTE FOR FUSION SCIENCE

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(Received – Sep. 21, 1990)

NIFS-58 Oct. 1990

RESEARCH REPORT
NIFS Series

NAGOYA, JAPAN

Abstract:

The significance of medium- and small-size devices is reviewed and discussed taking the case of ITER Physics R&D. It is pointed out that the heat flux density expected in a fusioning plasma will become as large as an unexperienced high level and it is not self-evident that one could extend the present energy confinement time scaling to predict ITER performances. The confinement of fusion plasma is considered to be related to heat, momentum and mass transport problems in a nonlinear non-equilibrium system with a finite source and dissipation. Therefore, a recent progress of basic experiments and theories in fluid dynamics may greatly be referred, where series of basic experiments are unveiling various modes of heat and mass transport in the nonlinear non-equilibrium system including a scenario and conditions of transition to turbulence. In the field of plasma confinement, the more complex scenario would be expected. Therefore, a systematic and basic research along the same line is highly desired on medium- and small-size devices in order to get insight into the physics of plasma confinement which will provide us the methodology for confinement improvement.

The Significance of Medium- or Small-size Devices in Fusion Research

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Key words:

Medium- and small-size devices, ITER Physics R&D, ITER design assumptions, The possible research on medium- and small-size devices, general principles, Heat flux density, energy confinement time scaling, Transport

1. Introduction

Almost thirty nine years have past since fusion research was unveiled by President Peron of Argentina and thirty five years have already past since Dr. H.J. Bhabha, Chairman of the First United Nation International Conference on the Peaceful Uses of Atomic Energy in Geneva, announced boldly that the method of controlling and utilizing fusion energy would be established within 20 years. During this period various concepts and configurations for confinement and heating , such as pinch, mirror, inertial confinement, stellarator and tokamak, have been investigated, and medium- and small-size devices have been constructed in many of the countries in order to establish database for producing and controlling high temperature plasmas and energetic researches on high temperature plasma physics, especially the elucidation and improvement of confinement properties for fusion, have been performed. Among all those researches confinement experiments on tokamak and stellarator devices have shown remarkable progress worldwide and the frontier of fusion research is now considered to be approaching the first historical moment of achieving a critical state in D-T fusioning plasmas.

The main fusion devices extensively investigated in magnetic confinement research are large tokamaks such as TFTR(USA), JET(EC) and JT-60(Japan). These devices have been put into operation and started experiments since 1982, 1983 and 1985, respectively. Particularly in JET scientists have been invited from ASDEX to realize H-mode in JET, which was found in a medium-size tokamak ASDEX in the Federal Republic of Germany in 1982, in order to overcome confinement degradation with additional heating, which JET was facing at that time. Necessary remodelling for divertor operation to achieve H-mode discharges was done and a great effort has been made to improve plasma parameters in JET. After changing a part of first wall materials with Beryllium in order to prevent dilution of the plasma due to mixing of impurities from the first wall, DT equivalent fusion relevant conditions($Q \leqq 0.7$) have been achieved in DD plasmas in JET and the achievement of a critical state($Q=1$) is now close at hand. The duration of the fusion relevant conditions are now limited to $\leqq$ 1s[1] , however a great effort of extending the duration is being performed with taking measures to deal with impurity problems.

On the otherhand ITER conceptual design work ,initiated in April 1988 under IAEA, is now at the final stage and a complete ITER concept definition has been made public with physics guidelines. Furthermore in order to justify the concept a database for ITER is being constructed with collecting data from world-wide medium- and small-size devices. As a result they foresee the possibility of achieving self-ignition conditions in a device with a major radius of 6 m, minor radius of 2.2 m, an ellipticity of 2.2 and a plasma current of 22 MA, if they could improve a numerical coefficient of the empirical L-mode scaling laws by a factor of 1.6 with some means. The reasons why the H-mode was not chosen as a guideline for a core plasma state in ITER are that a database for the H-mode is not sufficient to derive empirical scaling laws for energy confinement times and that the H-mode soon turns to be the L-mode and seems to be difficult to maintain due to the accumulation of impurities and the unavoidable increase of plasma densities comparing with the L-mode. The mechanism and software to make H-mode stationary (i.e. an operational scenario to establish a stationary H-mode with utilizing instabilities at plasma boundaries,so-called ELM) have been extensively investigated and developed on medium- and small-size devices. Many improved confinement modes other than the H-mode have been found in those devices, however little is understood about confinement properties. Moreover it is observed that the accumulation of impurities is more enhanced and results in dilution of the fuel in those modes. Therefore it is necessary to do research for evaluating those modes as a core plasma state. Particularly the development of a divertor mechanism, software or an entirely new mechanism against impurity accumulation in those modes is desirable. Consequently ITER is designed to have some possibilities to upgrade its plasma current up to 28 MA in order to aim at self-ignition at least under low level of He accumulation even if the improvement of a numerical coefficient of the L-mode scaling laws is unsuccessful. Many reports have been published under ITER project and a knowledge-base and database arranged therein is nothing but a compilation of the fruits from experimental efforts of medium- and small-size devices.

Taking a close look at fusion history and taking the role into consideration, which medium- and small-size devices have actually played in developing the methods for confinement improvement and contributing to a persistent effort in achieving a critical condition in DT burning plasmas, the significance of medium- and small-size devices is outstanding. For example the program for achieving a critical condition would remain impracticable if H-mode were not found in ASDEX and introduced in JET.

Here we summarize briefly the significance of medium- and small-size devices and describe the roles of those devices to solve problems clarified at issue under ITER conceptual design work.

2. The significance of medium- and small-size devices

The history of fusion research consists of invention and utilization of small devices. It is the history of studying plasma properties and producing higher temperature plasmas than in the past with developing plasma control, heating, fuelling , and exhausting methods or developing first wall materials and cleaning methods of the first wall surface (discharge cleaning). In other words the history of fusion research can be said to be the history of research on medium- and small-size devices. It is natural that ,in the case of initiating a new grade research of plasma confinement, a knowledge from those devices always provided a basis for judging:

- whether the relevant step is appropriate or not from a viewpoint of research development,
- whether a knowledge and working hypothesis acquired in the past can be used on the next step by the extrapolation or not,i.e., whether they could assure us of perfoming research development on the relevant step.

Here we reconsider the significance of medium- and small-size devices.

Recently, fusion research has reached the stage at which one can conceive a reactor core plasma in tokamaks, however, it still remains at the stage of the fusion equipment science and the mechanical structure science. In other words there are a lot of unpredictable properties without producing the plasma of the relevant size and shape. The results can not necessarily be characterized by non-dimensional parameters and are reflected from properties of the device itself. Aiming at producing and controlling plasmas with sufficient confidence, a basic research of high temperature plasmas has been performed on medium- and small-size devices , elucidating physical mechanisms, inventing devices of higher capabilities and discovering operational scenarios. Fusion research is advancing on such a basis. This type of research is indispensable in the development of fusion reactors even if one can conceive a reactor core plasma. From this point of view, the significance of medium- and small-size devices is summarized and exists at following points:

① the invention and testing of the <u>devices</u> based on original ideas and the propagation and improvement of them,

② the invention and testing of the <u>structures</u> based on original ideas and the propagation and improvement of them,

③ the basic understanding and systematization of the relevant phenomena and the improvement based on the first principles.

The items ① and ② should be born in mind even if a some type of fusion reactor has been completed. Referring to the item ② particularly, mediumand small-size devices won honor exceedingly. The most typical structure is a divertor which was invented and introduced into stellarators. As for tokamaks the divertor has been introduced into JFT-2a, DITE, ASDEX, JFT-2M, DIII-D, JET and JT-60 successively and it is now considered to be a standard equipment for the design of the next generation devices. Referring to the item ③, concrete points at which the significance of the medium- and small-size devices exists are as follows;

- to make knowledge obtained in each device common and establish confinement physics of high temperature plasmas,
- to actually predict plasma confinement in large devices,
- to propose the measures and the control scenarios for improving confinement
- to establish methodologies to simulate some phenomena which will be expected to occur in self-ignited fusion devices by utilizing medium- and small-size devices.
- to improve performances and capabilities of self-ignited fusion devices by these methodologies.

These aspects have been realized for a long time in the area of aeronautical engineering and shipping engineering in which the development of devices is based on fluid dynamics. Simulation is actually made use of particularly in developing a new airplane or ship, solving problems of the existing airplanes and ships and making higher performance devices. It is no exaggeration to say that medium- and small-size devices are standing at the turning point beyond which they will be able to keep their great significance in fusion research or not, depending on whether the above-mentioned methodologies can be established or not.

In chapter 3, the concrete role which medium- and small-size devices should play will be described, in order to contribute to the solution of the problems, for example, clarified at issue under ITER relevant physics R & D.

Here TFTR, JET and JT-60 can be categorized into medium-size devices.

3.The Role of Medium- and Small-size Devices in ITER Physics R&D

3.1 Tokamak Parameters

The parameters of the medium- and small-size devices are shown in Table 1, which have been nominated for the physics R & D in parallel with ITER EDA.

3.2 The role of Medium- and Small-size Devices

The ITER physics group started to work on the development of the long-term physics R&D and extracted ITER-related physics R&D items. The overall objectives of the ITER-related Physics R&D programme are to provide support to the design optimization and to complete the database necessary for taking the decision to start construction. The main specific areas to be covered:

① enhanced confinement,

② long-pulse operation and of discharge startup and shutdown (including non-inductive current drive),

③ power and particle exhaust physics (i.e., the combined fields of the plasma edge and plasma wall interaction as well as impurity control),

④ disruption control and operational limits,

⑤ physics of burning plasma,

⑥ heating and fuelling physics.

Here we discuss several issues related to those areas.

(1) Maintenance of Good Confinement

ITER design assumptions

- to assume enhancement of a numerical coefficient of L-mode scaling laws for energy confinement time by factors of 1.5-2,

- to assume preventive measures against impurity accumulations innevitably accompanied with enhanced confinement, in order to allow a long-pulse operation,
- to assume preventive measures to keep impurity contamination below the level of Zeff=2.0, in order to avoid dilution of core plasmas.

The possible research on medium- and small-size devices in this area

- to show the methodology of evaluating confinement characteristics totally, including the methods to evaluate robustness of the relevant core plasma state and impurity relevant characteristics.
- Plasmas have been revealed to respond in a variety of modes to external sources and boundary conditions. These modes include H-mode, L-mode, Z-mode and Pellet-mode, etc. Therefore the most urgent problem is to decide the type of mode to be utilized as a core plasma state in a self-ignited fusion device like ITER and provide the methodology to realize. In order to meet with this situation, an extensive research is desired, to provides an appropriate guideline for a core plasma confinement based on the understanding by the physics principles. With respect to this aspect we shall give more detailed discussions in chapter 4.

(2) Durability of Long Pulse Operation

ITER design assumptions

- to assume 100 sec for Q>20 and 1000 sec for Q $\sim$5,
- to assume non-inductive current drive.

The possible research on medium- and small-size devices in this area

- to discover the method to maintain good confinement(see (1)) and seek for a guideline for durable operation.

- There is no medium- or small-size device in which durability of long pulse operation can be demonstrated, with plasmas appropriate for a reactor core from the viewpoints of plasma densities, temperatures and impurity levels. Only the small tokamak(TRIAM-1M) has just shown 1 hour discharge with fairly low densities all over the world. The data base in this area is very poor, and should be reinforced. A new device to demonstrate is desirable especially to assure durability of long pulse operation. In those long pulse discharges we should clarify various time constants at issue which characterize interactions between plasmas and the first walls, and obtain a guideline for exhaustion of particles and heat, depending on input, to keep a core plasma at a desirable state.
- Existing current drive methods are providing the lowest efficiency necessary for ITER, including many technical problems to solve. There will be no gurantee for solving these problems only with technical development. The development of new appropriate current drive methods based on excellent ideas should be encouraged and many Proof of Principles experiments are necessary on medium- and small-size devices.

Long pulse operation and maintenance of good confinement should be achieved at the same time. Moreover long time durability should be related to other items from ③ to ⑥ and achieved with these items simultaneously.

(3) Design of Heat Removal and Diversion Mechanism

ITER design assumptions

- to assume 1 MW/ m^2 on the first walls near the main plasma
- to assume 10MW/ m^2 on the surface of divertor plates

The possible research on medium- and small-size devices in this area

- to establish the control methods:
  1. to produce low temperature and high density plasmas(T_e<10eV) just in front of the divertor plates,i.e., dense and cold divertor, in order to reduce impurity release from the divertor plates due to sputtering,
  2. to realize remote radiative cooling in divertor area in order to reduce heat flux onto divertor plates.
- to clarify transport phenomena through scrape-off layer plasmas including thermal particles, energetic electrons and ions under intensive current drive and(α - particle) heating and establish the method to deal with heat flux associated with these energetic particles.

(4) Disruption Control

ITER design assumptions

- to assume a fairly low frequency of disruptions because divertor plates and first walls can resist only 10 to 100 disruptions. If disruptions are too frequent, it becomes too difficult to maintain ITER because of frequent exchanges of first walls.
- to assume no group of energetic electrons (a few hundreds MeV and a few hundreds MJ) is produced at disruptions. These energetic electrons could cause fatal damages on divertor plates and cooling pipes. No data base exists about production of energetic electrons at disruptions.

The possible research of medium- and small-size devices in this area

- to construct the data base for disruptions including frequency and characterization. The frequency expected from the present data base amounts to be 5% of the total discharges when operated near the operation limits as required in Physics Phase and 2-4% in Technology Phase. If we could reduce this frequency, maintenance scenario would be greatly improved.

- to construct the data base for disruptions in the case of long pulse operation. The device capable of long pulse operation is needed to provide frequency data and characterization data and to provide methodology for avoiding or reducing disruptions.
- to characterize disruptions and clarify the mechanism.

(5) Physics of Burning Plasma

ITER design assumptions

- to assume achieving Q $\sim$5 for 800 sec to 1000 sec and providing research on physics of burning plasma.
- to assume DT plasma.

The possible research on medium- and small-size devices in this area

- Research on actually burning plasma is impossible except in JET and TFTR, however, a data base and knowledge at least necessary for preparation of studying physics of burning plasma should be established before construction of ITER. Particularly the constitution of plasma with respect to deuterium and tritium, which determine the burning rate, is very important. There is no gurantee for keeping plasma composition constant even if deuterium and tritium are fed equally. Among ion species impurity behavior is also important on a viewpoint of the dilution of a core plasma. All these behaviors are categorized in ion dynamics in a multi-ion component plasma and there have been no firm data base for that. The construction of this data base is one of the great subjects for the medium-size or larger devices, in which the effects of penetrating neutrals can be separated in studying ion dynamics.
- to establish the method of controlling ion temperatures to determine the

burning rate as well as the composition of the plasma.

- A burning plasma is heated up by 3.5 MeV α particles , which will give a great distortion to the distribution functions of the background plasma. At present a plasma under intense neutral beam heating is in a similar situation, however, the energy of the particles used for heating is 100 KeV at most and the distortion is small. The distortion of the distribution functions will affect greatly the confinement properties of the plasma and the accumulation of data to at least assure burning experiments in ITER is required. Systematic heating experiments with MeV Helium beam injection may be one of the most promising method in order to make this kind of database in the medium-size devices without actual burning plasmas.

4.The necessity for understanding plasma confinement from the general principles

The ITER scaling law for the L-mode energy confinement time as shown in the form of (1) is formulated, in a completely empirical manner using the data from medium- and small-size devices all over the world.

$$\tau_E^{IP89}=0.048M^{0.5}Ip^{0.85}R^{1.2}a^{0.3}k^{0.5}n^{0.1}B^{0.2}P^{-0.5} \qquad (1)$$

The empirical formula like (1) was firstly proposed and became to be known as Goldstone L-mode scaling law and has been revised many times to be formed into the ITER L-mode scaling law (1). This type of scaling law plays an important role of predicting the performance of ITER or next step devices. Connor[2] et al. have proposed the invariance principle which places constraints on the form of the energy confinement time scaling like (1). These constraints, deduced from the invariant scale transformation of a set of equations describing local plasma behavior, reduce the number of parameters which have to be investigated empirically and can serve to indicate which plasma models and equtions could describe the observed losses. However, to our regret, this has not been the final key to the solution of the problem. If we look at plasmas carefully, we are led to the possible conclusion that more than one and/or a variety of phenomena at different radial positions might be involved and effective alternately in the observed losses. Therefore, it may be possible that only a single set of equatons cannot describe the observed losses for a wide range of plasma parameters.[3)]

In dealing with complex phenomena, there are two approaches, microscopic theory and phenomenological theory. Ohkawa[4)] has proposed phenomenological models in plasma and discussed the plasma transport problems relating the energy confinement time scaling with the level of plasma turbulence. To consider tokamak transport, he made an analogy between the Reynolds' number in fluid dynamics and the anomaly factor in plasma transport theory. Based on an analogy with fluid dynamics, he catalogued various plasma transport models with the combinations of the type of the available free energy and the mixing length and gave the phenomenological method of estimating transport coefficients. The types of the free energy considered are the thermal energy of plasma, the magnetic potential of the trapped particles, the energy associated with the distortion of the distribution function due to heating and the free energy associated with the entropy of the plasma. If we could establish penomenological theory in plasma transport identifying the free energy source and its dissipation mechanism, we could confidently give the guidelines of improving plasma confinement based on the first principles, for example, by reducing the entropy term with controlling the profiles of fuelling and heating and by reducing the magnetic potential energy term of trapped particles with optimizing plasma shapes and operating in a second stability regime. A few small-size devices[5)] are designed and proposed to study a systematic change of plasma confinement according with the strength of the magnetic potential energy of the trapped particles. The free energy associated with the distortion of the distribution function due to heating could be smaller if the strength of the heating is smaller and the plasma density is higher. Therefore the heating method should be as gentle as possible and high plasma densities should be also preferable for better confinement.

In order to clarify phenomenological points at issue, let's consider the confinement problems from a little bit different points of view. In a fusioning plasma, a considerable amount of heat is supposed to be produced inside and transported outwards resulting in a large heat flux which may form a dissipative structure sometime in the process. We may find a similar situation in an externally heated plasmas and signs of a similar scenario from laminar flow to turbulent flow in the transport process with increasing input power. Table 2 shows plasma surface areas vs experimental input powers together with heat flux density(MW/m^2) on typical medium- and small-size devices. The heat flux density expected to flow out from a core plasma in ITER is also shown in the table. As shown in this table, the more heat should be carried out from the core plasma through the higher temperature plasmas than in the existing devices. Therefore, such a larger heat transport will require a very large temperature gradient if the transport coefficient be neoclassical. The largest temperature gradient observed so far is 32 kev/0.8m in a super-shot and in this case the energy input (heating) and particle supply (fuelling) at plasma center are performed at the same time by neutral beam injection. In ITER reference design, the temperature gradient assumed is about 20kev/2m and smaller than those in JET and DIII-D. On the otherhand, the heat flux density which should be carried out in ITER amounts to be $385KW/m^2$ and by far exceeds the values of $75KW/m^2$ and 207 KW/m^2, in JET and DIII-D, respectively, which means that the anomaly factor, defined by the ratio of a transport coefficient to a neoclassical one, should be larger if we simply extrapolate from the present data base. In other words, the transport expected in ITER must be naturally anomalous and it may be predictable that the turbulence be developed more strongly than observed in the present experiments.

Above arguements correspond to the facts that the temperature gradients seem to be unable to assume a large value for a given heat flux necessary to be carried out from a core plasma. The method to control density gradients is apparently important in order to keep steep temperature gradients. In Ohkawa's discussion, the ratio of the average transport coefficient to the maximum turbulent transport coefficient is assumed to be a function of the anomaly factor (the maximum turbulent transport coefficient/the neoclassical transport coefficient). If this be true, it will provide us with the phenomenological method to evaluate transport coefficients, though the hypothesis should be checked and confirmed. Judging from the present situations, an establishment of a phenomenological theory which can predict the confinement degradation with increasing heat flux density and an establishment of firm principles and bases which can provide us with the method to prevent degradation by reducing the the free energy source responsible for the turbulent transport, are urgently required. Only they could provide us with the guidelines how to choose a set of the most appropriate plasma paramaters. The more detailed and systematic research on medium- and small-size devices is desirable.

5. Concluding remarks

This report has been prepared to the Panel Activities for the Japan Atomic Energy Commission in 1989-1990. The significance of the medium- and small-size devices is described on the whole reffering to the roles which those devices should play in ITER Physics R&D. It is not our intention to cover whole area in detail. With respect to each topics, the more detailed review and investigation will be necessary.

In this report, it is pointed out that the heat flux density expected in a fusioning plasma will become as large as an unexperienced high level in the existing devices except TFTR, and it is not self-evident that one could extend the present energy confinement time scaling to predict ITER performances.
The confinement of fusion plasma is considered to be related to heat, momentum and mass transport problems in a nonlinear non-equilibrium system with a finite source and dissipation. Therefore, a recent progress of basic experiments and theories in fluid dynamics may greatly be referred. In that area, series of basic experiments are unveiling various modes of heat and mass transport in the nonlinear non-equilibrium system including a scenario and conditions of transition to turbulence. In the field of plasma confinement, the more complex scenario would be expected. Therefore, a systematic and basic research along the same line is highly desired on the medium- and small-size devices in order to get insight into the physics of plasma confinement.

Acknowledgement

The authors would like to acknowledge Drs. K. Yamamoto, M. Fujiwara, H. Obayashi, M. Wakatani, H. Toyama, K. Itoh, A. Fukuyama, S. Matsuda, Y. Shimomura T. Tsunematsu for fruitful discussions. In particular, discussions with K. Itoh is very helpful to make our points clear and valuable informations about ITER physics and engineering design are owed to Drs. T. Tsunematsu and Y. Shimomura. ITER reports are also due to Dr. S. Matsuda.

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Table Captions

Table 1 The parameters of the medium- and small-size devices nominated for the long term ITER Physics R&D.

Table 2 Heat ouflux density through the unit plasma surface.

Table 1

	Device name	Basic parameters					Additional heating (MW)						remarks
		R_0 (m)	a (m)	b/a	B_0 (T)	I_P (MA)	NBI (MW)	ICRH (MW)	ECRH (MW)	LH (MW)	FW (MW)	IBW (MW)	
J	JT-60U	3.4	1.1	1.8	4.2	6(div)	40	5		15			SN. general evaluation
A						7(lim)							
P	JFT-2M	1.31	0.35	1.51	1.5	0.5	1.6	3.0	0.25		0.6		SN/DN. confinement improvement
A	JIPPT-ⅡU	0.91	0.23	1	3	0.25	0.7	2		0.2		0.5	CD. high power density heating
N							0.5						
	TRIAM-1M	0.8	0.26	1.46	8	0.1				0.05			SC. 1hr discharge. CD
	WT-3	0.65	0.2	1	1.75			0.25	0.2	0.35			CD.ECH
	HT-2	0.41	0.1	1.5	2.5	0.05							CD. disruption control
	CSTN-III	0.4	0.1	1	0.1	0.01							heat&particle control
	HYBTOK-II	0.4	0.1	1	0.4	0.2							heat&particle control
	Heliotron-E *	2.2	0.2	(2)	2	0	4	2	1				heat&particle control
E	JET	2.96	1.25	1.7	3.4	7	20	30		10			SN. general evaluation
C	TORE SUPRA	2.25	0.7	1	4.5	1.7	4	6	2.4	6			SC. 30sec discharge. CD
	TEXTOR	1.75	0.5	1	2.6	0.65	2.5	4.5					PL. plasma wall interaction
	ASDEX-U	1.65	0.5	1.6	4.0	1.6	6	6					SN
	FTU	0.92	0.31	1	8.0	1.6			1.2	9		1.5	high toroidal magnetic field
	TCV	0.88	0.24	3.0	1.4	1.20					2(AW)		extreme non-circular. AWCD
	T.de VARENNES	0.85	0.27	1.0	1.5	0.3				1.4			CD
	RTP	0.72	0.18	1.0	2.8	0.20			0.6				
	COMPASS	0.56	0.23	1.7	1.9	0.4			2				disruption control
	START	0.22	0.18	1.6	0.5	0.2			2				
U	T-15	2.43	0.7	1	3.5	1.4	9		10				
S					4.5	2.3							
S	T-14	1.06	0.33	1	5	0.48	2	4					compression
R		0.4	0.12	1	12.8	1.2	2	4					
	T-3M	1.06	0.28	1	4.0	0.4	2						
		1.06	0.14	1.8	4.0	0.2	2.						non-circular
	TUMAN-3	0.55	0.24	1	1.2	0.2		1					
	FT-2	0.55	0.09	1	3.0	0.05				0.05			
	TVD	0.36	0.07	3	1.5	0.04			0.2			0.2	extreme non-circular
U	TFTR	2.5	0.9	1	5.2	3	30	6					general evaluation(w/o divertor)
S								14					
A	DIII-D	1.67	0.67	2.5	2.2	3.5	14	2	2			2	SN. high β CD
	PBX-M	1.65	0.3	3	1	0.6	7			2		1	bean shape. high β
												7	
	CCT	1.5	0.35	1.2	0.3	0.1		0.2	0.01	0.2		0.1	CD. fuelling
	TEXT	1	0.25	1	2.8	0.4			0.6				disruption. confinement improv.
									1.6				
	Alcator-C/Mod	0.65	0.21	1.8	10	3		6					high Bt. non-circular
	MTX	0.64	0.16	1	12	1			1.8				high Bt. FEL heating
	VERSATOR II	0.4	0.13	1	1.5	0.07			0.1	0.25			

Table 2

Device name	Plasma surface (m²)	Input power (MW)	Heat outflux density (kw/m²)
ITER	780	300	385
JET	200	14	70
JT-60	110	20	182
JT-60U	170	40	235
TFTR	76	30	395
DIII-D	58	12	207
ASDEX	26	4.4	169
PLT	21	3	143
ISX-B	11	2.5	227
JFT-2M	19	1.5	79
JIPPT-IIU	8.3	2	241