Proceedings of 1995 The First Taedok International Fusion Symposium on 
Advanced Tokamak Researches

March 28-29, 1995
Korea Atomic Energy Research Institute (KAERI)
Taedok Science Town, Taejon, Korea

Sponsored by:
Korean Physical Society
Korean Nuclear Society
Tokamak Research for a steady-state fusion reactor

H. Kishimoto (JAERI-Naka, Japan)
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Preface

This proceeding is from the First TAEDOK International Fusion Symposium on Advanced Tokamak Research, which was held at Korea Atomic Energy Research Institute (KAERI), Taedok Science Town, Korea, on March 28-29, 1995. The TAEDOK Symposium, where "TAEDOK(大德)" is the old name of the region in Korea where KAERI and the Science Town are located, has been an important part of our sincere institutional efforts for international cooperation in the research and development for the peaceful and safe use of nuclear energy including fusion.

The advanced tokamak is a newly emerging concept in fusion research, addressing the issue of the necessary economy for a commercial fusion reactor. The concept is expected by many to make the earlier deployment of economically competitive fusion power plants possible, when properly supported by theoretical refinement and technological development. Therefore, we believe the successful continuation of the TAEDOK symposium, which will keep a focus on such an important issue as advanced tokamak, is very important for international fusion community as well as for its host, KAERI and Korea. Needless to say, your interest and participation is essential.

Many constraints forced us to start this year's first TAEDOK symposium only with invited lectures. However, we will make the second TAEDOK symposium a truly international, although small, forum for dedicated discussions on the advanced tokamak issues, organized by International Organization/Programme Committees. We at KAERI and in Korea will be very happy to help the International Committees through local administrative supports, and thus to help make the symposium a success.

I wish the symposium proved, and will prove to be very productive and useful to all the participants and prospective participants from home and abroad. For guests from abroad, I wish you all enjoyed a very pleasant stay in the Science Town and in Korea. I will look forward to meeting you, along with many new participants, again in the Second TAEDOK Symposium.

Dr. Sung Kyu Kim
Symposium Secretary
Nuclear Fusion Laboratory, KAERI
1995 The First Taedok International Fusion Symposium
on Advanced Tokamak Researches

Time: 1995. 3. 28(Tuesday) - 29(Wednesday)
Place: Seminar Hall, 2nd floor, Main Building
                   Korea Atomic Energy Research Institute (KAERI), Taejon, Korea
Organized by: KAERI
Sponsored by: Korea Nuclear Society, Korea Physical Society

March 28, Tuesday

9:00 - 9:30 Registration
Opening Ceremony
9:30 - 9:40 Opening Dr. T. N. Lee (Chairman, KT-2 DAC)
9:40 - 9:50 Welcome President, KAERI

Session 1: Keynote
10:00 - 11:10 Opening Lecture H. Kishimoto (JAERI-Naka, Japan)
Tokamak Research for a steady-state fusion reactor
Chair: T. N. Lee (Postech)

Session 2: Experiments/Plans
11:10 - 12:10 Invited Lecture Ila D. C. Robinson (UKAEA-Fusion, UK)
Opportunities for advanced tokamak research on medium-sized machines
Chair: K. H. Chung (SNU)
12:10 - 13:30 LUNCH

13:30 - 14:30 Invited Lecture lib Jinchoon Kim (General Atomics, USA)
Role of the plasma rotation in improved confinements in tokamaks
14:30 - 15:30 Invited Lecture lic V. Strelkov (RSRC "Kurchatov", Russia)
ECH and ECCD at T-10 tokamak and at ITER

15:30 - 15:40 COFFEE BREAK

15:40 - 16:20 J. Xie (IPP-Hefei, China) Superconducting tokamak HT-7 project at IPP-Hefei
16:20 - 17:00 M. Mori (JAERI, Japan) Recent experimental research on JFT-2M
17:00 - 17:40 S. K. Kim (KAERI) Overview of the KT-2 design in view of advanced tokamak researches

18:00 - 20:00 RECEPTION
March 29, Wednesday

Session 3: Theory/Computation
Chair: D.-/ Choi (KAIST/KBSI)

9:00 - 10:00 Invited Lecture IIIa F. C. Schüller (FOM-Rijnhuizen, The Netherlands)

Self-organized microstructures and their role in tokamak transport

10:00 - 11:00 Invited Lecture IIIb C. S. Chang (KAIST/NYU/PPPL)

Effects of α-particle driven current in advanced tokamaks

11:00 - 11:10 --- COFFEE BREAK ---

11:10 J. K. Lee (Postech, Korea) MHD stability of highly shaped plasmas in large aspect-ratio tokamaks

11:50 B. G. Hong (KAERI, Korea) Simulation characteristic of the operation modes in KT-2 tokamak

12:30 - 14:00 --- LUNCH ---

Session 4: Fusion Engineering
Chair: M. K. Chung (KAERI)

14:00 - 15:00 Invited Lecture IVa R. Andreani (ENEA-Frascati, Italy)

Engineering R&D issues for steady-state tokamak fusion reactors

15:00 - 16:00 Invited Lecture IVb J. Huang (Southwest Institute, China)

Design and R&D activities for fusion reactor program in China

16:00 - 16:10 --- COFFEE BREAK ---

16:10 - 16:50 S. H. Chang (KAIST) Critical heat-flux issues in fusion reactor diverter design

16:50 - 17:30 J. Wyss (Thomcast) Progresses in high-power RF transmitter systems development for tokamak fusion reactor application

17:30 - 18:10 G. W. Hong (KAERI) Superconducting magnet development at KAERI

Closing ceremony
Chair: S. K. Kim (KAERI)

18:10 Announcements
18:15 Closing
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Tokamak Research for a steady-state fusion reactor

H. Kishimoto (JAERI-Naka, Japan)
TOKAMAK RESEARCH
FOR A STEADY-STATE FUSION REACTOR
Hiroshi KISHIMOTO
(Japan Atomic Energy Research Institute)

An attractive concept of steady state tokamak fusion reactor has been developed on the basis of experimental demonstration of high bootstrap current fraction discharges in large tokamaks. Recent tokamak experiments have succeeded to establish more strong scientific evidences for this reactor concept; simultaneous attainment of high-\(\beta_p\), high-\(\beta_n\), high bootstrap current fraction, high improved confinement factor, and high ion temperature discharges sustained with a fully non-inductive current drive. Critical issues for developing a steady state tokamak reactor have now become clear in both physical and engineering areas.

1. INTRODUCTION

Fusion energy offers the potential of the environmentally acceptable, economically competitive energy source with a virtually unlimited and widely available fuel supply. Growing world population, extending world-wide industrialization, environmental degradation, and concerns about the security and availability of present fuels have stressed the need for the fusion energy development. Nevertheless, fusion energy must be feasible as an energy source to attain the public acceptance and long-term efforts are requested for its development.

Tokamak is the top runner of the fusion development race. Tokamak plasma parameters have reached already around the break-even conditions as shown in Fig. 1 [1, 2]. Such tokamak plasmas are sustained basically by an inductively driven current and then a tokamak reactor could be a pulsed-power generator. An inductively driven tokamak reactor is simple in a physics sense but it is less feasible as a power plant [3]. The large central solenoid coil for current induction enlarigs the tokamak size. Repetitive electromagnetic forces and heat loads give significant fatigue damages to tokamak machine components. Pulsed operation reduces the plant availability. Pulsed power generation disturbs largely the utility power-line stability. And consequently, the electricity cost of an inductively driven tokamak reactor is intrinsically much higher than that of a steady state reactor. These difficulties have been preventing the practical concept development of a tokamak power reactor based on the inductive current drive.
To achieve the power plant feasibility, steady state burning of plasmas is inevitably required for a tokamak fusion reactor. The first critical issue of a steady state burn is the development of high efficiency non-inductive current drive scheme. However, any non-inductive current drive method (beam injection or radio-frequency waves) does not allow us to achieve sufficient efficiency in a reactor-relevant plasma density region. First demonstration of discharges with a high bootstrap current fraction beyond 80 percents of plasma current in JT-60 gave us a prospect to overcome this difficulty [4]. Based on this experiment, a new and practical concept of steady state tokamak fusion power reactor (SSTR) was developed in JAERI in 1989 [5]. In the same year the UCLA group proposed independently the similar steady state tokamak reactor concept (ARIES-I) [6]. Ever since the tokamak plasma research has progressed successfully to establish the scientific basis of this reactor concept. Many efforts have been made also for the technology development.

### 2. STEADY STATE TOKAMAK REACTOR CONCEPT

#### 2-1. Reduction of Recirculating Power

A feasible power plant essentially requires the reduction of recirculating power in the system to improve the plant power efficiency. Pulsed operation of a tokamak is associated with the joule loss in the poloidal power supply and the AC loss in the
super-conducting magnets. These losses require a large recirculating power beyond 100 MWe in addition to the usual inplant power consumption [3, 7]. In the meanwhile, the steady state tokamak reactor must be operated with the non-inductively driven current, which requires the power for current drive. High bootstrap current operation of a tokamak reactor has a substantial benefit to reduce the power for current drive and to improve the plant efficiency. The power flow of a steady state power reactor is given in Fig.2.

![Diagram of power flow in steady state tokamak reactor](image)

**Fig. 2. Power flow in steady state tokamak reactor**

The bootstrap current has a hollow shape profile in tokamak discharges and hence the current profile optimization for MHD stability becomes essential with non-inductive current drive near the plasma center [8,9]. Recent tokamak experiments and theoretical MHD analyses have predicted that the bootstrap current fraction is marginal around 80 percents and the residual 20 percents of plasma current should be driven non-inductively. Minimizing the plasma current to sustain the high-Q burning plasmas is another crucial issue of a steady state tokamak reactor to reduce the recirculating power.

2-2. High Bootstrap Current and High-$\beta_p$

The bootstrap current fraction $f_b(=l_{\text{bootstrap}}/l_p)$ increases with $\beta_p$ as expressed by

$$f_b \sim (a/R)^{1/2} \beta_p \sim (R/a)^{1/2} \beta_p q,$$

where $a$ is the plasma minor radius, $R$ the major radius, $\beta_p$ the poloidal beta, $\beta_n$ the normalized beta, and $q$ the safety factor. The Troyon relation $<\beta_i> = \beta_n l_p / a B_i$ with the
volume-averaged toroidal beta \( \langle \beta_t \rangle \) and the toroidal magnetic field \( B_t \) gives
\[
\beta_p \langle \beta_t \rangle = \beta_p^2 \kappa/4, \tag{2}
\]
where \( \kappa \) is the plasma ellipticity. It is found from these relations that a high-\( \beta_p \) operation is beneficial to achieve a high bootstrap current fraction \([5]\). This choice shows a clear contrast to that of an inductively driven tokamak reactor such as ITER, which employs a high current and low-\( \beta_p \) approach.

The normalized beta \( \beta_N \) is around 2.5-3 from a stability limit and is potentially improved by the current profile optimization. From this viewpoint, the current profile control to achieve high \( \beta_N \) should be important for both steady state (high \( \beta_p \)) and pulsed (high \( \beta_t \)) tokamak reactors. The above relations indicate that the parameter selection of low plasma current \( I_p \), high toroidal field \( B_t \), high aspect ratio \( R/a \), high safety factor \( q \), and high elongation \( \kappa \) is appropriate for high bootstrap fraction \( f_b \) and high poloidal beta \( \beta_p \) discharges in a steady state tokamak reactor.

2-3. High Aspect Ratio and High \( B_t \)

In the above considerations, it is found that the low plasma current discharges are feasible to a steady state tokamak reactor while the tokamak confinement is degraded with decrease of the plasma current. We must take the parameter choice to improve the confinement even at a low plasma current.

The tokamak L-mode scaling such as Takizuka scaling \([10]\) or ITEM-89P scaling \([11]\) suggests that the fusion triple product \( n\tau T \) can be given by the parameters as following:
\[
 n\tau T \sim (H l_p R/a)^2 q^{0.4-0.8}
\]
where \( H \) is the confinement improvement factor against the L-mode scaling. This parameter dependence of fusion product allow us to reduce the plasma current without degrading the fusion performance through the increase of aspect ratio \( R/a \) as well as of safety factor \( q \). The increases of \( R/a \) and \( q \) contribute also to enhance the improved bootstrap fraction. The larger \( q \) value requires the higher \( B_t \) and the lower \( I_p \). The choice of too small \( I_p \) does not guarantee the high fusion performance and therefore the highest possible \( B_t \) must be selected for a steady state fusion reactor based on a high bootstrap scenario.

2-4. Parameters for Power Reactor

Typical Parameters designed for steady state tokamak reactors are given below:

Both reactor designs adopt almost the same plasma parameters, but SSTR has the design base of the conservative technology while ARIES-I uses very aggressive technology such as high field (\( B_t = 11.3 \) T) and ceramics materials SiC in the blanket.
### Table

<table>
<thead>
<tr>
<th>ITEM</th>
<th>SSTR(JAERI)</th>
<th>ARIES-I(UCLA)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plasma current $I_p$</td>
<td>12 MA</td>
<td>10.2 MA</td>
</tr>
<tr>
<td>Toroidal field on major radius $B_t$</td>
<td>9 T</td>
<td>11.3 T</td>
</tr>
<tr>
<td>Major radius $R$</td>
<td>7 m</td>
<td>6.8 m</td>
</tr>
<tr>
<td>Minor radius $a$</td>
<td>1.7 m</td>
<td>1.5 m</td>
</tr>
<tr>
<td>Aspect ratio $R/a$</td>
<td>4.1</td>
<td>4.5</td>
</tr>
<tr>
<td>Elongation $\kappa$</td>
<td>1.85</td>
<td>1.6</td>
</tr>
<tr>
<td>Safety factor at 95% flux surface $q_{95}$</td>
<td>4.5</td>
<td>4.75</td>
</tr>
<tr>
<td>Toroidal beta $\langle \beta_t \rangle$</td>
<td>2.7 %</td>
<td>1.9 %</td>
</tr>
<tr>
<td>Normalized beta $\beta_n$</td>
<td>3.5</td>
<td>3.2</td>
</tr>
<tr>
<td>Bootstrap current fraction $f_b$</td>
<td>0.75</td>
<td>0.68</td>
</tr>
<tr>
<td>Fusion power $P_f$</td>
<td>3 GW</td>
<td>1.9 GW</td>
</tr>
<tr>
<td>Net electric power output $P_e$</td>
<td>1.1 GW</td>
<td>1 GW</td>
</tr>
<tr>
<td>Plant efficiency $\eta_p$</td>
<td>0.3</td>
<td>0.39</td>
</tr>
<tr>
<td>Coolant</td>
<td>light water</td>
<td>Helium gas</td>
</tr>
<tr>
<td>Blanket material</td>
<td>ferritic steel</td>
<td>ceramics SiC</td>
</tr>
</tbody>
</table>

### 3. PROGRESS IN FUSION PLASMA PHYSICS

#### 3-1. High Bootstrap Operation

High-$\beta_p$ and high bootstrap current discharges occupy the essential part of a steady state tokamak reactor concept. Success of the high-$\beta_p$ mode in JT-60 led to the high bootstrap operation [4]. Figure 3 shows that the bootstrap fraction $f_b$ increases with $\beta_p$ and the maximum $f_b$ reaches around 80% in JT-60, which exceeds the bootstrap fraction requirement of the steady state tokamak reactor design.

The high-$\beta_p$ mode is restricted by the occurrence of $\beta_p$ collapse associated with the rapid growth of MHD activities near the plasma center [12]. The key issues to achieve the high-$\beta_p$ and high bootstrap fraction required by a reactor are the selection of safety factor $q$ and the current and pressure profile control. The experiments in JT-60U indicates in Fig.4 that the high $\beta_p$ as well as high $\beta_n$ is guaranteed by a high $q$ value [8]. The high-$\beta_p$ plasmas are obtained in the peaked pressure and broad current profile discharges while these plasmas suffer from the $\beta_p$ collapse. The standard H-mode has broad pressure profiles and medium current profiles but both $\beta_p$ and $\beta_n$ in the H-mode are constrained by the occurrence of Edge Localized Modes (ELMs). The modest pressure profile and the most peaked current profile improve the normalized beta $\beta_n$ as well as $\beta_p$. 

1 - 7
3-2. Confinement Improvement

High-\(\beta_p\) discharges can be obtained by an intense central heating of plasmas. The high-\(\beta_p\) mode improves the energy confinement beyond the L-mode. The formation of transport barrier around the \(q=3\) region has been observed in the recent high-\(\beta_p\) experiments of JT-60U [13]. This barrier improves the ion energy transport but is not effective to improve the electron heat transport as well as the particle transport.

JT-60U has demonstrated the formation of double transport barriers near the \(q=3\) surface and the plasmas periphery. The later barrier is associated with the normal H-mode. This double transport barrier confinement mode is denoted as the...
high-$\beta_p$ H-mode and has achieved the highest fusion triple product $nT = 1.2 \times 10^{21}$ keV·s/m³, which is about 1/5 of the self-ignition reactor condition.

ELMs are effective to exhaust particles from a plasma through the peripheral barrier and to sustain the plasma particle balance although the occurrence of ELMs degrades the confinement. Degradation is not significant in the high-$\beta_p$ H-mode even with ELMs because the internal barrier sustains the energy confinement. The improved confinement of ElMy discharges depends on the safety factor $q$ as shown in Fig. 5: the H-factor decreases with the decrease of $q$ and a typical design value of $H=2$ in a reactor concept requires $q_{\text{eff}}>4$ ($q_{\text{gs}}\approx 0.7-0.8 \times q_{\text{eff}}$ in JT-60U). The high $q$ operation is desired again for a steady state confinement improvement with ELMs.

![ELMy H-mode, dW/dt=0](ELMy_H-mode_.dW/.dtt=0.jpg)

**Fig. 5 Confinement improvement factor $H$ versus safety factor $q_{\text{eff}}$ for ElMy H-mode discharges in JT-60U**

3-3. Current Profile Control

In high $\beta_p$ discharges, the current profile is coupled with the pressure profile and has a hollow shape. Stabilizing and destabilizing effects of negative shear associated with the hollow current profiles have been studied theoretically [9]. Current and pressure profile optimization is essential to improve the $\beta_n$, $\beta_p$ limits and the confinement as indicated in Fig. 6. In the steady state tokamak reactor design, high energy beams (> 1 MeV) are employed for the current profile control near the center for high-$\beta_p$, high bootstrap plasmas.

The experiments on JT-60U shows that the high bootstrap discharges with negative shear ($l_p/\beta_p \approx 0.6$ and the residual current is driven by beam) have a better confinement than the discharges with the monotonically increasing $q$ profile if the low $n$ modes are able to be suppressed. The excitation of the low $n$ modes depends on the minimum $q$ value.
3-4. Core-Plasma Demonstration for Steady-State Reactor

By optimizing the current and pressure profiles, fully noninductive current drive with combination of bootstrap current and beam driven current was demonstrated at $I_p=1$ MA and $B_t=3$ T ($q_{\text{eff}}=7.1$, $q_{95}=5.2$) in a high-$\beta_p$ H-mode with ELMs. Time evolution of typical parameters is shown in Fig. 7. The fully noninductive current drive plasma was sustained for about 0.7 s and the bootstrap current fraction of 74% and the beam driven current of 37% (over drive) were attained with a high plasma performance: $\beta_p=2.6$, $\beta_N=2.9$, H-factor=2.5, and $n_e(0)\tau_e T_e(0)=0.8 \times 10^{20}$ m$^{-3}$keV [8]. The data set ($\beta_p$, $\beta_N$, h-factor, bootstrap current fraction, etc.) of this discharge satisfies almost the design parameters of a steady state tokamak reactor (SSTR). The ion temperature at the plasma center was around 30 keV and the electron temperature was 3 keV.

Faraday rotation measurements at $r/a=0.3$ and 0.7 show that a flat current profile is maintained during the NB injection. No MHD activities except ELMs was observed. The success of this discharge provides an confidential basis of the high bootstrap steady state tokamak reactor.
3-5. Power and Particle Control

Confinement improvement and non-inductive current drive relevant to a reactor have been demonstrated experimentally. However, particle and heat exhaust is still the key issue of tokamak research.

The pumped divertor experiments on DIII-D was successful to exhaust particles from ELMy H-mode discharges [14]. The H-mode plasma density was controlled well and the helium ash exhaust was effective with the divertor.

In the meanwhile, remote radiative cooling of the plasma heat by a dense and cold divertor plasma has been investigated in many tokamaks. The radiative cooling enhanced with increase of the main plasma density $n_e$ because the divertor density increases with $n_e$ resulting in an effective cooling of heat flow from the main plasma. However, as $n_e$ rises up to near the discharge density limit, the ionization zone of a divertor plasma leaves the divertor plates. This detached divertor plasma develops to the MARFE (Multifaceted Asymmetric Radiation From the Edge) as shown in Fig. 8 and then the divertor functions such as heat shielding from the main plasma and
the suppression of particle back-flow from divertor plates are deteriorated significantly [15].

The radiation cooling up to 90% of plasma heat flow has been observed in the medium size tokamaks [16] but the cooling fraction in large tokamaks does not exceed about 60-70% owing to the MARFE formation as shown in Fig. 9. Divertor study to enhance the remote cooling of plasma and the particle exhaust without deteriorating the confinement performance is surely the most important subject of tokamak research for a reactor.

Fig. 8 Plasma detachment and MARFE formation in JT-60U

Fig. 9 Radiation loss fraction from divertor plasma versus main plasma density
4. ENGINEERING DEVELOPMENT

4-1. Critical Issues for Engineering Development

Extensive engineering R&D efforts for ITER are now in progress. Major R&D issues are; superconducting magnet technology for toroidal and poloidal field coils, divertor and first wall, heating and current drive, shield and blanket, tritium fuel system, remote maintenance, etc.. Specific R&D items for a steady state tokamak reactor are addressed as high field super-conducting magnet, divertor, negative NBI for heating and current drive, and low-activation structural materials.

4-2. High-Field Superconducting Magnet Technology

A high toroidal field ($B_t \geq 9$ T) has a special benefit for the steady state operation of a tokamak reactor at high $q$ and high $\beta_p$. The maximum field in the conductor is to be larger than 16 T as indicated in Fig. 10.

![Graph showing progress of superconducting magnet technology for tokamaks](image)

Fig. 10 Progress of superconducting magnet technology for tokamaks

A small scale coil made of Ti-doped Nb$_3$Sn conductor was developed by a tube method in JAERI and demonstrated the generation of 18 T field. The SSTR design adopted this conductor working at the maximum field of 16.5 T.

Recently remarkable progress has been made in the development of Nb$_3$Al conductor for high field large coils required by DEMO [17]. The Nd$_3$Al conductor has a good property of critical current density versus strain: the critical current degradation of the Nb$_3$Al conductor is only 5% at a typical 0.4% strain under 12 T
while that of the Nb$_3$Sn conductor is up to 30% in the same condition. The filament diameter of the Nb$3$Al is still large and is suitable for high field DC operation coils. These development results have forced the JT-60SU design to change from Nb$_3$Sn to Nb$3$Al.

The development of high-strength cryogenic steel is also important to achieve the reliability and the compactness of the magnet system. A high yield strength cryogenic steel JN1 with $\sigma_Y>1200$MPa has been developed for this purpose.

4-3. High Heat Flux Divertor Plates

High heat removal plasma facing components are essentially required especially for the divertor plates of a steady state reactor. A swirl tube, which has a twisted tape insert, has been developed and has demonstrated the highest heat removal capability at the tube surface of 38 MW/m$^2$ with a 10 m/s - 1 MPa cooling water [18]. A high heat transfer joint technology between the armor tiles and the cooling tube is the another critical issue. The armor tiles of carbon-carbon composites have been brazed onto heat sinks made of oxygen-free high conductivity copper tube with the swirl structure and have been able to endure the 20 MW/m$^2$, 30 s, $10^{3}$ thermal cycles.

Fig. 11 Divertor heat load of tokamaks
This armor structure has now the highest heat removal capability and its heat removal capability is shown in Fig. 11 as a function of burning time. The heat load of the present day tokamak divertors is below this critical limit. However, the divertor heat load of ITEM probably exceeds the technical limit with a long burning time and the steady state DEMO or power reactor must endure the heat load more than 4 times of the limit. Two approaches should be taken: one is to reduce the divertor heat load by developing a plasma control method for remote plasma cooling and another is the further technical development of heat removal.

4-4. Negative NBI System for Heating and Current Drive

High efficiency of heating and current drive, as well as high potential of current and pressure profile control, is the basic requirement to achieve a steady state operation of tokamak. Furthermore, momentum control capability is very beneficial to suppress the locked modes and also to improve the confinement through plasma rotation [13]. The negative NBI satisfies these requirements and intense development efforts have been made for the technologies [19]. The 500 keV, 10 MW N-NBI system is under construction for JT-60U. The MeV-level test bed is also in progress for ITER. These R&Ds are accessing the DEMO or ITER target of the N-NBI as shown in Fig. 12 and are able to provide the sufficient technical bases for the reactor N-NBI units.

4-5. Low-Activation Structural Materials
The proper selection and development of low-activation structural materials is a key to the achievement of the ultimate benefits of fusion as a safe, environmentally attractive and economically competitive energy source. The material is used at elevated temperatures while exposed to high neutron fluxes, high primary and secondary stresses, transient electromagnetic loads, a high temperature coolant and tritium breeding material, and hydrogen environments associated with plasmas.

Three stages of fusion material development are now considered as illustrated in Fig. 13. Austenitic steels are used in the present tokamaks and are considered as the candidate material of ITER [20]. Austenitic steels have the board industrial bases and sufficient experience in fusion but have luck of feasibility as a low activation candidate.

The SSTR concept employs the ferritic steels [5]. This material has also the board industrial bases but it has no experience in fusion because of its ferromagnetic property. The ferritic steels are feasible to be used at a temperature less than 450 °C and therefore the coolant must be the pressurized light water.

Vanadium-based alloys offer several advantages as a structural material such as high operating temperature up to 700 °C and good compatibility with lithium and PbLi alloy for a coolant [21]. However this kind of alloys has no sufficient industrial experiences. The SiC/SiC composites are the most aggressive selection of fusion materials on the basis of safety and environmental considerations and ARIES-1

![Fig. 13 Potential candidates of fusion structural material](image)
studies it [6]. This material has superior advantages of low activation characteristics and high temperature capability to about 1000 °C, which indicates the possibility of a helium cooled high efficiency power plan, but the most important constraints are the development of design methods of this materials, projected costs and the very high helium transmutation rate.

Development of fusion materials is the long term R&D subject and the international collaboration including the development of intense neutron source for irradiation tests will be pursued under the IEA framework.

5. Research and Development Programs

US and JAERI are proposing the next step devices TPX [22] and JT-60SU [22]. These programs are expected to contribute substantially to the development of a steady state tokamak reactor as complementary to ITER.

6. Conclusions

Significant physical and technological progress has been made towards a steady state tokamak power reactor. The key issues in the future development are the divertor physics and the high heat load components technology as well as the selection and development of low activation materials. Extending international collaborations is more essential to develop the fusion energy as a desire of all mankind in the coming 21th century.

REFERENCES


Opportunities for advanced tokamak research
on medium-sized machines

D. C. Robinson (UKAEA-Fusion, UK)
OPPORTUNITIES FOR ADVANCED TOKAMAK RESEARCH ON MEDIUM-SIZED DEVICES

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## CONCEPT IMPROVEMENTS

- As well as addressing problems for ITER, the international programme is seeking "concept improvements" which would be more attractive for a DEMO than the ITER-like tokamak.

- The most promising are the advanced tokamak, the spherical tokamak and the stellarator, though the reverse field pinch is also under consideration.

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'R&D PATH TOWARDS THE FIRST ELECTRICITY PRODUCING FUSION POWER PLANT (DEMO)'
"The aim of current drive experiments in this phase should be the demonstration of steady-state operation in plasmas having alpha heating power at least comparable to the externally applied power. Using the heating systems in their current drive mode, non-inductive current drive should be implemented for profile and burn control, for achieving modes of confinement, and for assessing the conditions and power requirements for the above type of steady-state operation. Depending on the outcome of these experiments, additional current drive power may have to be installed."

1) CURRENT DRIVE POWER SUFFICIENT FOR ASSESSING THE CONDITIONS AND POWER REQUIREMENTS

   initial: about 50MW

2) ADDITIONAL POWER CAN BE ADDED

   (same or different method[s])
   total: about 100MW

---

Steady State Operation

Conventional Scenario

\[ Q=4 \text{ - } 6, \ P_fusion=500 \text{ - } 1000 \text{ MW}, \]
\[ P_{cd}=120 \text{ - } 150 \text{ MW} \]
\[ \text{with } H=2, \ n=(5 \text{ - } 8)\times10^{19}\text{m}^{-3}, \]
\[ Z_{eff}=2.5 \text{ - } 3, \]
\[ I_b/I_p=(0.3 \text{ - } 0.5), \ q=4 \text{ - } 5, \]
Troyon parameter=2 \text{ - } 2.8

Advanced Scenario

If an advanced scenario is developed, a high Q and high fusion power operation will also be possible in ITER
ADVANCED TOKAMAKS

- A major drawback of the tokamak is that it is a pulsed device because its current is driven by a transformer. This results in thermal cycling damage to components and, for continuous electricity, in reactors running in parallel on a site or costly energy storage systems.

- In the advanced tokamak operation is continuous with the current driven by a combination of internal, pressure-driven processes ("bootstrap currents") and RF waves or neutral particle beams.

- To avoid unacceptably high recirculating power, >80% of the current must be pressure-driven. This requires higher plasma pressures and confinement times greater than normal. If the concept proves viable a DEMO would have a smaller volume and lower current than ITER, and therefore be cheaper than a conventional tokamak DEMO.

- As this is such a promising concept both the US (TPX) and Japan (JT60-SU) are planning large experiments costing ~$1 Billion. Many medium-sized current experiments, including COMPASS-D, TORE SUPRA in Europe and DIII-D in US, are addressing advanced tokamak issues, as well as the larger tokamaks TFTR, JT-60U and JET.
AN ATTRACTIVE FUSION REACTOR

- Steady state based upon maximising the bootstrap current \( I_{BS} / I_P \sim \sqrt{2R} \beta_p \geq 0.75 \)
- Moderate aspect ratio (A ~ 4.5)
- High field magnets (e.g. (Nb Ti)3 Sn)
- Second stability regime (high triangularity, q(0) > 1, wall stabilisation)
- Modest non-inductive current drive (NBCD, FWCD, ECCD, LHCD) with high efficiency and profile control
- Require high \( \beta_N \)

\[
\beta = \beta_N I / aB_T \quad \text{and} \quad \langle P \rangle \sim \beta_N I_P B_T / 2 \mu_0 a_p
\]

and \( H \left( H = \tau_e / \tau_{\text{iter}, 89P} \right) \), \( n \tau > \tau \sim (T A)^2 H^2 \)

e.g. ARIES I, SEA FP, SSTR, ARIES II/IV

\[ A = 4.5, I_P = 10 - 12 \text{MA}, B_T (T) = 9 - 11, R \sim 7 \text{m} \]

MHD STABILITY IN ADVANCED TOKAMAKS

- The need for a high bootstrap fraction in steady state reactors implies high \( \beta_p \) and low \( \beta \) unless the Troyon limit is exceeded by second stability or partial second stability operation.
- Need good alignment of bootstrap current with equilibrium current.
- Reversed central magnetic shear (non-monotonic q) appears to offer advantages over normal second stability profiles (q(0) > 2.0), leading to second stability in the central region and a good match of the bootstrap and equilibrium current profiles, but rely on wall stabilisation of low-n free boundary modes.

As \( \beta < C_T I_{aB}, C_T = 2.8 \), can be written as:

\[
\epsilon \beta_p \beta / \epsilon < \left( \frac{C_T}{20} \right)^2 \frac{1 + \kappa^2}{2}
\]

and high bootstrap requires \( \epsilon \beta_p \sim 1 \), then goal of advanced tokamak is to use current drive to increase \( \beta_N > C_T \)

UKAEA

\[ \text{UKAEA} \]

Government Division
Fusion
The goal of the advanced tokamak program is to optimize both energy confinement time and beta in steady-state non-inductively-driven discharge.

A reasonable target is to achieve $H=3$, $\beta_N=5$, and $f_{BS}=70\%$ simultaneously in steady-state.

These values have actually been achieved independently in a number of experiments for a finite duration.

For ballooning mode stability in the second regime requires low or negative shear near the centre, $q(o) \geq 2$ and sufficient triangularity, non-zero edge current helps.

For $n=1$ (kink) stability need high shear near the edge (or low edge current)

Good bootstrap alignment requires peaked pressure profiles and flat current profiles.

**First Stability**

\[ \frac{I_{BS}}{I_p} < 0.7 \] but kink stable without a wall

**Second Stability**

a) \[ \frac{I_{BS}}{I_p} > 0.9, q(o) \sim 2, \beta_N \sim 5 \] centrally peaked current profile, kink stability requires a wall

b) \[ \frac{I_{BS}}{I_p} > 0.9, q(o) \sim 2.5, \beta_N \sim 6 \] current peaked off axis, negative central shear, kink stability requires a wall ($r/a < 1.3$) ($\beta_N = 2$ if no wall) also stable to resistive modes ($q_{\text{min}} > 2$)

**Problem** :- resistive wall mode?

**Solution** :- plasma rotation and plasma boundary inertia, viscosity and resistivity
Comparison of experimental beta limits and theoretical scaling showing the operational envelopes of several tokamaks.

Normalised beta $\beta_N$ as a function of $\varepsilon\beta_p$ and corresponding $q$ and pressure profiles. Stability is calculated without wall stabilization ($A=3.86$, $\kappa=1.7$, $\delta=0.1$)

- High $\beta_N$ with broad pressure profile (A) and peaked current, high $q(\theta) \sim 2$
- High $q(\theta)$ in ARIES II/IV to access second stability regime
- Wall stabilisation of kink mode then important
- Hollow current profile approach to enhance bootstrap current ($\beta_N \sim 3$ no wall, $\beta_N \sim 4.8$ with wall ($r/a \sim 1.3$))

Note that $\beta_N > 3.5$ has been achieved on several devices.
Dependence of the calculated n=1 kink $\beta$-limit on the current density profile, $\ell_i$ and pressure profile

- Single null divertor configuration with $q \sim 3.1$ and a conducting wall at $r/a \sim 1.5$
- High $\beta_N$ at high $\ell_i$ with broad pressure profile
- High $\ell_i$ can be produced transiently with negative current ramps but how to maintain them in steady state in the face of the bootstrap current?

WALL STABILISATION BY PLASMA ROTATION

- Wall stabilisation essential for the n=1 kink mode at high $\beta_N$
- Plasma rotation of a few % of Alfvén speed required
- Some evidence for this from DIII-D experiments
- $\beta$ limit can increase by up to a factor of two

- $n=1$ stability boundaries for the wall position for the 'plasma mode' (circles) and the 'resistive wall mode' (triangles) at different rotation velocities versus $\beta^*$. The figure refers to advanced tokamak equilibria with $q_0 = 2.5$, $q_{\text{min}} \sim 2.2$ and $q_S = 3.7$ (hollow symbols) and $q_S = 4.1$ (filled symbols). The plasma mode is stable below the top curve and the resistive wall mode is stable above the respective curve.
WALL STABILISATION AND PLASMA ROTATION

Ideal modes unstable with resistive walls but rotation can help

- Impact of plasma rotation on low-n stability in DIII-D
- $\beta$ exceeds values calculated without wall stabilisation
  
  $\beta_N > 4 \ell_i$

- $m=3, n=1$ 'resistive wall' mode is seen to grow as frequency decreases

- **Problem**: high $\beta$ phase is transient

[Diagram showing Toroidal Rotation Frequency (kHz) for q = 2, 2.5, 3]

WALL STABILISATION AND PLASMA ROTATION

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- **Problem**: high $\beta$ phase is transient

[Diagram showing Toroidal Rotation Frequency (kHz) for q = 2, 2.5, 3]
ENERGY CONFINEMENT

L-mode: □
ELMy H-mode ○
ELM-free H-mode ●

$\tau_{E}[\text{ms}]$

$n_e[10^{10}\text{m}^{-3}]$
Directed inside and outside launch (10 lines at 60GHz), $2\omega_{ce}$, $B_T = 1.2T$, $I_p = 150kA$

- Off axis complete absorption, with co and counter driven currents. Varying $B$ and which lines are employed, gives flexible choice of $P(r)$ and $J_{ECCD}(r)$.

- Current density - (Normalised to 0.7422E 02 kA/m²)
- ECRH Power absorbed - (Normalised to 0.174BE 03 kW/m²)

Radius $r/a$
High $\beta$ ECRH and ECCD on COMPASS

$q_{95} \sim 4, B_T \sim 1.2T$

- Disruptive case, preceded by a predominantly 2/1, MHD mode

- Non disruptive case
**BOOTSTRAP CURRENT ON COMPASS-D IN VLOOP = 0 DISCHARGES (SCENE)**

- Up to 75% of current could be bootstrap driven, but $P_\perp > P//?$ - the remainder by ECCD in the core.

- Low or reversed central shear is predicted but $n=1$ MHD mode destroys the plasma.

- Systematic scan of antenna launch angles from co-current drive direction to counter-current drive direction gives systematic increase in both maximum plasma pressure and energy confinement time.

- Plasma is terminated with a large predominantly $n=1, m=2$ MHD mode.

- Calculations indicate plasma current is approximately 70% bootstrap driven.
ADVANCED TOKAMAK STUDIES ON TORE SUPRA

- Stationary states with magnetic shear reversal achieved with lower hybrid current drive and $V_{\text{loop}}=0$. Improved core confinement and high $\beta_p$ obtained for up to 15 seconds.

- Current profile modifications influence plasma confinement and performance. Improved confinement and MHD stability in high $\ell_i$ discharges.

- Experiments at 4T, 0.8MA and at 2T, 0.4MA show clear shear reversal from polarimetry, ray tracing and resistive diffusion simulations.

The Rebut-Lallia-Watkins global scaling fits ohmic and low power ICRH/LHCD discharges on Tore Supra.
HRLW FACTOR VS POLOIDAL BETA ON TORE SUPRA

\[ H = \frac{W_{e}^{\text{exp}}}{W_{e}^{\text{RLW}}} = 2 \]

G.T. Hoang

- At 0.4MA \( \beta_N \) up to 1.0 observed, sometimes with MHD activity, possibly associated with \( n=1 \) infernal modes arising from near the zero shear region, which destroys the confinement enhancement.

TYPICAL LHEP DISCHARGE ON TORE SUPRA
FULL NON-INDUCTIVE DISCHARGE LHEP TRANSITION AND MHD ACTIVITY ON TORE SUPRA

THE L.H.E.P PHASE ON TORE SUPRA

Phase = 0 deg. Shot 16337

- $P_h$ (MW)
- $I_p$ (MA)
- $T_e$ (KeV)
- MHD

- $\chi_e (m^2/s)$
- $T_e(0)$
- LH power (a.u)

- $t = 12s$
- $\Omega_{ohmic}$
- $\Omega_{bootstrap}$
- Magnetic shear

- Normalised radius

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Ray tracing/Fokker-Planck code
IDENT-D code.
CRONOS SIMULATION COUPLED WITH OFF AXIS CURRENT PROFILE ON TORE SUPRA

LOCAL TRANSPORT ANALYSIS OF LHEP #TS14386 ON TORE SUPRA

With power deposition given by ray tracing/2-D Fokker Planck calculation

- Thermal diffusivities
- Temperature profiles
- Power deposition and current density profiles

X. Litaudon, E. Joffrin
Stationary Non-Inductive Lower Hybrid Enhanced Performance Conclusions (TORE SUPRA)

1) Experiments at 4 Tesla, \( I_p = 0.8 \text{ MA} \rightarrow \text{central shear} = 0 \) (<0?)
   a) LHEP regime when \( V_{\text{loop}} \) nearly vanishes
   b) First steady-state operation \( (E(r) + 0) \)
      - control of flux consumption/current floating
      - stability after full equilibrium \( (\text{MHD}?) \)
      - phase dependence
   c) Power deposition profiles validated through:
      - Hard X-ray emission data
      - Ray-tracing/2-D Fokker-Planck simulations
      - Full simulation of the time evolution of all profiles and integrated data \( (\text{CRONOS}) \).
   d) Transport analysis shows clear reduction \( (\text{LOCO}) \)

2) Experiments at 2 Tesla, \( I_p \approx 0.4 \text{ MA} \rightarrow \text{central shear} < 0 \)
   a) LHEP regime also observed at \( V_{\text{loop}} = 0 \)
   b) All data is consistent with significant shear reversal
      - off-axis power deposition \( (\text{agreement RTFP and WDFP}) \)
      - stability after full equilibrium \( (\text{MHD}?) \)
      - phase dependence
   c) Same consistent analysis as before
   d) Strong reduction of transport in centre

DIII-D

Comparison of DIII-D high beta results to two scaling relations (a) Troyon \( (I/aB) \) (b) inclusion of internal inductance, \( (\ell I/aB) \).

- \( \beta \) limit increases with peaked current density profiles (high \( \ell I \)) and broad pressure profiles - but compatibility with transport, \( \alpha \) particle heating and bootstrap current?
- Importance of wall stabilisation
Plasma performance in DIII-D improves with shaping (high triangularity)

Performance in strongly shaped DIII-D discharges is already close to our goal

- Stability is the dominant factor for improved performance
- Confinement improvement also linked to plasma flow ($V_\phi$) and flow shear, edge current density, $q(0)$, $j$, $p$, $q$ profiles.
SECOND STABLE CORE WITH REVERSED SHEAR

- Central shear reversal by rapid elongation (DIII-D) or pellet injection (JET)
- Bootstrap current and reduced transport help sustain reversed shear in JET

![Graphs and images showing reversed shear and plasma parameters.](image)

OPTIMAL PROFILES; DIII-D VIRTUAL DISCHARGE

- VH-Mode with Second Stable Core
- Stable to n=1 kink with DIII-D wall
- Stable to ballooning, marginal near edge

(E. Lazarus, Phys. Fluids 1992)

(R. Hugon, Nucl. Fusion, 1992)
Temporal evolution of negative current ramp discharges illustrating the operational scenario to obtain a variation in χ and high βN for (a) DIII-D, (b) TFTR, and (c) JT-60.

The best confinement in DIII-D is obtained with finite edge current density.
Improved core confinement is observed on DIII-D in high $q(0)$, high $\beta_p$ discharges

In DIII-D and TFTR, improved core confinement is observed in high $q(0)$, high $\beta_p$ discharges, sustained by NBCD and bootstrap current.

As $q(0)$ rises above 2, MHD fluctuations disappear and core confinement improves, most dramatically in the core density. This result is highly suggestive of a correlation in the DIII-D second stable core regime and the JET PEP-mode both with negative central shear. The negative $q'$ allows the core to enter into the second stable regime to ballooning modes.

It remains to demonstrate simultaneously both high $\beta$ and reduced transport for the negative shear operation.

ADVANCED TOKAMAK STUDIES IN TFTR

Conditioning the limiter by using the injection of lithium pellets has led to a doubling of the energy confinement so that in D-T plasmas:

$$H = \frac{\tau_E}{\tau_E^{89p}} > 4$$

and by peaking the plasma current profile.

\[ \beta_N \geq 3 \]
Advanced Tokamak Studies in TFTR

Reversed shear configurations have also been generated but these tend to disrupt at lower values of $\beta_\text{N}$ ($\sim 2$) with an $n=2$ precursor.
- Operation region on the $\varepsilon \beta_p - \beta_N$ diagram in JT-60U. In, stability limits of the kink-ballooning mode calculated with ERATO are shown for A: a peaked $p(r)$ ($\frac{dp}{d\psi} \sim (1-\psi)^2$) with $q_{surf}/q_0=4$, B: a broad $p(r)$ ($\frac{dp}{d\psi} \sim (1-\psi)0.5+0.7$) with $q_{surf}/q_0=4$, C: a broad $p(r)$ with $q_{surf}/q_0=6$.

- Quasi steady-state ELMy H-mode with $\beta_p \sim 2.5-3$, $\beta_N \sim 2.5-3.1$ and H-factor $\sim 1.8-2.2$ in JT-60U. Fraction of bootstrap and beamdriven currents are $\sim 60\%$ and $48\%$. 
JET PERFORMANCE IMPROVEMENTS

2nd stability core VH-modes

- Negative shear in the core provides 2nd stability
  \[ \Rightarrow \text{increased } \beta_N \]
  \[ \Rightarrow \text{lower diffusivities (Rebut Lallia)} \]
- Edge remains in normal VH regime
- Experimental indications:
  * PEP + H in JET
  * LHEP in Tore Supra
  * rapid elongation + NBI in DIII-D
- Programmed PLHCD, Ip(t), flux expansion
  * Stable route to large negative shear
  * Goal: \( q_0 > 3; q_{\text{min}} > 2 \)
- Toroidal rotation to stabilise kink modes?
  (avoid soaking in)

COHERENT STEADY STATE REGIME (JET)

- Conditions to be achieved simultaneously
  
  i) \( I_{bs}/I \geq 0.7, \beta_p \geq 2 \)
  
  ii) \( H \geq 3 \)
  
  iv) \( \tau_p/\tau_E \geq 10 \)
  
  v) \( \gamma = \langle n \rangle R \text{LCD/PCD} > 0.4 \times 10^{20} \text{ m}^{-2} \text{ A/W} \)

- JT-60 and JET have achieved i), ii) and iii) transiently (\( \beta_N = 1.9 \) for JET)
- Condition iv) is untested
- MHD stability requires profile control
  - Seed current by FWCD or NBCD
  - Bulk current by LCHD, ECCD
  - Methods must be developed
- These regimes must be tested in D/T plasmas not foreseen on TPX and JT-60 Super-Upgrade
High $\beta_p$ H-modes

- Time development of high $\beta_N$/high $\beta_p$ discharge, $I_p = 1.0$ MA, $B_T = 1.7$T, $H = 2.2$, $q_{95} = 5.5$

(a) Pulse No: 33580
(b) Pulse No: 32344
Confinement of high $\beta_p$ plasmas

$H$ versus $\beta_p^{\text{dia}}$ for the high $\beta_p$ campaign

- $H > 2$ achieved over a wide range of $\beta_p$
DESIRABLE FEATURES FOR A MEDIUM-SIZED ADVANCED TOKAMAK

• Non-circular, good shaping flexibility, e.g. high triangularity, with robust feedback control, single and double null X points
• Medium to high field
• Large aspect ratio (\(C-A J^{-1}\))
• Strong additional heating capability preferably localised - \(P(r)\), high magnetic Reynolds number, low collisionality
• Localised current drive capability, preferably on and off axis which may require two methods, with feedback capability
• Ability to drive a large fraction of the current non-inductively
• Good diagnostics for \(p(r)\), \(n(r)\), \(T(r)\), \(q(r)\), \(J(r)\), \(V_\phi(r)\) and for mode activity, e.g. fine resolution e.c.e.
• Long pulse capability i.e \(\tau_{\text{pulse}} \geq \tau_\sigma\)
• Ability to test wall stabilisation and rotation effects
• Heat and particle exhaust compatible with the above features
• Ready access to improved confinement modes e.g. \(P>nBS\), preferably ohmic
• Capable of withstanding serious disruptions/vertical displacement events which may arise in early experiments with mis-aligned current

CONCLUSIONS

Improvement of the tokamak is a crucial issue. Key lines of research:-

• determine the optimum plasma shape and aspect ratio (trade off between physics and technology)
• determine the optimum combination of pressure and current profiles compatible with high \(H\), high \(\beta_N\) and large bootstrap current (solutions lying between the peaked and hollow current profiles have been proposed)
• determine the optimum method of current profile control, probably with feedback
• determine the method of controlling heat and particle exhaust which will be more demanding than that on ITER

The new generation of medium-sized tokamaks have an important role to play.
Role of the plasma rotation in improved confinements in tokamaks

Jinchoon Kim (General Atomics, USA)
ROLE OF PLASMA ROTATION FOR ENHANCED CONFINEMENT IN TOKAMAKS

by

Jinchoon Kim

General Atomics
San Diego, California

Outline of Talk

• Introduction
• Overview of Enhanced Confinement Modes
• Role of Plasma Rotations in Various Cases
• Generation of Plasma Rotations
• Conclusions

Presented at
Symposium for Advanced Tokamaks
Korea Atomic Energy Research Institute
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INTRODUCTION

- It has been recently realized that plasma rotation has a potentially important role in equilibrium, stability, and confinement.

- Stabilization of kink and ballooning modes has been linked to the toroidal rotation, theoretically and experimentally.

- Well-known confinement enhancements, H-Mode and VH-mode, are achieved more or less by virtue of plasma rotational shear, poloidal and toroidal.

- Present large tokamaks employ neutral beam injection heating which supplies not only energy but also momentum which provides the necessary plasma rotation.

- For the future tokamaks where RF is the primary heater, the lack of toroidal rotation is of serious concern.

OVERVIEW OF ENHANCED ENERGY CONFINEMENT MODES IN TOKAMAKS

- **H-Mode**
  - First discovered in ASDEX (Wagner PRL 82)
  - Formation of transport barrier in the edge
  - $\tau_E$ increased by a factor of 2 over L-mode
  - Observed in most magnetic confinement devices
  - Strong candidate scenario for ITER

- **VH-Mode**
  - First discovered in DIII-D (Jackson PF 92)
  - Transport barrier formed further inside plasma
  - $\tau_E$ increased by a factor of 2 over H-mode

- **Hi $\beta_P$ H-Mode**
  - Somewhat better confinement than H-mode
  - High fraction of bootstrap current (JT-60U, JET)

- **Resistive Wall Stabilization**
  - Shown experimentally and theoretically that toroidal rotation is required to achieve this.
  - Beta-limit can exceed the Troyon limit showing a good motivation for further investigation for reactor application.
CHARACTERISTICS OF H-MODE

- Sudden change of plasma properties occurs rapidly (~100 μsec) with heating power exceeding a certain threshold power (L-mode to H-mode transition).

- Experimentally noticed by
  - Increase of density, temperature, and stored energy
  - Decrease of edge neutral density (Dα signal)
  - Decrease of density fluctuation and magnetic fluctuation
  - Increase of plasma rotations
  - Increase of the edge Er shear (\( \frac{dE_r}{dr} \))

- H-mode is also accompanied by the presence of so called edge localized mode (ELM) which dumps out the plasma particles across the field, limiting the enhancement but providing a means of curtailing the density runaway.

- H-mode is perhaps the most robust and ubiquitous enhanced confinement phenomenon ever found in magnetic confinement devices, and yet its physics is not fully understood.
UNDERSTANDING OF H-MODE

• Discovery of impurity ion poloidal rotation spinup in electron diamagnetic drift direction ($V_\theta$) at L-H transition triggered tremendous theoretical activities (Groebner, PRL 90). The resulting $-V_\theta B_\phi$ yields a negative $E_r$ well in the plasma edge.

• Fluctuations (instabilities) are suppressed by the sheared flow velocity $V_E = E \times B / B_\bot B$ (Shaing PRL 89, Biglary PRL 90).

• However, it was discovered that the main ion poloidal rotation is in ion diamagnetic drift direction and the edge $E_r$ well is maintained by pressure gradient ($\nabla P$) rather than poloidal rotation, redirecting the theoretical thinking (Kim, PRL 1994).

• Subsequently, improved theories emerged to include $\nabla P$: transport bifurcation theory (Hinton and Stabler, PF 93), addition of $\nabla P$ to BDT theory (Carreras PPCF 94), extension of BDT theory including toroidicity, radial shear of $V_\phi$ and magnetic shear (Hahm, PP 94).

• The radial electric field is deduced from the measured quantities ($N_i$, $T_i$, $V_\theta$, $V_\phi$) based on the radial force balance equation. Which term is responsible for triggering L-H transition has yet to be resolved (causality).

$$E_r = \frac{1}{n_i Z_i e} \frac{dP_i}{dr} - V_{\phi i} B_\phi + V_{\phi i} B_\theta$$

• Some other critical issues for future devices includes the effect of wall recycling, properties of ELMs, and scaling of the H-mode threshold power, and at present they are not well understood.
A transport barrier forms in the shear layer of the radial electric field as indicated by the increasing gradient of the ion temperature.

The shear in the radial electric field at the plasma edge increases substantially at the L-H transition.
Radial electric field profile exhibits a sudden change upon L-H transition into negative $E_r$ and $E_r'$. This well shape formation is a typical signature of L-H transition in DIII-D and other tokamaks. Here the $E_r$ is deduced from the main ion rotation and pressure gradient measurement using He II line for CER spectroscopy in helium plasma.

**CHARACTERISTICS OF VH-MODE**

- $\tau_E$ improvement owing to
  - Penetration of $E_r \times B$ velocity shear further inside; $V_\phi$-shear further inside is responsible for $E_r'$
  - Edge ballooning mode stability

- Wall conditioning is important:
  - Boronization by means of glow discharge in 10% B$_2$D$_6$-90% Helium gas
  - Full carbon low recycling wall with He glow discharge

- A large second stable region at the plasma boundary may be important in ELM suppression which is necessary condition for achieving good VH-mode confinement.

- The VH-mode confinement quality was shown to be deteriorated as the toroidal rotation shear is reduced by means of a "magnetic braking of toroidal rotation" using an external $n=1$ error field coil.
VH-MODE THERMAL COEFFICIENT IS 1.5-2 TIMES HIGHER THAN PRE-BORONIZATION H-MODE CONFINEMENT.

\[ \tau_H = \frac{\tau_{th}}{\tau_{D3-JET}} \]

ELMS BLOCK PENETRATION OF ExB VELOCITY SHEAR AND REDUCE PEAK CONFINEMENT \((\kappa=2, \delta=0.9 \text{ DOUBLE NULL})\)

- \(D_\alpha-\text{DIV}\)
- \(T_e/T_{\text{JET-D3D}}\)
- \(\beta_+(l/eB)\)
- \(V_\phi(\rho \approx 0.6)\) (km/s)
- \(V_\phi(\rho \approx 0.8)\) (km/s)

\[ \tau_{D3-JET} = 0.106P_{\text{th}}^{-0.46}I_\text{p}^{1.03}R^{1.48} \]

\(P_L = P_{\text{th}} - \psi \)
MAGNETIC BRAKING OF PLASMA ROTATION

- Plasma toroidal rotation is reduced by an induced drag force provided by the application of a static non-axisymmetric resonant magnetic field perturbation of mostly low poloidal and toroidal mode numbers from an external coil (n1 coil) [1]

**RESISTIVE WALL STABILIZATION**

- It was an old notion that only an infinitely conducting wall can affect stabilization while a resistive wall merely slows down the growth rate.

- Wall stabilization of the external kink mode by rotating plasma has been theoretically predicted. *(Bondeson & Ward PRL 94)*

- Wall stabilization experiment and analysis have been carried out in DIII-D, which showed a beta limit improvement of 30% over the value predicted by the ideal calculation with no wall stabilization in the DIII-D tokamak. *(Tumbull, IAEA CN 94)*

- Toroidal rotation is an essential condition for wall stabilization to occur; Growth of $m, n = 1$ modes was seen to be associated with plasma rotation slowdown ($\Omega/2\pi < 1$ kHz) in the region between $q=2$ and $q=3$. *(Taylor, PP 95)*

**POSSIBLE CAUSES OF LOSS OF ROTATION**

- Saturated tearing modes
- ELMs
- Fast ion losses from TAEs and fishbones

![Graphs showing rotation frequency, $\delta B_{\theta}$, and fast ion loss over time](image)
COLLAPSE AT HIGH $\beta_N$ SHOWS IMPORTANCE OF ROTATION

- Slowly rotating $m/n = 3/1$ mode grows after plasma rotation has slowed down

- $3/1$ mode has features expected of a 'resistive wall' mode:
  - Slowly growing, slowly rotating (25 Hz) from onset: $\gamma \simeq \omega \simeq 0.3 \tau_{\text{tor}}^{-1}$

<table>
<thead>
<tr>
<th>Toroidal Rotation Frequency (kHz)</th>
<th>CER</th>
<th>Mirnov</th>
</tr>
</thead>
<tbody>
<tr>
<td>$q=2$</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$q=2.5$</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$q=3$</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

- Soft x-ray emission center

- $\delta B_0$ (T) vs. time (ms)

<table>
<thead>
<tr>
<th>$\delta B_0$ (T)</th>
<th>$\delta B_r$ (T)</th>
<th>$m/n = 3/1$</th>
<th>$2/1$</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.01</td>
<td>0.00</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>0.00</td>
<td>0.00</td>
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<td>0.00</td>
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<td>0.00</td>
<td>0.00</td>
<td>0.00</td>
<td>0.00</td>
</tr>
</tbody>
</table>

HIGH $\beta_p$ H-MODE

- JT-60U Results (Koide PPCF 94)
  - High $\beta_p$ discharge with $\nabla B$ drift toward X-point, balanced NBI. ($\beta_p = \frac{2 u < P >}{\beta_p^2} - 1.2$)
  - Transport barrier and toroidal rotation shear is formed well inside with enhancement factor of H-3.
  - $I_{\text{BS}} \sim 0.58 \text{ Ip}; \text{Peaked } T; E_r \text{ positive}; E_r' < 0$

- JET Results (Stork PPCF 94)
  - Low central particle fueling and flat density profile with strong gas fueling are the condition for this mode with ICRH only.
  - Confinement was 70 % better than the D3D-JET H-mode scaling. The second stability regime was accessed in the edge.
  - $I_{\text{BS}} \sim 0.7 \text{ Ip}; E_r \text{ positive}; E_r' < 0$
SOURCES OF PLASMA ROTATION

- Neutral Beam Injection (NBI)
  - Current large tokamaks employ NBI with the aiming direction generally into the direction of plasma current (co-injection).
  - Tangentially directed NBI provides toroidal momentum and additional plasma current.
  - Balanced NBI (i.e., no net toroidal momentum) is also used for supershot mode in TFTR or high-\(\beta_p\) H-mode in JT-60U.
  - An idea of using counter-NBI for \(J_r\) profile control to produce JxB rotational torque has been tested in DIII-D (Kim APS Bull 94).
- Electrode-driven Flow (Taylor PRL 1990)
  - Induce radial current by a biased electrode in the plasma edge to generate JxB torque.
  - H-mode was actually obtained (CCT, TEXTOR) by this technique, but impractical for larger devices.
- Rotating Magnetic Islands
  - Edge Islands driven by applying oscillating current to external coils (Jensen PF 91).
- RF-driven Flow
  - Alfven-wave to induce plasma flow by radial shear of Reynolds and magnetic stresses (Craddock PRL 91).
  - Ion Bernstein Wave (Experiment in PBX-M).

CONCLUSIONS

- For the enhanced confinement modes discussed here, the energy confinement is improved (\(H \sim 1.5 - 4\)) owing to fluctuation suppression by ExB flow velocity shear and the resulting formation of transport barrier.
- The radial electric field is created through plasma rotation and pressure gradient. The rotation is caused by particle momentum input or by \((J_r \times B)\) torque.
- Poloidal rotation spin-up is associated with L-H transition.
- For VH-mode, toroidal rotation spin-up is a necessary condition; \(VH \rightarrow H\) was demonstrated by “magnetic braking” of the toroidal rotation.
- For high-\(\beta_p\) H-mode, toroidal rotation shows up even if the net toroidal momentum input is zero.
- Wall stabilization of MHD modes is possible and higher beta limit can be achieved only if large enough toroidal rotation is present.
- Theory of rotation shear-induced fluctuation decorrelation in toroidal geometry brings out the importance of the toroidal rotation in general.
ECH and ECCD at T-10 tokamak and at ITER

V. Strelkov (RSRC “Kurchatov”, Russia)
빈 면
ECR as auxiliary power system on ITER and recent results in ECRH and ECCD on the T-10 Tokamak.

V. S. Strelkov, Nuclear Fusion Institute, RSRC "Kurchatov Institute", Moscow, Russia.

- Auxiliary power systems on ITER. — Why ECR?
- History of T-10 ECR heating and current drive experiments
- Recent results in ECR on T-10 ("second harmonic")
  - stabilization of \( m = 2 \)
  - current drive efficiency
  - avoidance of locked modes.
EXECUTIVE SUMMARY
of ITER workshop and technical meeting on heating and current drive (3-7 October 1994)

Auxiliary power systems on ITER must perform four functions:

1. **HEATING**, which further subdivides into:
   1.1 Providing sufficient power across the separatrix to access H-mode confinement (100 MW)
   1.2 Increasing the temperature to ignition (50 MW suffices)
   1.3 Supporting driven burn scenarios if confinement proves to be inadequate (100 MW)
   1.4 Maintaining adequate temperature during the current termination phase when the density exceeds the Greenwald limit.

2. Maintaining sufficient plasma rotation to avoid locked modes and to stabilize ideal and resistive kink instabilities via a conducting wall.

3. **Driving on-axis current** (~0.5 MA) to provide the seed current for high-bootstrap-current steady-state tokamak discharges.

4. **Driving off-axis current** (~2.5 MA) to maintain a reverse-shear profile needed to access high $\beta_N$-values in steady state tokamak discharges.

---

Table 2. Summary of Present ITER Heating and Current Drive R&D

<table>
<thead>
<tr>
<th>Method</th>
<th>Deliverables</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>ECH</td>
<td>1996,5 end of EDA</td>
<td>(+) Minimum impact on tokamak access</td>
</tr>
<tr>
<td></td>
<td>CW window for single tube 1MW, 170GHz, CW transmission guides</td>
<td>&quot;short pulse&quot; means several seconds limited by window heating</td>
</tr>
<tr>
<td></td>
<td>1MW, 170 GHz, short pulse mirror for launch angle</td>
<td>Potentially complicated tritium/vacuum boundary</td>
</tr>
<tr>
<td></td>
<td>1MW, 140GHz, CW tritium boundary windows</td>
<td>(+) Permits easily varied plasma configurations</td>
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<td>Potentially complicated tritium/ vacuum boundary</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(+) Permits easily varied plasma configurations</td>
</tr>
<tr>
<td>ICRF</td>
<td>NBI</td>
<td></td>
</tr>
<tr>
<td>------</td>
<td>------</td>
<td></td>
</tr>
<tr>
<td><strong>ICRF antenna design for heating and current drive</strong>&lt;br&gt;antenna prototype (test on a tokamak remain to be arranged)&lt;br&gt;RF generator system capable of dealing with ELM loading variations&lt;br&gt;(-) Needs to select between in-port and in-blanket designs&lt;br&gt;Tokamak tests with ELMs essential to qualify system&lt;br&gt;(-) Sensitive to plasma boundary, which must be near antenna&lt;br&gt;(+) Retains simple tritium/vacuum boundary</td>
<td><strong>NBI</strong>&lt;br&gt;0.5MeV, 10MW, JT-60U negative ion beam injector unit&lt;br&gt;initial operation of 1A, 1MeV, H accelerator and 1MeV, 0.1A, D accelerator&lt;br&gt;Reliability data on JT-60U 0.5MeV system&lt;br&gt;initial results from 0.25MeV LHD system&lt;br&gt;reliability data from test accelerators&lt;br&gt;insulator designs&lt;br&gt;1MeV power supply design&lt;br&gt;(+) Only method which imports angular momentum; preferred counter rotation not consistent with current drive&lt;br&gt;Requirements may be need to be revised to optimize driving plasma rotation&lt;br&gt;(-) Tritium boundary extended to NBI Unit</td>
<td></td>
</tr>
</tbody>
</table>
Table 1. System Function Capabilities
Successful technology

<table>
<thead>
<tr>
<th>Function/ System</th>
<th>Fast Wave</th>
<th>NBI</th>
<th>ECH</th>
<th>LHH</th>
</tr>
</thead>
<tbody>
<tr>
<td>Heating, H-mode access</td>
<td>OK at ω = 20Γ</td>
<td>OK, needs OK</td>
<td>E≥0.5 MeV</td>
<td></td>
</tr>
<tr>
<td>Rotation Maintenance</td>
<td>unlikely</td>
<td>Proven, requires</td>
<td>unlikely</td>
<td>E≥0.5 MeV</td>
</tr>
<tr>
<td>On-axis current drive</td>
<td>Predicted adequate with ν = 70 MHz, but physics demonstration needed</td>
<td>Proven, requires OK, but not proven?</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Off-axis current drive</td>
<td>No developed scenario because of high degree of spatial control</td>
<td>Predicted OK, but not proven</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Our conclusion: ECR is optimal!

T-10 - the oldest working tokamak in the world. The first ohmic heating plasma: July 1975. R = 1.5 m; a = 0.38 m; B = 4.5 T; Jp = 0.65 MA. ECH and ECCD experiments since 1979.

The HF power is injected on the first or second harmonic through 4 horizontal ports from the low-field side.

Excellent heating of electrons was shown. 80-90% of injected HF power are adsorbed by plasma core. The measurements of the distribution of HF power demonstrate a narrow peak in the adsorbed power distribution near the resonance zone.

(a) The variation of the central electron temperature (measured by electron cyclotron emission at the second harmonic) in a shot with high heating power P = 2 MW and averaged plasma density \( n_e = 1.5 \times 10^{19} \text{ cm}^{-3} \); (b) the spectrum of x-ray emission at the stage of high electron temperature in the same shot.
The efficiency of heating is weakly dependent on the ECR position for resonance inside the surface \( q = 2 \). The vertical dashed line corresponds to the \( q = 2 \) surface position.

**FIG. 1.** Experimental arrangement for ECCD experiments on T-10.
The second harmonic

Experimental arrangement for ECCD experiments on T-10.

\[ \lambda = 2.14 \text{ mm} \]
\[ f = 140 \text{ GHz} \]
\[ P_{MF} \leq 0.9 \text{ MW} \]
\[ \varphi = 21^\circ \]

\[ \lambda = 1.9 \text{ mm} \]
\[ f = 157 \text{ GHz} \]

**Co- and Counter-Method of estimation of $I_{CD}$**

\[ (V_0)_{th} = \frac{(V_0)_{exp} + (V_0)_{co}}{2} \]

\[ 2 \Delta V = (V_0)_{exp} - (V_0)_{co} \]

The plasma current is kept constant by the feedback system

\[ I_{CD} = I_p \frac{\Delta V}{(V_0)_{th}} \]

\[ I_{Co} \approx I_p \]

Co and \textit{Co+} ohmic discharges should be equal!
The experimental efficiency of the first harmonic current drive is in agreement with the theory.
ECRH STABILIZATION OF \( m = 2 \) MHD ACTIVITY

In low-\( q_b \) regimes, stabilization of the MHD activity has the "resonance" type. Only the direct heating of the \( m=2 \) island leads to considerable suppression of the \( m = 2 \) mode.

The time variation of the dependence of inverse value of the m=2 mode instant frequency \(360/(d(\text{PHASE})/dt)\) on the poloidal phase of the mode.

The spikes in the picture correspond to the mode-locking events. A strong mode-locking suppression takes place under ECRH.

Unintegrated signals of the MHD-coils were used.
Lock Mode without ECRH
Superconducting tokamak HT-7 project at IPP–Hefei

J. Xie (IPP–Hefei, China)
1. INTRODUCTION

Advanced tokamak operation mode is a way to realize the effective and compatible fusion reactor. Some projects are concentrating to improved \( \beta, T_e, f_{bs} \) and plasma wall interaction in steady state operation. The research in ASIPP paid great attention to that in parallel to the effort of ITER. Experiments will be conducted on the newly constructed HT-7 superconducting tokamak and the planning HT-7U project. Those devices are characterized of long pulse or continuous plasma, heating and current drive by rf waves with the ability of profile control. Different methods for controlling plasma wall interaction will be tested.

2. HT-7 SUPERCONDUCTING TOKAMAK [Fig. 1]

2.1 Main parameters

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>( R_a ) (m)</td>
<td>1.22</td>
</tr>
<tr>
<td>( a ) (m)</td>
<td>0.30</td>
</tr>
<tr>
<td>( B_t ) (Tesla)</td>
<td>2.5-3</td>
</tr>
<tr>
<td>( I_p ) (MA)</td>
<td>0.4-0.5</td>
</tr>
<tr>
<td>( \Phi ) (Vs)</td>
<td>1.7</td>
</tr>
<tr>
<td>( V_b ) (Volt)</td>
<td>30</td>
</tr>
<tr>
<td>( T_{dis} ) (Sec)</td>
<td>3-60</td>
</tr>
<tr>
<td>( P_{ICRH} ) (MW)</td>
<td>2.5</td>
</tr>
<tr>
<td>( P_{LHCD} ) (MW)</td>
<td>2</td>
</tr>
<tr>
<td>( P_{ECRH} ) (MW)</td>
<td>0.4</td>
</tr>
<tr>
<td>( N_e ) (m(^{-3}))</td>
<td>5*10(^{19})</td>
</tr>
</tbody>
</table>

2.2 Operation regime

* full power Ohmic discharge: \( I_p = 500 \text{ kA} \), with \( q \) close to 2
* LHCD: \( I_{rf} = 200-300 \text{ kA} \), \( N_e = 2*10^{19} \text{ m}^{-3} \)
* ICRH: $T_i = 2-3$ KeV
* IBCD: $I_{cd} = 100$ kA
* AC Operation: $I_p = 100$ kA
* LHCD + ICRH + ECRH
* Plasma edge control: Pump limiter, radiation layer, gas inlet
* High $\beta$ operation
* Improved confinement regime in long pulse operation
* Test of material or coating on first wall
* Working gas $\text{H}_2, \text{D}_2, \text{He}$

3. PHYSICS OBJECTIVES

3.1 LHCD

The main parameters of LH power source $P_{rf} = 2.4$ MW, $f = 2.45$ GHz, pulse length = 3-60 sec, two grills, each of $2*12$ subwaveguids with adjustable phaseshift [Fig. 2]. That is convenient to change the launch spectrum.

A code simulation showed that the rf power can penetrate to the plasma center with line averaged density of $2*10^{19} M^{-3}$, $B_t = 2$ tesla, $\eta_{cd} = (0.05-0.20)*10^{20} M^{-2} A/W$. Experiment of transformer recharging is prepared in case of high $\eta_{cd}$ achieved.

3.2 ICRF heating and current drive

Hydrogen minority heating is taken as the main candidate regime and Deuterium second harmonic heating as an alternative. The rf system parameters are:

$$P_{rf} = 2-3 \text{ MW}, \quad f = 15-40 \text{ MHz},$$

pulse length = 1 sec (full power), CW (0.6 MW power)

four half loop antenna with phasing, $T_i - 2-3$ KeV (H-mode scaling), $\tau_E - 20$ ms.

FWCD and IBCD will also be tested. The antennas are under design. Experimental database would be useful for HT-7U design.

Synergetic effect of IC and LH in current drive will be investigated. Because of the difficulty of the different operation conditions for the two waves, various combination regimes must be prepared by numerical simulation and experiment test.

3.3 Energetic tail of electrons and ions

In reactors the high energy ions will contribute to increase of fusion products by factor of 2-5, that is predicted by computational simulation and verified by experimental evidences. High energy electrons play an important
role on Macro-and Micro-instabilities. HT-7 experiment will pay great attention to this subject, by investigating the generation, velocity distribution and profile, thermaization, confinement, plasma-wall interaction of high energy particles, instabilities and confinement of background plasma.

3.4 Plasma-wall interaction

Under LHCD, ICRH and ECRH, a great amount of high energy particles are produced and impact the first wall. The recycling is an ambiguous issue. The following means are chosen to control particle influx:

* hot or cold wall operation
* coating of first wall: B, C or Si
* scrape off layer control by ergodic limiter and gas puffing
* corresponding discharge condition
* movable limiter and pumping limiter
* electrical field and sheared flow induced by electrode or rf power

3.5 AC operation

The power supply of poloidal system has been designed for AC operation by transformer induced current. ECRH will be benefit for current direction reversal. The impact of varied transverse field act on superconductor should be considered and experimentally tested.

3.6 Increasing $\beta_p$ and bootstrap current

An optimistic estimation shows $\beta_p$ could be 2.3 - 3 when $q_s = 4$ according to Troyon limit.

The bootstrap current may achieved 1/4 - 1/3 of $I_p$. The circular cross section is unfavorable to optimization of bootstrap current. As current drive and high energy density are adopted, bootstrap current has to be taken into account.

4. DIAGNOSTICS [Fig. 3]

* Electro-magnetic measurement
  for $I_p$, $V_t$, displacement, magnetic configuration, MHD oscillations
diamagnetic flux
* TV camera
  for plasma image in visible light
* Infra-red TV camera
  for limiter temperature
* Electrical probes
  for scrape off layer parameters such as $n_{cs}$, $T_{cs}$, and their fluctuations,
  floating potential and particle flux

5 - 5
5. HT-7 OPERATION

Through July 1994—March 1995, engineering test and tokamak discharges have been conducted. The superconducting toroidal field achieved 2.4 tesla. The first wall was conditioned by baking, glow discharge and Tayler discharges. The base pressure achieved $5 \times 10^{-6}$ Pa and the residual gas are dominated by working gas with a very small portion of CO and vapor. [Fig. 4]

The discharge parameter are shown in Figure 5. $I_p$ up to 15 kA with about 150 ms flat top. The loop voltage is about 2-3 volt. The spectroscopic measurement [Fig. 6], shows the main impurities. We can also estimate the $T_e \geq 800$ eV. The adjustment only last about 15 day, that means HT-7
subsystems are in good condition. Next campaign will do ohmic discharge experiment.

6. HT-7U PROJECT

6.1 Objectives
HT-7U has ambitions physics objectives comparing to the existing tokamaks in ASIPP.
* A big elongated superconducting tokamak, with pulse length from 1 minute to steady state operation.
* Advanced tokamak operation with of power additional heating only in long pulse.
* Fuel recycling and phasing plasma components test.

6.2 Parameters [Fig.7]

\[
\begin{align*}
R &= 1.5 \text{ M} \\
b/a &= 0.7/0.35 \text{ M} \\
K &= 2 \\
B_t &= 4-4.5 \text{ Tesla} \\
I_p &= 0.7-1 \text{ MA}
\end{align*}
\]

6.3 Schedule and budget

<table>
<thead>
<tr>
<th>Activity</th>
<th>Year</th>
</tr>
</thead>
<tbody>
<tr>
<td>design and R &amp; D</td>
<td>1994—1996</td>
</tr>
<tr>
<td>construction</td>
<td>1996—2000</td>
</tr>
<tr>
<td>operation</td>
<td>2001—2010</td>
</tr>
</tbody>
</table>

budget 120 Million Chinese ¥ + existing facilite
Figure Captions

Fig. 1  HT-7 Tokamak
Fig. 2  HT-7 LHCD System
Fig. 3  HT-7 Diagnostics
Fig. 4a Mass-spectrometry before conditioning
Fig. 4b Mass-spectrometry after conditioning
Fig. 5  Discharge Waveforms \( I_p, V_e, \Delta \phi \)
Fig. 6a Monochrometer Curves
Fig. 6b Spectroscopy
Fig. 7  HT-7U Tokamak
Reference onase

2.45 GHz Oscillator
Isolator
Band Pass Filter
Directional Coupler
2 Way Power Divider
Isolators
PIN Diode Modulators

To Antenna. 1.

To Antenna. 2.

Attenuator
Phase shifter
Driver Amplifier
Isolator
Directional Coupler
High Power Amplifier
Klystron (100KW)
ARC-Detector

Circulator + Dummy HP-Load
~ 10–30m rectangular Waveguide
DC-Break

Bi-directional Coupler
RF-Window
ARC-detector
3 Way Power Divider

Fig. 2  HT-7 LHCD System
Fig. 3 HT-7 Tokamak Diagnostics
Fig. 4a  Mass-spectrometry before conditioning
Fig. 4b  Mass-spectrometry after conditioning
Fig. 5  Discharge Waveforms $I_p, V_e, \phi$
Fig. 6a  Monochrometer Curves
Cr I (4254.4 Å, 4274.8 Å, 4289.7 Å)

Fig. 6b  Spectrum
Cr I (4254.4 Å, 474.8 Å, 4289.1 Å)

Fig 6b  Spectrum
The Primarily Design of The HT-7U(Upgrade) (ASIPP)
Recent experimental research on JFT-2M

M. Mori (JAERI, Japan)
Recent Experimental Research on JFT-2M

M. Mori and the JFT-2M team

Naka Fusion Research Establishment, Japan Atomic Energy Research Institute, Naka-machi, Naka-gun, Ibaraki-ken, 311-01, Japan

Abstract

Recent experimental efforts on JFT-2M have been devoted to investigations on particle and heat control with boundary plasma modification and challenges to disruption control. We have succeeded in demonstrating an active density control during the H-mode with helical magnetic perturbation with n=4. The poloidal mode number, m, of the helical perturbation must be chosen so that the resonant surface is close to the separatrix; m/n ~ q_s, where q_s is a surface safety factor. It has been found that scrape-off-layer current induced by the divertor biasing can modify in-out asymmetry in particle flux from 0.4 to 1.8 and that in heat flux from 0.25 to 0.37. It has been found that both of local ECH and helical magnetic field can be effective to stabilize tearing modes and to avoid disruptions.

We will begin in 1995 intensive study on the particle and heat flux control with new closed divertor with pumping, by replacing the old open divertor. The purpose of this program is understanding divertor plasma physics and making database for an optimized divertor geometry with a dense and cold divertor plasma.

[1] Introduction

Achieving ignition condition and sustaining it in a steady state are necessary for thermonuclear fusion reactor. Improved confinement mode is necessary to achieve the ignition. For instance, confinement enhancement factor, H-factor defined by the confinement time normalized by ITER-89 L-mode power law, must be larger than 2 in the recent ITER design. We, researchers on the nuclear fusion, have succeeded in finding and developing some improved confinement modes: H-mode, VH-mode, high-\(\beta_p\) mode, high-\(\beta_p\) H-mode, super shot, and so on. However, these improved confinements are achieved only transiently, so far. Our present goal is to demonstrate sustaining the improved confinement modes in steady state. Toward the goal, we have to develop a non-inductive current drive technique feasible in high density, high temperature and ignited plasma. It is also necessary to develop techniques for particle and heat control (particle fueling, particle exhaust, heat exhaust) being compatible with the improved confinement mode. According these motivations, recent experimental efforts on JFT-2M have been devoted to the following investigations and challenge to contribute to ITER Physics R&D activity and the JT-60U project, making the best of flexibility coming from medium size of JFT-2M, in which non-circular plasmas with divertor configuration can be produced (R=1.31 m, a=0.35 m, \(\kappa=1.1.4\), \(B_{t\text{max}}=1.4\) T, \(I_{p\text{max}}=300\) kA in divertor configuration).
(a) particle and heat control with boundary plasma modification
(b) disruption control
(c) non-inductive current drive with fast wave (FWCD)
(d) contributing to ITER H-mode database activity

Recent experimental results on (a) and (b), and a plan in 1995 will be described in this paper.

[2] Improvement with Boundary Plasma Modification

2.1 Steady H-mode with helical magnetic perturbation

Plasma density and radiation loss increase continuously in an ELM (Edge Localized Mode) -free H-mode, even without gas puffing. The density rise has to be suppressed to realize steady H-mode operations. We have tried an active density control during the H-mode with helical magnetic perturbation produced by three types of perturbation coil systems, those are shown schematically in Fig. 1. The time evolution of the H-mode with the perturbation field (produced by EML-coil and Ladder-coil) is shown in Fig. 2. The plasma in a single null divertor configuration was heated by tangential neutral beam injection of 1 MW from 600 ms to 800 ms. The helical perturbation was applied from 700 ms, with a rise time of 50 ms. Many ELMs, indicated by Hα bursts, are induced in the flat-top of the perturbation. The increases in density and the radiation loss are suppressed by this edge phenomenon. The steady H-mode is achieved with the help of these ELMs induced by the helical magnetic perturbation.

Fig. 1 Helical Perturbation Coil Systems Fig. 2 Steady H-mode with helical magnetic perturbation (EML + Ladder)

The repetition rate of ELMs was an increasing function of the perturbation field. The perturbation field consists of many helical field components, each of which is characterized by toroidal mode number, $n$, and poloidal mode number, $m$. We have examine correlation between the repetition rate and each helical field component using experimental data obtained with various combinations of the perturbation coils. There was no positive correlation between the repetition rate and the low $n$ helical field ($n=1, 2$ or $3$). However, we have found that the repetition rate is a monotonically increasing function of helical field component with $m=12~20$. 
and n=4. It means that the ELMs are induced effectively in JFT-2M by the (m=12−20, n=4) helical field, that can make resonant island of m/n=3~5 close to the separatrix[1].

This conclusion is consistent with the experimental q_s dependence of ELM excitation by the helical perturbation, as shown in Fig.3, where the value of q_s is changed by B_t. ELM can not be seen in H_α emission with q_s=5, but is induced with q_s=3 and q_s=2.5. Estimated resonant helical field with n=4 on each magnetic surface is plotted as a function of radial position in each discharge in Fig.3. The resonant field near the separatrix in the q_s=3 case or the q_s=2.5 case is five times as strong as in the q_s=5 case.

![Fig. 3 q_s dependence of ELM activity](image)

### 2.2 Divertor Biasing

It is still an open question whether an external heating power of 50 MW in the present ITER design is enough to achieve the H-mode or not. This is a reason why establishing a scaling for the H-mode threshold power is one of urgent tasks in physics R&D on ITER-EDA. In such present situation, developing a way to reduce the threshold power is an important research issue. We have demonstrated successfully in JFT-2M that the threshold power can be reduced significantly by negative radial electric field introduced in SOL (scrape-off-layer) plasma with divertor biasing[2, 3]. Electric potential of all divertor plates installed at the bottom of the vacuum vessel were biased in negative against the grounded vessel, as shown in Fig. 4(a). The negative radial electric field is produced in an extremely narrow region about 2 cm outside the separatrix. The threshold power was reduced from 350 kW to 150 kW by the negative biasing of -80V, as shown in Fig. 4(b). The neutral particle influx into the main plasma was reduced with the negative biasing as a result of a drop in the radial particle diffusion in SOL. The reduction of the neutral particle seems to be a reason for a lowering of the threshold power.
Fig. 4 Reduction of the H-mode threshold power with Er induced by divertor biasing

It is possible to induce electric current along magnetic field in SOL, when biasing voltage is applied between inside and outside plates. We have found that in-out asymmetry of particle and heat fluxes flowing into the divertor plates is affected by the SOL current of 1.5 kA. Since the asymmetry existing with no biasing inevitably causes high heat load on one side of the divertor plates, the control of the asymmetry is very important for lowering the heat load. Figure 5 shows the two biasing schemes: (i) a positive biasing where the current flows from inside to outside and (ii) a negative biasing for reversing current. The profile of electron density and electron temperature on the divertor plates was measured in various conditions: positive biasing (+110 V, 310 A), negative biasing (-130 V, -210 A), and no biasing. The density at the ion side divertor (inside) and the temperature at the electron side (outside) changed significantly depending on the SOL current. As shown in Fig. 6, The input power of only 20-40 kW supplied by the biasing modified the in-out asymmetry in the particle flux from 0.4 to 1.8 and that in the heat flux from 0.25 to 0.37, where a net heating power was 350 kW.

Fig. 5 SOL current induced by divertor biasing

(i) positive bias $E_r$ and $j$ are induced in SOL.
(ii) negative bias no $E_r$ in SOL

$j$ is induced in SOL.

Fig. 6 In-out asymmetry of particle flux and heat flux

[3] Disruption Control

3.1 Disruption avoidance by local ECH

Many types of disruptions seem to be caused by tearing mode with helical island structure of magnetic surface, namely magnetic island. Since the magnetic island formation is originated by poloidal asymmetry of the plasma current density on the rational magnetic surface, the stability can be much affected by local modulation of poloidal current density
profile[4, 5, 6]. We have applied local ECH to (m=2, n=1) tearing mode to modify the poloidal profile of current density in JFT-2M[7, 8] as shown in Fig. 7. Electromagnetic wave with the frequency of 60 GHz was launched in the second harmonic extraordinary mode with the injection angle of 80 degrees (nearly perpendicularly) to the magnetic axis. Since plasma current is sustained by toroidal electric field, the O-point local heating in the magnetic island results in the reduction of resistivity and then increase of the current density at the O-point. This modification reduces the poloidal asymmetry of the current profile, and could stabilize the tearing mode.

![Diagram of ECH antenna](image)

**Fig.7 Local ECH to stabilize tearing mode**

A tearing mode with (m=2, n=1) structure often grows in limiter discharges when the surface safety factor, $q_s$, stays around 3 as shown in the time behavior of $dB/dt$ in Fig. 8, where $dB/dt$ is fluctuating poloidal magnetic field measured by a pick-up coil. The amplitude of the magnetic fluctuations obtained by integration of $dB/dt$ continuously grows with the decreasing frequency from 3-5 kHz toward 0 Hz until the plasma disruption occurs. The electron temperature ($T_e$) at minor radius of 0.7a decreases with large fluctuations by the appearance of the large m=2 magnetic island. The temperature at the X-point of the island corresponds to the peak temperature of the fluctuation, and the O-point temperature the bottom. The island width inferred from the $T_e$ fluctuations (measured by electron cyclotron emission) and the $T_e$-profile (measured by TVTS) (see Fig. 9) reaches to 0.2a-0.3a.

We have applied local ECH to stabilize the tearing mode. The mode activity has been reduced and sometimes completely suppressed as shown in Fig. 10 when the wave power is deposited in a narrow region of r/a=0.68-0.72; the $T_e$ fluctuations and the magnetic fluctuations have been reduced and the disruption has been avoided successfully. The minimum power to avoid the disruption was 20% of the ohmic input power. The heating position where the ECH is effective for the mode suppression corresponds to the location of the q=2 magnetic surface; the disruption could not be avoided (up to the injected ECH power of 0.14 MW) when the...
heating point was either inside or outside of the q=2 magnetic surface. The width of the effective heating region is extremely narrow compared with the island width.

![Fig. 9 Electron temperature profile with a large m=2 magnetic island (measured by TVTS)](image)

![Fig. 10 Dependence of tearing mode stabilization on radial heating position](image)

In order to confirm that the O-point heating is a key to reduce the island width, we have applied the ECH power modulated by the phase-shifted magnetic pick-up signal as shown in Fig. 11 [9]. This method is equivalent to changing the power deposition poloidally in the island, because the m=2 island rotates poloidally. The width of each ECH pulse (~90kW at the peak of the pulse) was about 0.1 ms. Experimental result shows clearly that heating the O-point is effective for the suppression of the mode, but heating the X-point is not (Fig. 11). The recent computational simulation with the reduced set of the nonlinear resistive MHD equations [10] is consistent with these experimental observations in the O-point heating and the X-point heating.

![Fig. 11 Comparison between O-point heating and X-point heating](image)

We have tried to control the density limit disruption with local ECH. It was found that the m=2 mode activity can be reduced when the heating point is located in the magnetic island. As a result, the density limit was improved by about 10%. A loss of island heating due to appearance of the wave cut-off seemed to limit the mode stabilization, where the cut-off density of the ECH wave is 2.1x10^19 m^-3. Since operations with the toroidal field of 2.1 T will be
available in 1995, disruption control experiments with higher cut-off density will be possible. To demonstrate the applicability of this control technique in future tokamaks like ITER, we are planning to have experiments with feedback control of heating position.

3.2 Disruption control with External helical magnetic field

It has been found that external helical magnetic field is also effective to suppress the tearing mode having a large amplitude. The time behavior of plasma current and pick-up of magnetic fluctuations is shown in Fig. 12 for a disrupted discharge (gray lines). An MHD activity began to grow when the value of $q_5$ became 4 at 140 ms. The analysis of the magnetic pick-up signals indicates that the main helical structure of the magnetic fluctuations was $(m=4, n=1)$ at the beginning of this instability but changed to $(m=2, n=1)$ soon. Following this structural change, the frequency decreased gradually from 5 kHz with increasing amplitude. When the frequency reached about zero and the fluctuating magnetic field reached about 30 G, the plasma disrupted. This type of instability has been suppressed by the external helical magnetic field of the Saddle coils; as a result, the disruption has been avoided, as shown in Fig. 12. Here a Saddle coil DC current of 0.9 kA was applied from 200 ms to 250 ms, after the rise of the instability. The $(m=2, n=1)$ field was the major helical component in the external field; the radial field was about 10 G near the $q=2$ surface, about 1% of the poloidal field. The amplitude of the magnetic fluctuations reduced by an order of magnitude with a 20% reduction of the frequency. Since the suppression of the $(m=2, n=1)$ mode was not affected by the polarity of the external field, the stabilization is not brought about by compensation for error field. The behavior of the fluctuating magnetic field indicates that the rotation speed of the island is modulated directly by $j \times B (m=2, n=1)$ force, where $j$ is toroidal current density in the island and $B_{(m=2, n=1)}$ is the $(m=2, n=1)$ component of the external field. Oscillating velocity shear induced around the island by the modulation seems to provide the stabilizing effect.

If the natural rotation speed is low, we can not expect a strong stabilizing effect of the DC helical field because the induced velocity shear can not be high. However, it would be possible to improve the velocity shear by enhancing the island rotation with rotating $(m=2, n=1)$ helical field. External rotating $(m=2, n=1)$ field generated by eight saddle coils (driven by two independent power supplies) has been applied to demonstrate the rotation control of the
island, as shown in Fig. 13. The typical plasma parameters used in this experiment are $q_S \sim 3$, $n_e \sim 1 \times 10^{13} \text{ cm}^{-3}$, $B_t \sim 1.2 \text{ T}$, $I_p \sim 240 \text{ kA}$, $R \sim 1.3 \text{ m}$ and $a \sim 0.3 \text{ m}$. The direction of the rotation, frequency and amplitude of the magnetic field are controllable. The maximum frequency limited by the power supplies is 5 kHz, and the amplitude of the magnetic field at this frequency is $\sim 5 \text{ G}$. One of pick-up coils being sensible almost only to the magnetic fluctuations of plasma was used to check locking and unlocking of the magnetic island motion to the external magnetic field rotation.

The Fourier spectrum of pick-up signal and the saddle coil current are shown in Fig. 13. The external field was applied from 620 ms to 800 ms with frequency sweep. The mode frequency went down at 620 ms suddenly from 2.8 kHz to 2.5 kHz and then ramped up gradually to 3.2 kHz, following the external field rotation; the rotation control of the island has been demonstrated successfully.

The mode locking to the rotating external field is observed when the frequency of the external field rotation is close to the natural frequency of the island rotation. The phase lag of the magnetic island behind the applied field is sustained during the locking. When the frequency of the external field is higher than the natural frequency of $(m=2, n=1)$ mode, the phase of the external field is in advance to that of the magnetic island. This indicates that the external field pulls up the magnetic island rotation against a viscous friction from the surrounding plasma. When the phase lag become close to 90 deg., the locking is terminated. The phase lag is negative when the external field frequency is lower than the natural frequency of the island rotation. The critical phase lag is about -90 deg. in this case.

![Fig. 13 Island Rotation Control by Rotating Helical Field](image)


Improvement in particle and heat flux control in H-mode plasma is one of the most important research issues. We are investigating the effect of the helical magnetic perturbation to control plasma density and the divertor biasing to control the particle and heat fluxes into the divertor plates as described in this paper. Adding these studies, we will begin in 1995 the intensive study on the particle and heat flux control with a new closed divertor with effective pumping of about $1.3 \text{ m}^3/\text{s}$ for hydrogen, by replacing the old open divertor of which plates are made of graphite (see Fig. 14).
The purpose of this program is understanding divertor plasma physics and making database for an optimized divertor geometry with a dense and cold divertor plasma in 1995 to contribute the design and experimental plan on JT-60U divertor modification. The divertor plates of the new divertor are made of stainless steel to investigate phenomena associated with a metal divertor. The H-mode confinement property and the threshold power for the H-transition with the closed divertor will be also studied.

Since the saddle coils and the ladder coils inside the vessel have been removed to install the new closed divertor, it is impossible to do the disruption control experiment with the helical magnetic field in 1995. We are proposing to re-install more optimized helical coil system in 1996 as one of the next steps of disruption control study. Since the operation with $B_t=2.1$ T will be available from 1995 experiments, the cut-off density for 60 GHz ECH will increase, and therefore the disruption control will be examined at the density limit and the radiation limit. Feedback control of heating position with ECH will be also tried in 1996.

Developing current drive technique is another important issue for a steady fusion reactor. We are investigating electron current drive with fast wave (FWCD) launched by a four loops antenna system. We have already demonstrated the electron heating and the anisotropy in electron velocity distribution function. But, the electron temperature is too low to have sufficient efficiency to achieve a certain amount of current drive in present JFT-2M. We have a plan to make ECH power up from 200 kW to 600 kW in the $B_t=2.1$ T operations for preparing higher electron temperature. After the power-up, it will be possible to observe FWCD in global plasma parameter: loop voltage.

Dedication

Dr. Hikosuke Maeda, the head of the JFT-2M team, breathed his last on 29 November 1994. We would like to dedicate this paper in token of our appreciation and respect for his life devoted to investigation of plasma physics and controlled thermonuclear fusion.
References


Overview of the KT-2 design in view of advanced tokamak researches

S. K. Kim (KAERI)
KT-2 Design Features for Advanced Tokamak Research

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1. MOTIVATION

In 1993, JET Joint Undertaking officially declared it had achieved the "scientific breakeven" where an equal amount of fusion energy was produced as has been put into the plasma, i.e., fusion efficiency(Q) of 1.1. This statement was followed by an expression of the confidence in the achievement of the ignition on the ITER (International Thermonuclear Experimental Reactor) during the first phase of its operation. The ITER machine is presently under engineering design process in cooperation between the US, European Union, Japan, and Russia. An ignition reactor such as the ITER can produce fusion power without any external power input for heating and maintaining the plasma (Q=∞). The necessary heating power is provided internally with self-sufficiency, i.e. efficient confinement and thermalization of the energetic 3.5 MeV alpha particles produced from each fusion reaction. The machine is planned for operation by 2005.

During the planned second phase of its operation, ITER will concentrate on steady-state operation studies with 100% non-inductive current drive from various techniques including NBI, FWCD, and so on[1]. Meanwhile, since mid-80's, there have been considerable efforts in alternative tokamak reactor concepts for the required improvement of the poor economy which ITER-type tokamak reactors suffer. This need is widely shared within the fusion community, and it originates from the conservatism in the ITER design philosophy. The efforts for a more economical tokamak reactor can be summarized as the optimized utilization of the bootstrap current and of active current profile control, and thus achieving steady-state operation with minimum of external...
current drive and reducing the plant size if possible. In particular, the ARIES design efforts[2] and the SSTR design activities at JAERI[3] have been considered as the most noticeable outcome from these efforts. In the 90's, these efforts are being conceptualized as "advanced tokamak"[4], and is becoming one of the main issues in tokamak research. For example, extension of the machine life of JET to 1998 has been adopted in 1994 as the new "Framework Programme IV" for studies in the advanced tokamak operation scenario, as well as in advanced divertor concepts such as Mark-II and Mark-IIGB for ITER.

2. The "Advanced Tokamak" Concept

In terms of tokamak physics, the "advanced tokamak" concept is motivated first of all by the desire to minimize plasma current and disruption forces while maximizing poloidal beta and bootstrap current fraction. In case of the ARIES reactor design, this lead to a tokamak configuration with a large aspect ratio ($A = 4.5$) and high magnetic field. Thus employment of superconducting magnets is a mandate for an advanced tokamak reactor. Such a tokamak will operate in the parameter regime where $H_f \geq 2.3$, $\varepsilon \beta_p \approx 1$, $q_a > 1$, simultaneously. The total plasma pressure, or $\beta_n = \beta/(l/aB)$, can be simultaneously high in an advanced tokamak if operated with $q_a > 1$ and negative shear at core, $q_0' < 0$. To facilitate high $\beta_p$ and $\beta_n$, strong plasma shaping is mandatory with the elongation $\kappa \approx 2$ and the triangularity $\delta \geq 0.5$.

Optimized utilization of the bootstrap current production is an essential feature of the advanced tokamak concept. The bootstrap current fraction to the total plasma current must be maximized to minimize the need for an external current drive. In this regard, large-aspect-ratio ($R/a \gg 1$) configuration is preferable since $l_{BS} \propto A^{1/2} \beta_p \propto A^{1/2} \beta_n \approx q_n$. In particular, the BS current profile should be consistent with the total current profile required for MHD-stable operation, i.e., a good BS current alignment should be maintained. For this, it is extremely important to be able to precisely control the current/pressure profiles with suitable means of localized current drive.

The ARIES reactor concept, for example, anticipates a completely steady-state compact reactor with superconducting magnets, with two current drives for the seed current at the plasma core and for profile control at the edge. In addition, reasonable extrapolation of current technology has been assumed for the analysis. Typically the profile control is achieved with lower hybrid current drive(LHCD), and the seed current is driven by neutral beam injection(NBI) for SSTR, and fast-wave current drive(FWCD) for ARIES, and both for ITER. Electron cyclotron current drive(ECCD) is being studied for an alternative profile control tool. A detailed design study of advanced tokamak reactors indicate that significant improvement in economy both for the cost of electricity(COE) and for the plant size, by factors ranging 2 to 4 (see Table 1 below).
Table 1. Improved Economy of an Advanced Tokamak Reactor.
(Quoted in R. Goldston, Plasma Physics and Controlled Fusion 1994)

<table>
<thead>
<tr>
<th></th>
<th>500 MWe</th>
<th>1000 MWe</th>
<th>2000 MWe</th>
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<tbody>
<tr>
<td></td>
<td>Nom.</td>
<td>ADV.</td>
<td>Nom.</td>
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<tr>
<td>COE, $/kWh</td>
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<td>0.212</td>
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<tr>
<td>Capital Cost, B$</td>
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<td>4.5</td>
<td>13.0</td>
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<tr>
<td>Steam-State</td>
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<tr>
<td>COE</td>
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<td>0.149</td>
<td>0.187</td>
</tr>
<tr>
<td>Capital Cost</td>
<td>8.3</td>
<td>4.1</td>
<td>11.0</td>
</tr>
</tbody>
</table>

Reference COE = $0.15/kwh. (twice the 1994 US market rate).
Nom(inal physics): $H_2$, BN$_2.5$, ADV(anced physics): $H_4$, P$_N^6$.

For the case of ARIES-I (1,000 MWe, 2,544 MWt), COE was estimated to be 65 mill/kWh (constant 1988 dollars), compare this with that of advanced fission reactors, 47 to 78 mill/kWh, and of coal-fired plants, 50 mill/kWh. These estimations manifests the importance of studies to provide the necessary realism to this "advanced tokamak" concept.

3. Machine Design of the KT-2 Tokamak and the Program

KAERI started the KT-2 medium-sized tokamak in 1992 under the governmental "Long-Term Atomic Energy R&D Plan, 1992-2001". The goal of the KT-2 project has been set to establish a viable nuclear fusion energy research program at KAERI and in Korea, exploiting existing nuclear engineering infrastructure in synergistic combination with basic tokamak device engineering developed through the design, construction, and operation of the nation's first tokamak KT-1 in the 80's, and with the participation of domestic plasma physics and engineering research groups. It is strongly intended to involve advanced nuclear and other engineering capabilities in fusion research through KT-2 project and the associated fusion engineering projects, which include developments of superconducting magnet technologies, low-activation structural and nuclear materials, fusion power reactor systems study, and so on.

These goals and mandates of the KT-2 tokamak project resulted in both conservatism and progressivism in its philosophy. The conservatism stems from the fact that KT-2 will be the first major tokamak experiment in Korea, and thus the mandate that, above all, it must operate successfully. The risk in its design, construction, operation should be minimized through precise and realistic estimation of technical and other resources available domestically and, if any, from abroad. The progressivism is derived from the same fact: once the machine operates successfully, we have the freedom of pushing the tokamak physics and fusion engineering research as far as it is compatible with the machine. This progressivism is also strongly supported by the
institute's main task: development of technologies for peaceful uses of nuclear energy including fusion. Efforts have been concentrated to put together these two apparently incompatible mandates. The result is the KT-2 tokamak design where these efforts are embodied in setting the research goal progressively in advanced tokamak studies with a conservatively designed large-aspect-ratio configuration[5]. Below, the relevancies of KT-2 machine design and research program to advanced tokamak research will be outlined.

4. KT-2 Design Features for Advanced Tokamak Research

Basic specifications of the KT-2 tokamak are summarized in Table 2 below. It is a typical medium-sized machine, with an estimated OH confinement time of $\sim 50$ msec. In summary, it is a large-aspect-ratio tokamak with intensive heating suitable for studies of physics and engineering issues in advanced tokamak operation. Design features of KT-2 tokamak machine relevant to such purposes are following:

(1) Large aspect ratio

The KT-2 tokamak has a relatively large aspect ratio ($R/a=5.4\sim7.0$). relatively higher bootstrap current fraction since: advantage for advanced tokamak operation

\[ I_{bs} \propto A^{-1/2} \beta_p \propto A^{1/2} \beta_N q_a \]

where $\beta \leq \beta_N \frac{I_p}{a B_t q_o}$ or $(\varepsilon \beta_p / \varepsilon / \varepsilon) \leq 0.03 \frac{1+x^2}{2 q_o^2}$

This advantage renders the LAR configuration most attractive for commercial tokamak fusion power plants. For advanced tokamak reactors, in addition, high magnetic field and thus the employment of superconducting magnets are essential to achieve the required confinement. However, KT-2 adopts ordinary-conducting TF magnets due to lack of domestic superconducting magnet technologies which is still in an embryonic stage. With the maximum designed magnetic field on axis of 3 Tesla, reasonable confinement characteristics are expected ($\tau_{\text{ITER-89P}} \approx 50$ msec). Furthermore, the LAR configuration led to a relatively simpler device engineering with minimized risks in construction, and thus embodies the conservatism in the machine design.

(2) Long-pulse operation capability

The large aspect ratio configuration of KT-2 design provides ample room for the engineering of the center solenoid for the necessary magnetic flux and related coil assembly. In addition, the PF magnets are all located outside the TF coils, except the QQ' set for fast feedback control. This combination of the LAR configuration and the PF coil arrangement results in the uniquely high poloidal magnetic flux available from the PF system of KT-2, amounting to 9.9 Volt-second in bipolar operation. In Figure 1 below
Figure 1. A cutaway view of the KT-2 Tokamak and the magnet system
Figure 2. The KT-2 PF magnets system and coil specifications.
Table 2. Specifications of the KT-2 Tokamak

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
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<td>Major/minor radius, R/a (m)</td>
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<tr>
<td>Number of magnets</td>
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<td>Ripple, ( \delta B_r ) (%)</td>
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<td>Plasma current, ( I_p ) (kA)</td>
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<td>Flux swing, air-core, (V-sec)</td>
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<td>Current flat-top, ( t_c ) (sec)</td>
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<td>Energy confinement time, ( \tau_e ) (msec)</td>
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<td>Density, ( n_e ) (m(^{-3}))</td>
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<td>Heating, at commissioning(1999) (MW)</td>
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<td>Motor-Generator (MVA(GJ))</td>
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Table 3. KT-2 Operation modes.

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<th>Parameters</th>
<th>OH Baseline*</th>
<th>OH Long Pulse</th>
<th>1MW HiBS</th>
<th>5MW Baseline*</th>
<th>5MW HiBS*</th>
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<td>Toroidal field, ( B_t ) (T)</td>
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<td>2.0</td>
</tr>
<tr>
<td>Plasma Current (kA)</td>
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<tr>
<td>( q_0 )</td>
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<tr>
<td>( q_0^5 )</td>
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<td>6.2</td>
<td>2.79</td>
<td>3.2</td>
</tr>
<tr>
<td>( I_t )</td>
<td>1.0</td>
<td>-</td>
<td>-</td>
<td>0.95</td>
<td>0.90</td>
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<td>Average electron density ((10^{19} \text{m}^3))</td>
<td>10.0</td>
<td>3.5</td>
<td>2.6</td>
<td>10.0</td>
<td>5.5</td>
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<tr>
<td>Loop voltage (volt)</td>
<td>1.8</td>
<td>-</td>
<td>-</td>
<td>0.25</td>
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<td>Electron temperature (keV)</td>
<td>0.5</td>
<td>0.35</td>
<td>0.85</td>
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<tr>
<td>( \beta_{N=5}/(I_p/aB_t) )</td>
<td>0.75</td>
<td>0.4</td>
<td>1.27</td>
<td>2.8</td>
<td>3.8</td>
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<tr>
<td>( \varepsilon )</td>
<td>0.11</td>
<td>0.15</td>
<td>0.42</td>
<td>0.41</td>
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<td>Energy confinement time (ms)</td>
<td>80</td>
<td>80</td>
<td>25</td>
<td>60</td>
<td>30</td>
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<tr>
<td>( H_t )</td>
<td>2.0</td>
<td>2.0</td>
<td>2.0</td>
<td>3.0</td>
<td>3.0</td>
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<tr>
<td>Bootstrap fraction (%)</td>
<td>-</td>
<td>-</td>
<td>50</td>
<td>45</td>
<td>80</td>
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<tr>
<td>ICRF power (MW)</td>
<td>-</td>
<td>-</td>
<td>1.0</td>
<td>5.0</td>
<td>5.0</td>
</tr>
<tr>
<td>ECRH power (MW)</td>
<td>-</td>
<td>-</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
</tr>
</tbody>
</table>

* Different from CDR94, results from TSC with Coppi-Tang transport.
§ Discharge parameters changed from CDR94, where \( B_t/I_p = 1.5/200 \).
a cutaway view of the KT-2 machine including the magnets is shown. Details of the PF magnet system are in Figure 2, with discharge parameters of the various operation modes of KT-2 (see Table 3 above). These KT-2 operation modes are deduced from the PF system with design boundary parameters as tabulated in Table 4 below, and from equilibrium analysis.

The high magnetic flux available from the KT-2 PF system facilitates the discharge duration as long as over 30 seconds (see Table 4) with reduced current and field (the HiBS mode), and over 4 seconds in OH discharge at full parameters of 3 Tesla and 500 kA. Numerical modelling analysis of typical KT-2 discharges[6] indicates current relaxation time scales of maximum ~1 second. Therefore, it is readily possible to study pseudo-steady-state, advanced tokamak discharges on KT-2 where effects of profile control and its development in time can be observed in enough detail.

To see the effects of such long-pulse operations on the magnet structure and its integrity, temperature build-up and thermal cycles of the magnets has been analysed for the conceptual design with ANSYS. Preliminary results indicate that active cooling of the magnets with chilled water and/or optional LN$_2$, along with 10 to 15 minutes interdischarge rest time, should be sufficient. However, the KT-2 TF magnets should eventually be upgraded with superconducting ones, not only for completely steady-state operation, but more importantly for improving the confinement to necessary level for studies of advanced tokamak reactors. This will materialize in KT-3, planned ca. 2010.

(3) Plasma shaping and stability capabilities

To produce high-beta plasmas necessary for advanced tokamak research, KT-2 should have appropriate shaping capability of the magnetic surfaces. Generally an elongation $\kappa > 1.8$ and a triangularity $\delta > 0.5$ is necessary. For KT-2, the PF magnets

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Operation Modes</th>
<th>OH Baseline</th>
<th>OH Long-Pulse</th>
<th>5 MW Baseline</th>
<th>5 MW HiBS</th>
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<tr>
<td></td>
<td></td>
<td>Baseline†</td>
<td>Long-Pulse‡</td>
<td>Baseline†</td>
<td>HiBS</td>
</tr>
<tr>
<td>Max. Toroidal field, $B_0$ (T)</td>
<td>3.0</td>
<td>3.0</td>
<td>3.0</td>
<td>1.5</td>
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<tr>
<td>Plasma Current $I_P$ (MA)</td>
<td>500</td>
<td>200</td>
<td>500</td>
<td>200</td>
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<tr>
<td>Max. Elongation $\kappa$</td>
<td>1.8</td>
<td>1.8</td>
<td>1.8</td>
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<tr>
<td>Max. Triangularity $\delta$</td>
<td>0.6</td>
<td>0.6</td>
<td>0.6</td>
<td>0.6</td>
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<tr>
<td>Max. Discharge duration, $\tau_a + \tau_f + \tau_d$</td>
<td>7.3</td>
<td>7.9</td>
<td>22.6</td>
<td>35.8</td>
<td></td>
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<tr>
<td>Ramp-up $\tau_u$ (sec)</td>
<td>0.5</td>
<td>0.2</td>
<td>0.5</td>
<td>0.2</td>
<td></td>
</tr>
<tr>
<td>Flat-top $\tau_f$ (sec)</td>
<td>4.2</td>
<td>6.6</td>
<td>19.5</td>
<td>34.5</td>
<td></td>
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<tr>
<td>Ramp-down $\tau_d$ (sec)</td>
<td>2.6</td>
<td>1.1</td>
<td>2.6</td>
<td>1.1</td>
<td></td>
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<tr>
<td>Flat-top Electron Temperature (eV)</td>
<td>560</td>
<td>-</td>
<td>1477</td>
<td>-</td>
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<tr>
<td>Flat-top Loop Voltage (V)</td>
<td>1.57</td>
<td>-</td>
<td>0.28</td>
<td>-</td>
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</tr>
<tr>
<td>Max. Flux Swing $\Delta \psi$ (V-s)</td>
<td>9.8</td>
<td>-</td>
<td>9.9</td>
<td>-</td>
<td></td>
</tr>
<tr>
<td>Confinement Enhancement Factor $H_f$</td>
<td>2</td>
<td>-</td>
<td>2</td>
<td>-</td>
<td></td>
</tr>
</tbody>
</table>

* Double-null operation; † PF design baseline modes.
system and the power supply is designed for reference values of $\kappa = 1.8$ and $\delta = 0.6$. For heavily elongated plasmas as in KT-2, vertical instabilities make it very difficult to achieve $\kappa > 2.0$. In KT-2, a passive stabilizer system, assisted with fast feedback control via the coil set QQ', has been designed and being analysed and debugged for the necessary performance against vertical instability in elongated plasmas, including the eddy current effects on the machine. Preliminary results indicate that a 5 mm excursion (rise time $\approx 10$ msec) be stabilized in about 50 msec with 50 V/turn in the Q coil. Design work of the whole feedback control system including diagnostics and the thyristor amplifiers is being carried out in cooperation with domestic industry and research groups from and abroad. Unlike the TF magnets which is powered by the 166 MVA MFG, the PF magnets will be powered directly from the 154kV grid system (20MVA at the commissioning, 40 MVA upgrade by 2001) for efficient and stable switching required for the feedback control.

(4) Double-null (DN) divertor

The divertor is considered a must for improved confinement with $H \geq 2$ and for efficient control of particle inventory and of the heat flux transported out of the plasma in a fashion required for advanced tokamak operation. In addition, it helps plasma shaping with $\delta > 0.5$ ($\kappa > 1.8$). In the initial commissioning period of KT-2(1999), the CFC tile-covered chamber wall will be used temporarily as the divertor plates since a separate divertor structure will not be available. A slot-type divertor structure with NEG and cryo pumping is being studied and designed, and is planned to be installed before 2000 when the auxiliary heatings will start to operate. In addition, the plasma facing components will be protected with CFC tiles for the heat flux during the RF-heated discharges and disruptions. The plasma modelling, design and engineering of the divertor is one of the main research topics on KT-2. A dedicated divertor plasma simulator machine, a 2m-long linear RF (3 kW) plasma source with versatile diagnostics and experimentation capabilities, is under construction and will be utilized to study divertor plate materials and their properties as well as investigations of divertor operating techniques.

5. KT-2 Programs for Advanced Tokamak Research

In addition to these machine design features, the KT-2 project involves other facilities and programs for advanced tokamak research as well as other engineering projects for KT-2 and its future upgrades. The KT-2 machine design and preliminary MHD/transport analysis with heating included resulted in the operation modes tabulated in Table 3 above. Based on these KT-2 operation modes, programs for advanced tokamak studies are under development as following:
Intensive heating of total 8-10 MW will be applied to KT-2 to produce near-beta-limit discharges, reaching as high as $\beta_N = 3.8$ in the "5MW, High Bootstrap" operation mode. Heating/CD with NBI, ICRF, ECRH, and LH are planned. As described in Table 3, the 5MW High Bootstrap mode has intentionally a lower current and field to study the near-beta-limit discharges and bootstrap current effects. In Table 5, plasma properties during the OH Baseline and the 5MW Baseline modes are compared from snapshot equilibrium calculations. The 5MW Baseline mode reveals substantial increase in plasma beta with heating(SOH), accompanied by changes in the current and the $q$-profiles. A more detailed analysis of the 5MW HiBS mode is being carried out and results are presented in this conference[6].

During the first two years of its operation including the commissioning period (1999-2000), the two OH modes and the 1MW "High Bootstrap" mode, where $f_{ES} = 0.5$ is expected, will be mainly investigation target. The machine will operate with NBI(300kW), ECRH(500kW), and ICRF(1MW). Due to the infantile stage of development of the FWCD technologies, operational emphasis will be on the current drive with NBI and the electron heating with ECRH for preparatory FWCD studies, especially antenna development and transmitter operation. The NBI system will be developed at KAERI, starting from a large area ion source (1995-96). Gyrotron and RF transmitter systems will be developed in cooperation with commercial suppliers and/or other institutes. A small scale(~50 kW) RF transmitter is under development for the purpose of obtaining operation experiences for these megawatt transmitters. A pellet injector will be added during this period for efficient fuelling and core density profile control.

After the commissioning, by 2001, ICRF power will be upgraded to 5 MW, NBI to 1 MW, and ECRH to 1 MW, and 1 MW LHCD system will be added, thus facilitating the current drive (with FWCD backed up by NBI) and profile control (ECCD

### Table 5. KT-2 Design Base Equilibria

<table>
<thead>
<tr>
<th></th>
<th>OH Baseline</th>
<th>5 MW Baseline</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>SOF</td>
<td>EOF</td>
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<tr>
<td>$I_p$ (kA)</td>
<td>500</td>
<td>500</td>
</tr>
<tr>
<td>Beta (%)</td>
<td>0.738</td>
<td>0.746</td>
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<tr>
<td>Beta_p</td>
<td>0.623</td>
<td>0.617</td>
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<tr>
<td>$q_0$</td>
<td>0.749</td>
<td>0.730</td>
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<tr>
<td>$q_{ES}$</td>
<td>2.648</td>
<td>2.637</td>
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<tr>
<td>$I_{f}$</td>
<td>0.908</td>
<td>0.900</td>
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<tr>
<td>flux* (V-s)</td>
<td>2.319</td>
<td>-4.319</td>
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</table>

SOF: Start of Flattop, SOH: Start of Heating, EOF: End of Flattop
(*) Biased operation starting from 5.53 Volt-sec.
and LHCD) study of the high beta discharges in KT-2. During this and the following years with the full design-value operations, the "5MW HiBS" mode and the "5MW Baseline" mode will be the main targets of investigations. Development efforts for the "advanced tokamak" operation scenarios in these modes are already initiated, including both analytical and numerical analyses of the heating and the current drive processes and their effects on the stability and transport in KT-2.

(2) Profile diagnostics development

Diagnostics development have been carried out in cooperation with domestic research groups at various universities. Such extramural cooperations is the key element of organizing a nationwide research group centered around the common goal, KT-2, and will continue to be expanded in the coming years in a close consultation with the participating institutions.

The R&D emphasis is on assuring the profile measurement capabilities with necessary resolutions and on diverter diagnostics. Presently, ECE heterodyne receivers and FIR multichannel laser interfero-polarimeter is being developed, with single channel systems being tested on the KT-1 tokamak as well as an soft X-ray photoelectron spectrometer. In addition, developments of VUV/XUV spectrometers for extensive CHERS measurement (passive and active), neutral particle analyzers(STA and time-of-flight), X-ray pulse height analysis system, and so on, are being carried out. In 1996, design and construction of a Thomson scattering system and SXR cameras will start, planned to be operational during the commissioning phase(1999). A linear RF plasma source is constructed as a diverter plasma simulator, for studies of divertor operation modes/materials and diagnostics development.

(3) Superconducting magnet technologies

For advanced tokamak research, especially in a large aspect ratio tokamak, a high toroidal field of 8-12 Tesla are crucial to achieve necessary confinement at low plasma current as well as steady-state operation capabilities. For KT-2, due to lack of suitable domestic technology, immediate employment of the superconducting TF magnets has been temporarily avoided. Instead, long-pulse operation capability of the KT-2 magnets system of over 20 seconds satisfies the immediate need for current drive and profile control studies relevant to advanced tokamak operation. Therefore, development of superconducting magnet technologies are very urgent and will be the main engineering R&D program.

A feasibility study where R&D demands are evaluated in detail, with the goal of designing and fabricating in parallel with the KT-2 machine a large-bore superconducting
magnet for one of the PF coil of KT-2 (coilset 7). The aim of this study is to establish the minimum domestic device engineering and technology necessary for the development and operation of the large-scale superconducting magnet system suitable for tokamaks. Therefore, it will be a cooperative research with foreign institutes as well as domestic industry and research groups. A superconducting upgrade of KT-2 (currently conceptualized as KT-3 to be operational ca. 2010) is planned for the necessary confinement improvement and the completely continuous operation for reactor-relevant studies, in combination with advanced operation techniques obtained from the KT-2.

(4) Reactor materials and systems study

Although in a somewhat weaker relation to the KT-2 project, group of projects are being organized in fusion reactor engineering, some of them as long-term projects like the superconducting magnet development described above. These include diverter/PFC material and thermal hydraulic design studies for KT-2, structural and nuclear

<table>
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<tr>
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<th>93</th>
<th>94</th>
<th>95</th>
<th>96</th>
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<td>1MW ICRF Procurement</td>
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<td>5MW upgrade/install</td>
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<td>Development Ion Source</td>
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<td>Construction, 300kW prototype</td>
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<tr>
<td>1 MW NBI upgrade</td>
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<td>0.5 MW ECRH on KT-1</td>
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<td>0.5 MW ECRH on KT-2</td>
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<td>ECRH upgrade to 1MW</td>
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</table>

*: preparation,  : execution

MILESTONES:
3. 1995. 9. CONSTRUCTION START
5. 1998. 1. Completion of Assembly
6. 1998. 12. FIRST PLASMA (from grid with 20 MVA)
7. 1999. 6. Start "1 MW HiBS" operation with NBI(0.3MW), ICRF (1MW), and ECRH(0.5 MW)
8. 1999. 12. Full-Parameter "OH Baseline" operation(3 Tesla, 500 kA)
9. 2001. 1. Start "5 MW Baseline/HiBS" operation combined with ICRF(5MW), NBI, ECRH, and LH (1 MW each).
materials for fusion reactors, and reactor systems study with references to the ARIES and the SSTR tokamak reactor concepts. These activities will be carried out in close relation to the other nuclear engineering projects at KAERI. Examples are: the unique, newly inaugurated high-neutron-flux research reactor HANARO, the fission/fusion nuclear data development and management, low-activation alloy research for reactor structural materials, and neutronics/radiation shielding R&D.

4. Conclusion

The KT-2 tokamak project at Korea Atomic Energy Research Institute (KAERI) effectively combines (a) the conservatism for successful machine construction and operation and (b) the progressivism of pursuing the advanced tokamak research on a dedicated medium-sized machine, in a machine design of a large aspect ratio, divertor tokamak with intense RF and beam heating of total 8-10 MW. The machine will start operation in 1998, and the "advanced tokamak" operation by 2001 (see Table 6 in the foregoing page). The project is also supported concurrently with relevant fusion engineering R&D programs such as superconducting magnets, divertor/PFC and structural materials, and reactor systems study, and so on. These efforts are all aiming the conceptual design of the superconducting upgrade of KT-2, or the KT-3, planned to be operational ca. 2010.

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REFERENCES


Self-organized microstructures and their role in tokamak transport

F. C. Schüller (FOM-Rijnhuizen, The Netherlands)
SELF-ORGANIZED MAGNETIC MICROSTRUCTURES
AND THEIR ROLE IN TOKAMAK PLASMAS

F.C. SCHÜLLER

FOM-Institute for Plasma Physics 'Rijnhuizen',
Association Euratom-FOM

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Chen Chi Chu
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B.J. Grobben
G.C.H.M. Verhaag
1. Introduction

2. Indications of magnetic structures:
   - experiment
   - theory

3. RTP and the Thomson-scattering system:
   - set-up
   - error analysis

4. Experimental results:
   - ohmic plasmas
   - ECR-heated plasmas
   - time dependence
   - current- and density-dependence
   - pellet-injection

5. Discussion
   - central filaments
   - off-centre structures
   - thermal barriers
   - negative shear

6. Future plans

7. Conclusions

INDICATIONS OF MAGNETIZED STRUCTURES

What are they?
- magnetic islands
- current sheets
- filaments = long thin threads with enhanced pressure/temperature/current density?
  stretched along magnetic field lines

Experimental Observations in Non-Tokamak
(but magnetized) Plasmas
- solar coronal loops
  (Parker)
- plasma focus
  (Bostick)
- magnetized arcs
  (Schram)
- MHD-generator plasmas
  (Rietjens)

"streamers = bundle of sub-streamers"
INDICATIONS OF MAGNETIZED STRUCTURES

Experimental Observations in Tokamak Plasmas:
- visible light filaments in TFTR (Zweben)
- correlation lengths of magnetic fluctuation
  \[ L_{c||} \geq 20 \text{ mtr}, \quad L_{c\perp} \leq 10 \text{ cm} \] in JET (Malacarne, Duperrex)
- Pellet ablation irregularities (Dubois)
- LIDAR profiles JET (Nave)
- Thomson scattering in ECR-heated plasmas: PBX (Hsuan), RTP

FIG 1 Images of D₂ light emission from the inner wall region of TFTR for a typical neutral-beam-heated plasma. Fig 1(a) shows the camera field of view, which for (b)-(d) is the whole field of view. The photos on (b)-(d) are single, video-selected, snapshot for 15 ps pulse, taken from a typical TFTR NBI discharge. The D₂ light features irregularly elongated filaments that vary on a petawatt-scale from static plasma. Note that the temporal variation of D₂ is due to a modulation of the local neutral density. The bright disc at the left is due to enhanced, reflected light at the edge of a pellet in the tokamak.
Top: q-profile with shear plateaus for shot # 4159
Bottom: q-profile superimposed on the Hα signal

Reference: LIDAR profiles in JET (Nave et al, NF 32 (1992))
TVTS data of the temperature and the density profiles for an outside O-mode launcher with about 70 kW ECH power, \( n_e = 0.5 \times 10^{13} \text{ cm}^{-3} \).

"Profile consistency"  
Montgomery  
Biskamp  
and others:

\[
p(r) = i(r) = \left( 1 + \left( \frac{q_a}{q_0} - 1 \right) \frac{r^2}{a^2} \right)^{-2} \]

"Filamentary MHD"  
Kinney, Tajima  
McWilliam, Petviashili

TVTS data of the temperature and the density profiles for an outside O-mode launcher with about 80 kW ECH power, \( n_e = 1.1 \times 10^{13} \text{ cm}^{-3} \).
**CONFIRMATION OF KADOMTSEV-TAYLOR CURRENT PROFILES**

\[ j = j_0 \left(1 + \left(q_0 - 1\right) \frac{r^2}{a^2}\right)^2 \] is equivalent to:

\[ j / (B_T/\mu_0 R) = \frac{2}{q_0} \left(1 + 2\pi \left(\frac{1}{q_0} - \frac{1}{q_0}\right) \left(r/\sqrt{\mu_0 R I/B_T}\right)^2\right) \]

**TEXTOR polarimeter results (Soltwisch et al. IAEA, Kyoto, 1986)**

<table>
<thead>
<tr>
<th>I(kA)</th>
<th>B_T(T)</th>
<th>q(a)</th>
<th>q(0)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>485</td>
<td>1.58</td>
<td>2.10</td>
</tr>
<tr>
<td>2</td>
<td>473</td>
<td>1.70</td>
<td>2.33</td>
</tr>
<tr>
<td>3</td>
<td>476</td>
<td>2.51</td>
<td>3.30</td>
</tr>
<tr>
<td>4</td>
<td>340</td>
<td>1.70</td>
<td>3.17</td>
</tr>
<tr>
<td>5</td>
<td>202</td>
<td>1.70</td>
<td>5.46</td>
</tr>
<tr>
<td>6</td>
<td>191</td>
<td>2.00</td>
<td>6.34</td>
</tr>
</tbody>
</table>

Thin lines: experimental results; fat line: K-T profile \( q_a = 3, \) \( q_0 = 0.75 \)

Rehui-Lallia semi-empirical transport model:
- quite successful in describing transport in a tokamak
- based on existence of chains of small islands.
- These have not been demonstrated experimentally, and their properties are conjectured, not measured.
"Filamentary Magnetohydrodynamic Plasmas"

Kinney, Tajima, McWilliams, Petviashvili, 1985

Phys. Plasmas 2, 260 (1985)

Surface of constant $B$ for fixed dipole magnitude run in which the energy per dipole is 0 on (a) at $t=0$ and (b) at $t=3$, as the field begins to organise.

Continuation of the run from the previous figure, almost generation of magnetic field into flux tubes at times (a) $t=0$ and (b) $t=1.35$

Field line Tracing

(Novikov, de Rovnag)

The effect of 2 filaments of 400 A placed on the $q=1$ and $q=2$ surface of a 100 kA discharge in RTP $g_a=4.2$
Magnetic field line tracing for a filamented plasma in RTP-geometry for $\tilde{q}_a = 4.5$ plasma. Number of filaments: 101,500 with $\Delta l = 0.8\text{ A}$.

Rational $q$-values lead to local decrease of magnetic diffusivity $\tilde{m}$.

Because of stochasticity $q(r)$ has to be redefined as a surface average:

Boronized vessel

**ECRH**
- 2 gyrotrons
  - 60 GHz, 200 kW, 100 ms
- Low Field Side O-mode
- High Field Side X-mode

Since February '94:
  - $2\omega_{ce}$ at 110 GHz, 500 kW, 200 ms

**Pellet injector**
- 8 shot
  - tiltable in poloidal plane
<table>
<thead>
<tr>
<th>RTP Diagnostics</th>
</tr>
</thead>
</table>

**Magnetics**

- Interferometer: 19 ch $n_e(r,t)$
- ECE heterodyne receiver: 20 ch $T_e(r,t)$
- Thomson scattering: $n_e(r), T_e(r)$
- Neutral Particle Analysis: $T_i$ (coarse)
- Soft X-ray tomography: 80 ch, 5 cameras
- Visible light tomography: 80 ch, 5 cameras
- Bolometer

**Spectroscopy**

- ECE, Visible light, XUV, Soft X-ray

**Transmitted ECR power:** 9 ch measurement
The TV Thomson scattering system

Laser: Ruby, 25 J, <0.2 mrad
Detector: CCD
100 spatial channels
64 wavelength channels
Spatial resolution: $\Delta z = 1.7 \text{ mm} \left(\frac{a}{100}\right)$

$\Delta \% = 1\%$, when photon yield is large: $0.5\%$

Example of measured spectrum:
conditions typical of what follows:
ECH; $T \approx 2\text{ keV}; n \approx 2 \times 10^{19} \text{ m}^{-3}$

Error estimated from $\chi^2$: $dT = 0.18\text{ keV}$
Addition of 15 spectra; systematic distortions are small

\[
T_e = 1.7 \text{ keV} \pm 0.1; \quad n_e = 3.0 \pm 0.10^{19} \text{ m}^{-3}
\]

Error analysis

Two independent estimators of the error on \(T_e\)
agreement \(\Rightarrow\) trust error bars

- Weighted fit procedure:
  scatter of data points around model \(\Rightarrow\) error from \(x^2\):

- Simulation of measurement, including sensitivities etc.
  Assume Poisson statistics of photo-electrons is dominant contribution to error:
Plasma light

- Measured separately by not firing the laser

- H$_\alpha$ - of same order as scattering signal - is cut out of the spectrum

- Otherwise, plasma light is very low (<5%) thanks to
  - high brightness of laser
  - short gating of detector (100 ns)

- Neglected in present analysis

---

Examples of High n, ohmic profiles

![Graphs and data](datafile:/usr1/thomscat/data/a931216.019)

\[ q_n = 6.8 \]

\[ P_n = 130 \, kW \]

\[ \frac{P(2)}{P_0} = \left(1 + \frac{2}{N_0} \right)^2 \]
Example of quasi periodic structures in a ohmic discharge, during gaspuffing ($n_e=4\times10^{19} \text{ m}^{-3}$). The pixels of the CCD detector were regrouped to increase the spatial resolution. $\Delta z = 8.5 \times 10^{-4} \text{ m}$ $\Rightarrow \Delta z/\alpha = 5 \times 10^{-3}$.
Profiles with central ECH

Cartoon summary of observed features

Several peaks in the ECH deposition region

Steps

Quasi-periodical structures
Examples of STEPS

$q_a = 5.6$

Averaged profile over 15 identical pulses

$P_{EC} = 270\,\text{kW}$

$P_{EC} = 120\,\text{kW}$

$P_{EC} = 120\,\text{kW}$

record

standard

averaged

$V T_e = 0.7\,\text{MeV/m}$

$0.2$

$0.1$

gradient wiped out because of pulse-to-pulse variations in plasma position $z = \pm 3\,\text{mm}$

Far away from step:

$V P_e = 2 \cdot 10^6\,\text{N/m}^3$

$6 \cdot 10^5 \leq 6 \cdot 10^4$

$J_\perp > 10^6\,\text{A/m}^2$

$3 \cdot 10^5 \leq 3 \cdot 10^4$

$J_\perp / J_\parallel > 0.2$

$0.06 \leq 0.006$

Examples of FILAMENTS

$P_{EC} = 270\,\text{kW}$

$P_{EC} = 120\,\text{kW}$

$P_{EC} = 120\,\text{kW}$

$G_a = 10$

$G_a = 4.6$

$G_a = 3.3$

Two necessary conditions for filaments:

$* \quad G_a < 1$

$* \quad P_{EC}/\text{vol} \gg j^2$
Example of FILAMENTS, 0.3 ms after switching off ECRH

DISCUSSION

CENTRAL FILAMENTS

* Closed flux tubes
  i.e. \( m/n = q(r) \) ?

* Same size as dominant \( k \) in density fluctuations measurements

* Why only in the centre?
  Low shear?

* Local magnetic diffusion time = 100 \( \mu \)s
  so filaments will carry excess current density;
  structures cannot be magnetic islands unless shear is inverted

* Extremely well-confined:
  - \( \chi = 10^{-2} \text{ to } 10^{-1} \text{ m}^2/\text{s} \)
  - long lived: \( \tau = 0.1 \text{ to } 0.3 \tau_E \)

Filaments are still there!
(\( \text{step} \)) \( \tau_{Le}^{\text{global}} = 3 \text{ ms} \)

Filaments die out in about 0.3 confinement time
DISCUSSION

OFF-CENTRE STRUCTURES

- Same size as central filaments = 1 cm
  statistically significant, but less pronounced in $\tilde{T}$ / $T$ and $\tilde{p}$

- Often $T$-profile non-monotonic whilst $p$-profile shows only steps

- Averaging over many identical discharges:
  - preferred positions? (under investigation)
  - rational $q$-surfaces?
  April 95: 19-channel polarimeter may give answer

- $T_e$-enhancement $\rightarrow$ $j$-enhancement:
  they cannot be standard magnetic islands, since strong shear has the wrong sign.

STEP NEAR $q = 1$

- Transport barrier:
  has also been observed by:
  - heat pulse propagation gives $\chi$ across $q=1 = 0.1 \text{ m}^2/\text{s}$
  - puncture of $q=1$ surface by pellets

- Strong pressure gradient implies:
  $j_\perp \leq j_\parallel$
  i.e. a poloidal current sheet

- How does this fit with sawtooth collapse modes?
DISCUSSION

EVIDENCE FOR q=1 THERMAL BARRIER

Phase and amplitude heat pulse propagation by modulated ECRH in RTP

Profiles of $T_e$, $n_e$, and $p$ measured with the TS diagnostic at the instant when the pellet is at: $z_{pet}=60$ mm (a-c), $z_{pet}=29$ mm (d-f), and $z_{pet}=85$ mm (g-i), where negative values denote positions after the pellet crossed the plasma center. The horizontal bands indicate plasma positions already traversed by the pellet. The open symbols (*) represent profiles before pellet injection.
THE INFLUENCE OF MULTIPLE THERMAL BARRIERS ON CONFINEMENT:

\[ \chi_{\text{eff}} = \frac{1}{\langle \chi(r) \rangle} \].

Therefore confinement will be determined by:

- the value of \( \chi^{\text{low}} \)
- the duty cycle:
  \[ \frac{W^{\text{low}}}{W^{\text{low}} + W^{\text{high}}} \]
- Not by \( \chi^{\text{high}} \).

example:

H-mode is thermal barrier at edge but improves \( T_E \) with factor 2.

NEGATIVE SHEAR DAMPS TURBULENCE

Off-axis ECRH on RTP at high density causes stationary
hollow \( T_E \)-profiles \( \rightarrow \) hollow j-profiles

central electron-ion coupling requires inward heat flux at \( \chi_E \)-values close to neo-classical values in areas with negative shear
JET: extreme low $n_e$ values in areas with negative shear caused by LHCD

Pulse No: 33160, LIDAR $T_e$ (R) along $Z_1 = Z_{\text{mag}}$

- $I_p = 3\text{MA}$
- $B_T = 3.4\text{T}$
- $P_{\text{LH}} = 5.5\text{MW}$
- $n_0(0) = 1.2 \times 10^{19} \text{m}^{-3}$
- $t = 10.5\text{s (no LH)}$
- $t = 10.52\text{s (no LH)}$
- $t = 13.75\text{s (LH)}$

FUTURE PLANS

HARDWARE

- Tangential scattering (2nd half '95)
  - is current-density filamented?
- Double pulse measurements
  - evolution of structures in time
- Polarimetry
  - relation between rational $q$ and structures

THEORY

- further development Hamiltonian chaos model for B-field distribution
- magnetic structures and Ohms Law
- thermal barriers
- negative shear
EXPERIMENTS

- Radial displacement of plasma column:
  - off-axis observation chord
  - poloidal dimension of structures

- Off-axis ECRH:
  - role of q=1 surface
  - structures outside q=1 radius

- 110 GHZ ECRH:
  - deposition profile width → structure width
  - one order of magnitude larger power density
  - exploration of negative shear region

CONCLUSIONS

* The magnetic topology of tokamak plasmas consisting of nested magnetic surfaces is easy to destroy and has probably to be abandoned as a working model.

* The broken state of tokamak plasmas only shows up in measurements if the spatial resolution is refined to a few per cent of the minor radius combined with a time resolution better than 10 µs.

* Localized additional heating with a high power density is able to highlight the broken state:
  - extremely hot closed flux tubes in the centre;
  - very steep temperature/pressure gradients near the q=1 radius;
  - irregularities in temperature/pressure further out towards the edge.

* Ohmic discharges show similar but much less pronounced behaviour.
CONCLUSIONS

* Transport inside flux tubes must be very low.

* Lifetime of the structures are a sizable fraction of the global confinement time.

* The structures are very likely related to density fluctuations as observed by collective scattering and other fluctuation measurements.

* The difference between the structures near the centre and those in the conduction zone is probably caused by the difference in shear.

* The structures near the centre could be described as hot magnetic islands embedded in a region of inverted shear.

* The structures with enhanced temperature and pressure in the conduction zone are difficult to understand as magnetic islands but could be described as "filaments" proposed by Taylor.

* Unbroken magnetic surfaces can act as thermal barriers

* The existence of thermal barriers changes drastically the interpretation of experimental diffusivity values in the light of theoretical models

* The damping of turbulence in areas with negative shear is one of the most important findings to strengthen the case of ADVANCED TOKAMAKS
Effects of α-particle driven current in advanced tokamaks

C. S. Chang (KAIST/NYU/PPPL)
Effects of alpha-driven current in advanced tokamaks

Introduction

- A tokamak fusion reactor requires an adequate toroidal electric current profile.
- Ohmic current has been the easiest one to drive. However, it is transient. A steady current is preferred.
- Neoclassical bootstrap effect may be able to supply most of the necessary current (amount). However, its radial profile is not satisfactory. Current need to be added at the center and middle (profile control).
- Well-known ideas for profile control

  FWCD-center
  LHCD-middle
  (ECCD and NBICD may be other options)

  They drive localized current.
  Multiple C sources are required.

  NBICD induces plasma rotation, coupling to another non-trivial problem
Effects of alpha-driven current in advanced tokamaks

This study raises two issues.

1. FWCD can be significantly modified by the presence of $\alpha$-particle in a tokamak reactor.

   * Electron current drive by toroidally directed fast waves ($e$-FWCD) is an attractive CD scheme because of the proven penetration capability of the fast waves into the plasma center.
   * Numerous experimental and theoretical studies

   * However, in a fusion reactor, there are energetic alpha particles which can absorb a significant portion of the fast wave energy by Doppler-shifted ICRH.

   * It is found that the absorption of toroidally directed fast waves by alpha particles results in a alpha current driven by asymmetric (in $v_\parallel$) ICRH. → $\alpha$-ICRCD

   * Radial profiles for $e$-FWCD and $\alpha$-ICRCD are quite different.

2. [Warning!]

   Fast wave parameters should be chosen carefully

   * Bad Choice : $\alpha$-ICRCD can be negative → destructive

   * Good Choice : $\alpha$-ICRCD can add current at the mid-radius → no need for a 2nd current drive scheme

C. S. Chang
Coulomb scattering operator

\[ C(f) = -\nabla \cdot \vec{R}_c, \quad C = C_v + C_\xi \]
\[ C_v = -\nabla \cdot (\vec{R}_c \dot{\xi}), \quad C_\xi = -\nabla \cdot (\vec{R}_\xi \dot{\xi}) \]

For \( v > v_c = \text{critical slowing down speed} \), the slowing down process is dominant.

\[ C_v(f) = C_v(f) \approx \frac{1}{T_s v^2 \partial v} \left[ v^2 \left( 1 + \frac{v_c^2}{v^2} \right) f \right] \]

Connor and Cordey, Nucl. Fusion 16, 185(1974)

For \( v < v_c \), the pitch angle scattering effect is strong.

Ordering

\[ C_v(f) \gg Q(f) \gg C_\xi(f) \quad \text{for } v > v_c \]
\[ C_\xi(f) \sim C_v(f) \gg Q(f) \quad \text{for } v < v_c \]

ICRH operator

\[ \vec{r}^{RF} = \Gamma_{RF} \vec{v} \]
\[ \Gamma_{RF} = -\delta(\omega - i \Omega - k \nu) \]
\[ \frac{\pi Z_\alpha^2}{8M^2} \left( |E_+ J_{n-1}|^2 + |E_- J_{n+1}|^2 \right) \frac{\partial f_0}{\partial v_\perp} \]

Express \( \vec{r}^{RF} \) in \( \dot{v} \) and \( \dot{\xi} \)

\[ \vec{r}^{RF} = -\delta(t - t_R) \tau_b D_b \sqrt{1 - \xi^2} \frac{\partial}{\partial v} \left[ \sqrt{1 - \xi^2} \dot{v} - \xi \dot{\xi} \right] \]
\[ \tau_b = \text{bounce time} \]
\[ D_b = \frac{\pi Z_\alpha^2}{8M^2 |\Omega| |\nu|} \left( |E_+ J_{n-1}|^2 + |E_- J_{n+1}|^2 \right) \]

Orbit averaged ICRH operator

\[ \{ Q \} = \{ -\nabla \cdot \vec{r}^{RF} \} = \{ Q \}_v + \{ Q \}_\xi \]
\[ \{ Q \}_v = \frac{1}{v^2 \partial v} \left[ v^2 (1 - \xi^2) D_b \frac{\partial f_0}{\partial v} \right] \]
\[ \{ Q \}_\xi = -\frac{1}{v} \xi R \frac{B_m}{\nu} \frac{\partial}{\partial \xi_m} \left[ \xi R (1 - \xi^2) D_b \frac{\partial f_0}{\partial v} \right] \]

\[ \xi_R = \frac{\nu}{v} |_{R=\text{Resonance}}, \quad \xi_m = \frac{\nu}{v} |_{m=\text{midplane}} \]
\[ B_R = B |_{R}, \quad B_m = B |_{m} \]
Effects of alpha-driven current in advanced tokamaks

\[
\tilde{\Gamma}^{RF} \text{ can be decomposed into:}
\]

\[
\tilde{\Gamma}^{RF} = \tilde{\Gamma}_v^R F \hat{v} + \tilde{\Gamma}_\xi^R F \hat{\xi}
\]

\[
\tilde{\Gamma}_v^R F \text{ induces banana trapping of passing alphas.}
\]

If RF is effective on \( v^\parallel < 0 \) alone

\[
v > v_c = \text{critical slowing down speed}
\]

\[
\begin{align*}
\text{weak pitch angle scattering:} & \quad j_a(v > v_c) > 0 \\
\text{strong pitch angle scattering:} & \quad j_a(v < v_c) \approx 0
\end{align*}
\]

C. S. Chang
Ion cyclotron resonance heating of α particle

\[ \Delta \epsilon = B_R \Delta \mu \]

\[ \text{ion cyclotron orbit} \]

\[ \text{circularly polarized wave} \]

\[ \text{Heating in } v_L \]

\[ j = j_e + j_\alpha \]

\[ j_\alpha = Z_\alpha e n_\alpha \left( 1 - \frac{Z_\alpha}{Z_{\text{eff}}} F_p \right) u_{\alpha \parallel} \]

\[ Z_\alpha = 2 \]

\[ u_{\parallel} = \frac{B}{B_0} \int d^3 v v_{\parallel} f_i > \]

\[ B_0 = < B > \]

\[ F_p = \text{passing electron fraction} \]

\[ \frac{Z_\alpha}{Z_{\text{eff}}} F_p : \text{electron screening factor} \]

Notice here that it is hard to drive α current at \( r \to 0 \) \((F_p \to 1)\) because \( Z_{\text{eff}} \sim 2 \) is expected.

\[ 1 - \frac{Z_\alpha}{Z_{\text{eff}}} F_p \ll 1 \quad \text{for } r \to 0 \]

Also notice that, even if we drive \( j_\alpha \) at \( r \to 0 \) with a very strong wave power, \( j_\alpha \) at \( r \to 0 \) is highly sensitive to \( Z_{\text{eff}} \) values \((\sim 2)\).
Effects of alpha-driven current in advanced tokamaks

**Alpha flow driven by ICRH**

Alpha gyro-kinetic equation

\[ \vec{v}_{gc} \cdot \nabla f = C(f) + Q(f) + S \]

\( S \): alpha source = \( S_\alpha \delta(v - v_0) \)

Use a double ordering scheme to solve for \( f \)

(1) Energetic ions have weak collisions

\[ f = f^{(0)} + f^{(1)} + \ldots \]

0th order

\[ \vec{v}_{gc} \cdot \nabla f^{(0)} = 0 \]

\( f^{(0)} \) is constant along guiding center orbit.

1st order

\[ \vec{v}_{gc} \cdot \nabla f^{(1)} = C(f^{(0)}) + Q(f^{(0)}) + S \]

average over the guiding center orbit

\[ \{\vec{v}_{gc} \cdot \nabla f^{(1)}\} = 0 = \{C\}(f^{(0)}) + \{Q\}(f^{(0)}) + \{S\} \]

(2) For economy, ICRH power \( \ll \) alpha power

\( \rightarrow \) Small-Q ordering \( (Q \ll S \sim C) \)

\[ f^{(0)} = f_0 + f^{(0)}_{RF}, \quad f^{(0)}_{RF} \ll f_0 \]

0th order

\[ \{C\}(f_0) + \{S\} = 0 \]

\[ f_0 = \frac{\{S_\alpha\} \tau_s}{4\pi(v^3 + v_0^2)} \quad \text{for } v \leq v_0 \]

: slowing down distribution

1st order

\[ \{C\}(f^{(0)}_{RF}) + \{Q\}(f^{(0)}_{RF}) = 0 \]

\[ \frac{1}{\tau_s v^2} \frac{\partial}{\partial v} \left[ (v^3 + v_0^3) f^{(0)}_{RF} \right] = -\{Q\}(f_0) \]

\[ \Rightarrow f^{(0)}_{RF} = -\frac{\tau_s}{v^3 + v_0^3} \int_0^v dv' v^2 \{Q\}(f_0), \quad f \cdot v > v_c \]
Effects of alpha-driven current in advanced tokamaks

For $\alpha$-ICRCD, $\{Q\}$ needs to be odd in $v_\parallel$.

$$j_\alpha \neq 0 \quad \rightarrow \quad f^{(0)}_{RF} \text{ is odd} \quad \rightarrow \quad \{Q\} \text{ is odd}$$

This can be achieved by having a directional $k_\parallel$ spectrum and an in-out asymmetric $|E|^2$ profile.

Divide $\{Q\}$ into two components:

$$\{Q\} = \{Q_v\} + \{Q_\xi\}$$

$$\begin{align*}
\{Q_v\}(f_0) &= \frac{1}{v^2} \frac{\partial}{\partial v} \left[ v^2(1 - \xi_R^2) D_v \frac{\partial f_0}{\partial v} \right] \\
\{Q_\xi\}(f_0) &= \frac{1}{v^2} \xi_R B_m \frac{\partial}{\partial \xi_m} \left[ \xi_R(1 - \xi_R^2) D_v \frac{\partial f_0}{\partial \xi_m} \right]
\end{align*}$$

where $(1 - \xi_R^2)/B_R = (1 - \xi_m^2)/B_m$

Notice that $j_\alpha^v$ and $j_\alpha^\xi$ have opposite signs.

$$j_{z,FWCD} = j_\alpha^\xi \parallel j_\alpha^v \parallel - j_\alpha^\xi \rightarrow$$

Maximize $j_\alpha^\xi$ and minimize $j_\alpha^\xi$ ($k_\parallel < 0$)

When $j_{z,FWCD} \approx 0$, we can use

$$j_\alpha^\xi > 0 \quad (k_\parallel > 0)$$
Absorbed power

\[ P_{RF} = \left( \int d^3 \nu \frac{1}{2} M v^2 Q(f_0) \right) \]

\[ = \frac{1}{2} M \frac{B_0}{B_m} \int_{-1}^{1} d\xi_m \int_0^{n_\nu} d\nu v^4 \tau_b(Q)(f_0) \]

Here the integration is over all the alpha particles (0 < \nu < v_0, passing, trapped).

Current drive efficiency

\[ \gamma_{CD} = \frac{I_{a,RF}[MA] \tau_e[m^{-3}] R[r_p]}{P_{RF}[MW]} \]

\[ \tau_e = 2n_0/3 \quad \text{for } n = n_0(1 - \frac{r_e^2}{a^2})^{1/2} \]

Demonstration of \( \alpha \)-ICRD

We use ITER CDA design parameters.

\[ R_0 = 6cm, \quad a = 2.15m \]
\[ n_0 = 1.8 \times 10^{20} m^{-3}, \quad T_0 = 20keV \]
\[ n_e = n_0(1 - \frac{r_e^2}{a^2})^{1/2}, \quad T_e = T_0(1 - \frac{r_e^2}{a^2})^{3/4} \]

We use a circular flux surface geometry for simplicity (with Shafranov shift).

We use an assumed profile for \( |E_+|^2 \)

\[ |E_+|^2 = |E_{+0}|^2 \left( \frac{\tau_e^4}{a^4} + 2 \right) \left\{ \exp \left[ -\frac{\theta^4}{(\pi/4)^4} \right] - \exp[-1] \right\} \]

\[ = 0 \quad \text{for } |\theta| > \pi/4 \]

\( (r_e, \theta) \) is with respect to the geometric center of the poloidal wall cross-section.
We can easily see that we can have an asymmetric Fig. (a) if \( Q \) is not symmetric in the sign of \( \eta \). (Q) is also not symmetric in \( y \) if it is directional and

\[
\sum_{k=1}^{n} R_k = 2 \text{ (MW/m}^2\text{)}
\]

\[
\sum_{k=1}^{n} P_k = 2 \text{ (MW/m}^2\text{)}
\]

\[
\text{Absorbed Power (MW/m}^2\text{)} \quad \text{Current (MA/m}^2\text{)} \quad \text{Current (MA/m}^2\text{)}
\]

\[
0.0 \quad 0.05 \quad 0.10 \quad 0.0 \quad 0.1 \quad 0.2 \quad 0.3
\]

\[
0.0 \quad 1.0 \quad 2.0 \quad 3.0
\]

\[
\text{Fig. 3}
\]

\[
\text{CENTER LINE}
\]

\[
\text{REF}
\]


\[
\text{Asymmetric} \quad \text{Asymmetric} \quad \text{Asymmetric} \quad \text{Asymmetric} \quad \text{Asymmetric} \quad \text{Asymmetric}
\]

\[
\text{Weaker} \quad \text{Weak} \quad \text{RF} \quad \text{field} \quad \text{Stronger} \quad \text{RF} \quad \text{field}
\]

\[
\text{at different horizontal locations and feel the horizontally asymmetric}
\]

\[
\text{RF power density.}
\]

\[
\eta \quad \text{is directional and}
\]

\[
|\text{rf}| \quad \text{is not symmetric in} \eta
\]
$I_{\text{eff}} = 0.113992$
Conclusio

- The presence of alpha particles in a fusion-reactor plasma may make a significant difference in FWCD.

- $\alpha$-ICRCD can yield a broad current (due to Doppler broadening).

- An improper choice of the wave frequency and wave number can yield a destructive synergism between $j_e^{RF}$ and $j_\alpha^{RF}$.

$$j_e^{RF} > 0 \quad \text{but} \quad j_\alpha^{RF} < 0$$

- A proper choice can produce a constructive synergism

$$\begin{align*}
  j_e^{RF} > 0 & \quad \text{at the center} \\
  j_\alpha^{RF} > 0 & \quad \text{at the midradius.}
\end{align*}$$

  We may not need a second current drive scheme for a revered-shear equilibrium.

- For a current profile control of the bootstrap-dominant discharge, we may not need a high efficiency.

  $$\gamma \sim 0.1 \times 10^{20} \text{AW}^{-1}\text{m}^{-2} \text{may be adequate.}$$

- Experimental verification of the $\alpha$-ICRCD mechanism in a present-day tokamak may be possible by using ICRH interaction with NBICD. [Radial transport of beam ions by ICRH (Chang, et. al. 1991) has been reported in Textor, Phys. Rev. Lett. (1994)]
<W v c

CO
I

01

FtG 1. Companion of ibc present analytic solution, f^e, with a numerical
solution, (a) Twcwiimcnsional contour plots of tail ton distributiony^c in
the velocity space with constant />» assumption 1 he pan'lcl anJ perpendicular particle speed is normalized to the ion-electron cntical slowingdown speed to easily represent the region of validity given fc* the appro*!nation <a{. + » J « ) / » J > I . The dotted arcle shows the locus of E,
Hydrofen annonly heattnf in a deuterium plasma is considered with
T^/E, — 100. (b) Numerical solution of Ref. I The contour structu. e h
sharper near the resonance pitch angle because of the smaller A, numbers
taken, and hence because of the sharp variance of 0 , in pitch angle near the
resonance pitch argl^.

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MHD stability of highly shaped plasmas in large-aspect-ratio tokamaks

J. K. Lee (Postech, Korea)
MHD STABILITY OF HIGHLY SHAPED HIGH-BETA TOKAMAK PLASMAS IN LARGE ASPECT RATIO LIMITS

1995.3.29

Jae Koo Lee
POSTECH

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Figure 5.1 The KT-2 Poloidal field magnet system configuration.
3-D NONLINEAR RESISTIVE MHD SIMULATIONS FOR DIII-D

- CART-I: Strauss equations with large aspect ratio
- CART-II: Izzo and Monticello equations with more toroidal effects

FEATURES:
- Stability Using Theoretical Equilibria for Highly Noncircular Diverted Tokamaks

3-D RESISTIVE MHD CODES

<table>
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<th>CARTESIAN</th>
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Advantages of CART (Noncircular, Diverted, Scrape-Off Layer, Baffles, EDDT)

Disadvantages of CART (Efficient, Physics Terms)

\[
\begin{align*}
\frac{d\alpha}{dt} &= \frac{\partial \Psi}{\partial \alpha} + \frac{1}{\alpha} \frac{\partial \Psi}{\partial \beta} \frac{d\beta}{dt} \\
\frac{d\beta}{dt} &= \frac{\partial \Psi}{\partial \beta} + \frac{1}{\beta} \frac{\partial \Psi}{\partial \alpha} \frac{d\alpha}{dt}
\end{align*}
\]

\[
\begin{align*}
\frac{d\alpha}{dt} &= \frac{B_0}{\rho} \frac{\partial}{\partial \alpha} \left( \frac{1}{\sin^2 \theta} \frac{\partial}{\partial \alpha} \right) \\
\frac{d\beta}{dt} &= \frac{B_0}{\rho} \frac{\partial}{\partial \beta} \left( \frac{1}{\sin^2 \theta} \frac{\partial}{\partial \beta} \right)
\end{align*}
\]

**Grid**

- Coarse Grid (32x32)
- Large Grid (64x64, 128x128)

**Wall Distance**

- \( d = 0.1 \text{ m} \)
- \( d = 0.5 \text{ m} \)
- \( d = 1.0 \text{ m} \)
- \( d = 1.5 \text{ m} \)

**Convergence**

- \( 32 \times 128 \)
- \( 64 \times 256 \)

**Equation**

- \( \Delta x = \Delta y = \Delta z = 0.005 \text{ m} \)

**Diagram**

- Visualization of field lines and plasma behavior in a tokamak environment.
COMPARISON WITH HIB (High-β, Circular Equilibrium)

COMPARISON WITH GATO (High-β, DIII-D Shot 57456)
NONCIRCULARITY EFFECTS ON MAGNETIC PERTURBATION
(Based on DIII-D Parameters)—Theory
(1) Triangularity (θ) reduces the nonlinear amplitudes
(2) Indentation (ι) increases the nonlinear amplitudes
(3) Elongation (κ) increases the nonlinear amplitudes

DIII-D MACHINE PARAMETERS

R = 1.69 m
a = 0.67 m
a/R = 0.40
b/a = 2.0
B = 2.1 T
I = 3.0 MA (limiter)
  2.5 MA (divertor)
Neutral beams:
  P = 12 MW
  E = 80 keV
ECH heating:
  P = 1.3 MW
  f = 60 GHz
IBW heating
  f = 30-80 MHz
  (now testing)
Ideal MHD kink instabilities for highly elongated tokamaks were analyzed using GATO. The $n=0$ results for the $K=6$ EST were consistent with the CAK results. The axisymmetric mode with $n=0$ is also a shifting mode for the EST. The position of the wall necessary to stabilize the $n=0$ kink mode for an EST with $K=6$, is greater than 0.3 $a$ except at low $\beta$, and peaked current profile. We were not able to stabilize the $n=0$ kink mode for a $K=6$ tokamak with an approximate racetrack plasma shape. The racetrack results are in agreement with other results.

Fig. 3. $n=0$ shifting mode for a $K=6$ racetrack equilibrium.

Fig. 4. Perturbed potential for the $n=0$ and $n=1$ modes with four cases.

Fig. 5. Typical mode structure corresponding to the $D=0$ case of Fig. 4. In (a), the perturbed magnetic flux contours are shown for the $n=0$ and $n=1$ modes with four different types of ideal plasmas. In (b), the perturbed flow potential contours are shown corresponding to (a).

$E=\beta, 2.5A, 1.5T, \beta_p=0.09, \beta_I=8.8\%$ racetrack

JK Lee, Nucl. Fusion 198.
• COMPARISON OF FLUCTUATION
  - DIII-D Shot 55791 (2400 ms): (Base Equilibrium)
  - Exact Vacuum Vessel Shape Needed for Good Comparison
  - Resistivity/Viscosity Models Important (Plasma and Vacuum Region)
  - Excellent Agreement in Outboard Side (Both Phases and Amplitudes)
  - Inboard Side Sensitive to Nonlinear Effects
MAGNETIC ISLAND EVOLUTION

LARGE MAGNETIC ISLANDS IN THE EDGE REGION MAY LEAD TO STOCHASTIC MAGNETIC FIELDS, MODE-LOCK, AND DISRUPTION.

- DIII-D 55791 t=2400 ms ($B_i/B_E = 0.72\%$)
  - 2/1 Island Size: 9 cm at Outboard Side
- Later Time ($B_i/B_E = 4.7\%$)
  - Large 2/1 Island with Smaller 3/2 and 4/3 Islands
  - Stochastic Edge Magnetic Field, Mode-Locking, and Disruption may Follow.
MODE CASCADE SIMULATION

MOST UNSTABLE TOROIDAL MODE NUMBER \( n \) CHANGING FROM 3 TO 2 TO 1 IN BETA-SEQUENCE MAY BE RELATED TO RESISTIVE GLOBAL MODES

(A) Free Boundary CART-II Results without FLR Effect
(B) Fixed Boundary CART-II Results with FLR Effect

Reality Could be Between (A) and (B)
- ITER APPLICATION (16 MA, 5 T, $\beta = 4\%$)

$\delta_0 = 1.08$
$\rho_0 = 5$
$I = 16.4\, \text{MA}$
$B = 5\, \text{T}$
$\beta = 4\%$ ($\beta_p = 1\%$)
1.2 KT-2 physics requirements for design

- Based on known MHD stability principles.
- Follow ITER physics rules.

(1) Plasma beta limit

- Troyon limit.
- Ideal ballooning modes are widely believed to be behind this limit
  \[ \beta \leq \beta_T = C_T (I_p/aB_0) \% \] with \( C_T = 3.0 \).

(2) Plasma current limit

- Stability conditions of the ideal kink modes set an upper limit
  on plasma current, i.e. a limit on the safety factor \( q \) at the edge.
- For KT-2, \( q^* \approx 2.5 \) is chosen:

  \[
  q_m = \frac{B_0 R_0}{2a} \left( \int \frac{ds}{R^2 B_s} \right) \geq q^* \frac{(1.17 - 0.65 \frac{a_r}{R})}{(1 - (\frac{a_r}{R})^2)} > 2.5.
  \]

  \( q_m \): safety factor at 95% magnetic flux surface

  \[ q^* = \frac{5a_r^2 B_0}{R_0 a_s} \left( 1 + 28 \frac{a_r}{R} - 1.28 \left( \frac{a_r}{R} \right)^2 \right) \]

(3) Density limit

- Murakami-Hugill limit, which is closely related to
  radiation cooling in tokamaks fuelled by gas-puffing:

  \[ n \leq n_{MH} = \frac{2B_0}{R a_s} \left( 10^{20} m^{-3} \right). \]
In Figures 3 and 4, PF coil current waveforms and snapshot equilibrium profiles of q, pressure, and plasma currents are shown for the 5MW baseline case. Need for accurate profile control is evident from hollow current profiles and negative shear in the high-\beta discharge core with the heating. These snapshot calculations will soon be replaced with a detailed MHD stability calculations. For RF heated discharges, analyses of the deposition and thermalization process as well as bootstrap current should be included in the baseline analyses. This process will be our next step, centered around the tokamak simulation code TSC with other codes with specialized applications, for example, FWCD full wave simulation code[7], and so on.

*Research supported by Korea Ministry of Science and Technology and KEPCO.

SUMMARY & CONCLUSION

- MHD ANALYSES VIA CART FOR LOW-n MHD MODES (AXISYMMETRIC, KINK, TEARING)

- EFFECTS OF SHAPING FACTORS (ELONGATION, INDENTATION, TRIANGULARITY)

- EFFECTS OF HIGH-BETA, HIGH-CURRENT RESISTIVITY, WALL SHAPE

- COMPARISON WITH EXPERIMENTS

- FLEXIBLE & WIDE APPLICATIONS

- COMPLEMENTARY TOOLS
Simulation characteristic of the operation modes in KT-2 tokamak

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Simulation Characteristics of the Operation Modes in the KT-2 Tokamak

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Abstract

To develop operation scenarios of KT-2 tokamak, 3 operation modes (OH, high $\beta$ and high bootstrap) deduced from zero dimensional steady-state power balance are examined with TSC (Tokamak Simulation Code) time-dependent transport code. Plasma profiles are evaluated self-consistently during simulations and plasma shapes are maintained by feedback control on PF coil currents. Simulations determine the profiles of plasma pressure, density and bootstrap current in the current flattop, and provide guidances for KT-2 conceptual design such as likely plasma parameters and design data of PF coil system. Simulations show operation modes which are typical of KT-2 expected discharges are compatible with the KT-2 PF system design specifications [1].

I. Introduction

KT-2 device concept [1] includes such new physics issues as high bootstrap current, non-inductive current drive, possibility of steady-state tokamak operation, and the second stability regimes, none of which is supported yet by solid experimental evidences. However, the baseline performance and the corresponding operation regimes of the KT-2 tokamak are derived within the framework of existing tokamak design principles such as used in the ITER design [2]. The design principle for the baseline KT-2 tokamak is, in fact, inclusive of all the operation regimes in each of which the KT-2 tokamak is conceptualized, wherever possible.

In this study, performance and operation regimes of KT-2 tokamak are estimated based on zero-dimensional power balance. From this estimation, 3 operation modes (OH, high $\beta$, high bootstrap) which are typical of operation modes expected in KT-2 tokamak are deduced and examined with the TSC transport code [3]. Time-dependent simulation is a powerful tool in predicting performance of new tokamak since it provides self-consistent profiles of densities, temperatures, heating and current drive. TSC evolves MHD equations describing transport time-scale evolution of axisymmetric magnetized tokamak plasma. Plasma profiles are evaluated self-consistently and plasma shapes are maintained by feedback control on PF coil currents.

MHD stability of KT-2 equilibria is investigated through $\beta$, density limit and $(l_e, q)$ diagram during simulation. Compatibility with the KT-2 PF system design specification is investigated through volt-second accounting and PF coil current waveform which is an output from the simulation.
II. KT-2 Performance and the Operation Regime

The basic specifications of the KT-2 tokamak given in Table 1 is so chosen as to minimize the machine size and input power requirement, while satisfying the requirements from its research programs as described in the concept definition[1] as well as requirements from plasma physics and device engineering standpoints.

II.1 Zero-dimensional Power Balance

KT-2 plasma performance is estimated by solving zero dimensional power balance equation. In the thermal equilibrium states during the discharge, plasma heating power is balanced by heat loss through conduction and radiation:

\[ P_{\text{con}} + P_{\text{rad}} = P_{\text{OH}} + P_{\text{aux}}. \]  \hspace{1cm} (1)

where \( P_{\text{OH}} \) is the auxilliary heating power and \( P_{\text{OH}} \) is the ohmic heating power which can be expressed as, excluding bootstrap currents, following:

\[ P_{\text{OH}} = (1 - f_{\text{b,c}}) R_p^2 R_s \times 10^{-6} \text{vol} \quad (\text{MW/m}^3). \]  \hspace{1cm} (2)

Here \( f_{\text{b,c}} \) is the bootstrap current fraction \( l_{\text{b,c}} \), \( \text{vol} \) is the plasma volume. The radiation loss, \( P_{\text{rad}} \) can be broken down into bremsstrahlung and synchrotron radiations:

\[ P_{\text{rad}} = P_{\text{brem}} + P_{\text{sync}} \quad (\text{MW/m}^3). \]  \hspace{1cm} (3)

Conduction loss, \( P_{\text{con}} \) consists of electron and ion contributions:

\[ P_{\text{con}} = P_{\text{tr},i} + P_{\text{tr},e}. \]  \hspace{1cm} (4)

where

\[ P_{\text{tr},i} = 2.403 \times 10^{-22} \frac{n_i T_i}{\tau_i} \quad (\text{MW/m}^3), \]  \hspace{1cm} (5)

\[ P_{\text{tr},e} = 2.403 \times 10^{-22} \frac{n_e T_e}{\tau_e} \quad (\text{MW/m}^3). \]  \hspace{1cm} (6)

For a prediction of the energy confinement time \( \tau_E \), the so-called ITER L-mode power scaling[2] is adopted:

\[ \tau_E = \tau_i = \tau_{\text{E}}. \]  \hspace{1cm} (7)

where

\[ \tau_{\text{E}} = H_f 0.048 l_4^{0.85} R_0^{1.2} a^{0.3} X_0^{0.5} B_0^{0.1} D_0^{0.2} A^{0.5} (P_{\text{con}} \cdot \text{vol})^{-0.5} \]  \hspace{1cm} (8)

Confinement improvement possibility represented by recently observed new tokamak discharges such as H mode[4], pellet mode[5], improved L mode[6], and VH mode[7] are incorporated into eq.(8) as the confinement improvement factor \( H_f \).

II.2 Physics Rules
Plasma beta limit
We use a beta-limit known as Troyon limit[8],
\[ \beta \leq \beta_T = C_T \left( I_p / a B_0 \right) \% \] (9)
where \( C_T \) is Troyon coefficient.

Plasma current limit
Plasma current is limited by stability conditions of the ideal kink modes, i.e. a limit on the safety factor \( q \) at the edge. For KT-2, \( q \approx 2.5 \) is chosen;
\[ q_{95} \approx q_\star \frac{\left(1.17 - 0.65 \frac{a}{R} \right)}{\left(1 - \left(\frac{a}{R}\right)^2\right)^2} > 2.5, \] (10)
where \( q_{95} \) is the safety factor at magnetic flux surface with 95% of the flux at the separatrix, and \( q_\star \) is given as following:
\[ q_\star = \frac{5a^2 B_0}{R_0 I_p} \frac{1 + q_{95}^2 (1 + 2q_{95}^2 - 1.2q_{95}^3)}{2}. \] (11)

Density limit
For density limit, Murakami-Hugill limit[10], which is closely related to radiation cooling in tokamaks fuelled by gas-puffing is adopted:
\[ n \leq n_{MIL} = \frac{2B_0}{R_0 q_\star} \times \left(10^{20} \text{ m}^{-3}\right). \] (12)

Bootstrap current
For the bootstrap current fraction, the following experimental observation is employed[2]:
\[ I_{bs} / I_p = C_{bs} \left( q_{95} / q_o \right). \] (13)
where
\[ C_{bs} = 1.32 - 0.235 \left( \frac{q_{95}}{q_o} \right) + 0.0185 \left( \frac{q_{95}}{q_o} \right)^2 \] (14)
\[ \beta_{pa} = \beta \left( B_0 \frac{2\pi <a>}{\mu_0 I_p} \right)^2. \] (15)
Here \( q_o \) and \( <a> \) are the safety factor on magnetic axis and the "effective" plasma minor radius, respectively

II.3 Operating Regimes
The plasma performance of KT-2 is obtained with the pressure and current profiles assumed as
where \( a_r \) is a profile shape control parameter.

Figure 1 summaries plasma performance and operating regimes of KT-2 for (a) \( H_f = 2.0 \) and (b) 3.0 as function of the heating power. KT-2 machine and plasma parameters used are: \( A = 5.6 \), \( R_o = 1.4 \) m, \( B_c = 3.0 \) T, \( I_p = 500 \) kA, \( \alpha_o = 0.5 \), \( \alpha_r = \alpha_i = 1.0 \). For the \( H_f = 2.0 \) case, about 10 MW heating power is required to reach the beta limit. However, this heating power drops to about 5 MW for \( H_f = 3.0 \).

To study the bootstrap current effects in FWCD at KT-2, it is necessary to control the plasma current and density, toroidal field, and the aspect ratio accordingly. In Figure 2, the bootstrap current fraction expected for the heating power of 1 MW and 5 MW is indicated as function of \( q_{98} \), i.e., the plasma current when \( H_f = 3.0 \). The bootstrap current fraction increases as the plasma current is lowered and as the toroidal field is lowered. The bootstrap current fraction bigger than 80% will be achievable with 5 MW heating.

### III. KT-2 Discharge Simulation

From Fig. 1, operation modes of KT-2 can be deduced which are typical of KT-2 tokamak discharges; (1) OH mode, (2) high \( \beta \) mode, (3) high bootstrap mode. In this section TSC time-dependent transport simulations of 3 operation modes are presented to determine the profiles of plasma pressure, density and bootstrap current profiles in the current flattop, and to demonstrate the compatibility with KT-2 PF system design specifications through MHD stability consideration, volt second accounting and PF coil current waveform from plasma shape control.

#### III.1 Transport Equations

For transport equations, we follow the original paper describing TSC model in Ref. 3. In cylindrical coordinate, we express magnetic field as

\[
B = \nabla \phi \times \nabla \psi + g \nabla \phi, \tag{17}
\]

where \( \phi \) is a toroidal angle, \( \psi \) is a poloidal flux and \( g \) is a toroidal field function, and plasma momentum density as

\[
m = \nabla \phi \times \nabla A + \omega \nabla \phi + \nabla \Omega, \tag{18}
\]

where \( A \) is a stream function, \( \omega \) is it's toroidal component and \( \Omega \) is a potential.

The plasma force balance equation is given by

\[
\frac{\partial m}{\partial t} + F_\phi(m) = j \times B - \nabla p, \tag{19}
\]

The scalar equations are obtained by operating \( \{ \nabla \cdot \} \), \( \{ \nabla \phi \cdot \nabla \times \} \) and \( \{ \nabla \phi \cdot \} \) to Eq.(19).
\[
\frac{\partial}{\partial t} \nabla^2 \Omega + \nabla \left[ \frac{\Delta^* \Psi}{\mu_0 x^2} \nabla \Psi + \frac{\mathbf{E}}{\mu_0 x^2} \nabla \mathbf{g} + \nabla \mathbf{v} - \nu_2 \nabla (\nabla^2 \Omega) \right] = 0, \\
\frac{\partial}{\partial t} \Delta^* \mathbf{A} + x^2 \nabla \left[ \frac{\Delta^* \Psi}{\mu_0 x^2} \nabla \Psi \nabla \mathbf{A} + \frac{\mathbf{E}}{\mu_0 x^2} \nabla \mathbf{g} \nabla \mathbf{v} - \frac{V_1}{x^2} \nabla (\nabla^* \mathbf{A}) \right] = 0, \\
\frac{\partial}{\partial t} \omega + \mu_0^{-1} \nabla \phi \times \nabla g \cdot \nabla \Psi - \nu_1 \Delta^* \omega = 0, \\
\end{align}
where $\Delta^* = x^2 \nabla \cdot x^2 \nabla$ is Grad-Shafranov operator. In steady state Eqs. (20) - (22) reduce to Grad-Shafranov equilibrium equation:
\[
\nabla^* \Psi + \mu_0 x^2 \frac{d}{d \Psi} \rho(\Psi) + \frac{1}{2} \frac{d}{d \Psi} g^2(\Psi) = 0,
\]
Equations for the poloidal flux and toroidal field function are obtained from Faraday's Law and Ohm's Law.
\[
\frac{\partial}{\partial t} \Psi + \frac{1}{\rho \Phi}(\nabla \phi \times \nabla \mathbf{A} \cdot \nabla \Psi + \nabla \Omega \cdot \nabla \Psi) = x^2 \nabla \phi \cdot \mathbf{R}, \\
\frac{\partial}{\partial t} g + x^2 \nabla \cdot \left[ \frac{E}{\rho_0 x^2} (\nabla \phi \times \nabla \mathbf{A} + \nabla \Omega) - \frac{Q}{\rho_0 x^2} \nabla \phi \times \Psi - \nabla \phi \times \mathbf{R} \right]
\]
where $\mathbf{R}$ contains the nonideal terms and $\rho = \rho_0 \mathbf{M}_i$ is a mass density.

TSC evolves thermodynamic variables with respect to magnetic surface containing a fixed toroidal flux. Toroidal flux $\mathbf{\Phi}$ within a constant $\Psi$ surface is defined as:
\[
\mathbf{\Phi} = \frac{1}{2\pi} \int_{\Psi \subset \Psi_e} d\tau B \cdot \nabla \phi = \int_{\Psi \subset \Psi_e} d\mathbf{x} d\tau \mathbf{g}(\mathbf{x}, \tau),
\]
The one-dimensional equations for differential electron number density $N_e = n_e \frac{\partial V}{\partial \mathbf{\Phi}}$, differential total entropy density $\sigma = \rho(\frac{\partial V}{\partial \mathbf{\Phi}})^{5/3}$, and differential electron entropy density $\sigma_e = \rho_e(\frac{\partial V}{\partial \mathbf{\Phi}})^{5/3}$ are
\[
\frac{\partial}{\partial t} N_e = -\frac{\partial}{\partial \mathbf{\Phi}} (N_e \Gamma) + S_N \\
\frac{\partial}{\partial t} \sigma = \frac{2}{3} (\frac{\partial V}{\partial \mathbf{\Phi}})^{2/3} \left[ V_e \frac{\partial K}{\partial \mathbf{\Phi}} - \frac{\partial}{\partial \mathbf{\Phi}} (Q_e + Q_e) + \frac{\partial V}{\partial \mathbf{\Phi}} (S_e + S_e - R_e) \right].
\]
\[
\frac{\partial}{\partial t} \sigma_e = \frac{2}{3} \left( N_s - M \right) + \frac{1}{2} \frac{\partial K}{\partial \Phi} \left( Q_1 + Q_2 \right) + \frac{\partial V}{\partial \Phi} \left( S + S_i - R_e \right)
\]  

(29)

In Eqs. (27)-(28), the $S_1$, $S_2$, $S_3$ are external sources of particles, electron and ion energy and $R_e$ is energy loss due to radiation.

III.2 Physics modelling used in simulation

The particle flux and electron and ion heat fluxes are defined as [10]

\[
\Gamma = 2\pi d \left[ \frac{q^2 \cdot \nabla \phi}{\nabla \phi} \right], \\
Q_i = \frac{\partial V}{\partial \phi} \left[ \frac{q^2 \cdot \nabla \phi}{\nabla \phi} \right], \\
Q_e = \frac{\partial V}{\partial \phi} \left[ \frac{q^2 \cdot \nabla \phi}{\nabla \phi} \right], \\
\]

(30-32)

In Eqs. (30)-(32), random heat fluxes are of the form

\[
\langle q_i \cdot \nabla \phi \rangle = -\chi_i \left[ \nabla \phi \right] \left( n_e \frac{\partial T_e}{\partial \phi} \right), \\
\langle q_e \cdot \nabla \phi \rangle = -\chi_e \left[ \nabla \phi \right] \left( n_e \frac{\partial T_e}{\partial \phi} \right),
\]

(33-34)

and the electron and ion thermal conductivities are modeled as

\[
\chi_e = \chi_{TEM} = \chi_{TEM} = \frac{1}{2} \left( 1.25 \times 10^2 \right) \frac{a}{\nabla \phi} \left( xB_T \right)^{0.3} Z^{0.2} (1 + \frac{\sigma_H}{4}), \\
\chi_e = \frac{1}{2} \left( 1.25 \times 10^2 \right) \frac{a}{\nabla \phi} \left( xB_T \right)^{0.3} Z^{0.2} (1 + \frac{\sigma_H}{4}),
\]

(35-36)

and the density profile is taken to be of the form

\[
n_e(\Phi, t) = n_0^2(\Phi) \left[ 1 - \Phi^{a(0)} \right] + n_b(\Phi),
\]

(42)

Here $P(\Phi)$ is the total heating power minus the total radiated power inside the surface $\Phi$.

The factor $f_m$ in Eq.(36) is used to account for the time averaged effect of sawtooth instability inside $q=1$ surface. Thus thermal conductivity is enhanced inside the $q=1$ surface according to the prescription:

\[
f_m = 1 \quad \text{for } q > 1, \quad f_m = \frac{1}{a_{124}} \quad \text{for } q < 1
\]

(40-41)

The density profile is taken to be of the form

\[
n_e(\Phi, t) = n_0^2(\Phi) \left[ 1 - \Phi^{a(0)} \right] + n_b(\Phi),
\]

(42)
where $\Phi$ is the normalized poloidal flux and $n_b(t)$ is the density at plasma boundary. This model for density evolution is adopted since a satisfactory dynamic transport model for the density profile is existed and it would be difficult to infer the actual source terms in the presence of both gas fueling and recycling.

For bootstrap current model, we use Hirshman’s expression[11] which is valid in neo-classical banana regime is used:

$$I_{bs} = \int d\Psi \frac{\partial \Psi}{\partial \Psi} \langle J \cdot \vec{B} \rangle_{bs}$$

with

$$\langle J \cdot \vec{B} \rangle_{bs} = RB_c \phi_c(\Psi) (L_{31} + \frac{T_i}{Z_i T_e} (A_1 + a_r A_2) + L_{32} A_2),$$

$$L_{31} = x(0.754 + 2.21Z, + Z_i^2 + x(0.348 + 1.243Z_i + Z_i^2)) / D(x)$$

$$L_{32} = -x(0.884 + 2.074Z_i) / D(x)$$

$$A_1 = \frac{d\phi_c}{\rho d\Psi}$$

$$A_2 = \frac{dT_i}{T_d d\Psi}$$

$$a_r = \frac{1.172}{1 + 0.462x}$$

$$D = 1.414Z_i + Z_i^2 + x(0.754 + 2.657Z_i + Z_i^2) + x^2(0.348 + 1.243Z_i + Z_i^2)$$

$$x = \frac{f_T}{1 - f_T}$$

In Eq. (45), $f_T$ is trapped particle fraction.

Neoclassical corrections to resistivity are used as in Ref. 12 and the effect of sawtooth instability is taken into account by enhancing resistivity inside $q=1$ surface:

$$\eta_1 = \eta_{nc}$$

for $q \geq 1$ (46)

$$\eta_1 = a_{120} \eta_{nc} + (1 - a_{120}) \eta_{nc}$$

for $q < 1$ (47)

III.3 The OH mode

In the OH operation mode, the confinement characteristics of a LAR tokamak will be studied with operation parameters spanning the whole designed values.

As shown in Fig.3 toroidal magnetic field is maintained 3.0 T for the entire simulation and plasma current is ramped from 20 kA at $t=0.0$ sec to the flat-top value of 500 kA at 0.5 sec. The plasma $\beta$ is expected to be 0.5 % if energy confinement is improved ($H_t > 2$ in Sec.II). We used free parameters, $a_{121}=0.06$ and $a_{122}=0.4$ in Eqs.(37) and (38) to give improved OH
confinement.

Fig. 4 shows snapshots of plasma-vacuum interface at various times during plasma evolution. The plasma is grown from an inboard limited circular plasma at $t=0.0$ sec, becomes diverted at $t\sim 0.4$ sec, and reaches a elongation of $\kappa=1.8$ and triangularity of $\delta=0.6$ at start of flattop. The plasma shape (minor radius = 25 cm, elongation $\kappa=1.8$ and triangularity $\delta=0.6$) is maintained by feedback control on PF coil currents and PF coil currents are output from the simulation.

The plasma density profile assumed for the simulation is shown in Fig. 5. The central electron density is programmed to increase linearly with time to the flattop value of $\langle n \rangle \equiv 1.0 \times 10^{20}$ m$^{-3}$ with peak to average value, $\frac{n_0}{\langle n \rangle} = 0.6$. Ratio of average density to Murakami limit is about 0.7 and effective charge $Z_{\text{eff}}$ is set to 2.0.

Flattop value of central electron and ion temperatures are 1.2 keV and 1.15 keV respectively as shown in Fig. 6 and energy confinement time is calculated as 80 mesecond.

Fig. 7 shows flattop value of volume averaged $\beta \sim 0.5 \%$, plasma internal inductance $l_i \sim 1.0$, and poloidal beta $\sim 0.55$. It has been observed on JET that, for a given plasma equilibrium, plasma internal inductance cannot be exceeded due to a major disruption. This condition is expressed by plotting $l_i/2$ as a function of $q_{\text{eq}}$, since these quantities can be related to MHD stability of kink and tearing modes. A trajectory of KT-2 equilibria in $l_i-q$ space, in which shaded region represents the region of instability, remains stable region during entire current ramp up and flattop period.

Fig. 8 shows the evolution of the central and edge safety factor, and of the radius of $q=1$ surface. $q_{\text{eq}}$ reaches about 2.6 and $q=1$ radius is frozen at about 14 cm in the current flattop.

PF coil currents waveforms obtained as an output from simulation are displayed in Fig. 9. The PF system provide a total of 2.7 V $\cdot$ s during the current ramp up and maintain the equilibrium during current ramp up and flattop within its design specifications.

### III.4 High $\beta$ Mode

According to the performance estimations indicated in Fig. 1, it is possible with FWCD power of 5 MW to achieve a discharge with $\beta$ close to Troyon beta limit if $H_t > 2$ is achieved.

Fig. 10 shows schematic of high $\beta$ mode simulation. As in the case of OH mode, toroidal magnetic field is maintained 3.0 T for the entire simulation and plasma current is ramped from 20 kA at $t=0.0$ sec to the flattop value of 500 kA at 0.5 sec. At 0.85 sec 5 MW auxiliary is supplied to plasma and as a consequence, plasma $\beta$ is expected to reach 1.9 % and $\beta_n (=\beta/\beta_f)$ is 2.9. The free parameters $a_{12}=0.055$, $a_{22}=0.4$ in Eqs.(37) and (38) are used to give improved confinement. Heat deposition profile is assumed as

$$S_{\text{ICRH}}(\psi) = \frac{d^2 (\psi - \psi_0)(1 - \psi / \psi_0)^d \psi_0^{a_2}}{(\psi - \psi_0)^2 + d^2},$$

where $\psi$ is the normalized poloidal flux.
When current profile reaches its steady state after heating is supplied, poloidal flux contour and current profile as a function of toroidal flux are shown in Fig. 11. Bootstrap current is seen to be 45% of total toroidal plasma current.

The plasma density profile assumed for the simulation is same as OH mode case and flattop value of central electron and ion temperatures are 6.0 keV and 3.2 keV respectively as shown in Fig. 12 and energy confinement time reduces to 50 msec due to heating degradation.

Fig. 13 shows flattop value of volume averaged $\beta \sim 1.9\%$, plasma internal inductance $l_i \sim 1.0$, and poloidal beta $\sim 2.1$. Rise time for plasma $\beta$ is seen to be about 0.5 sec. A trajectory of high $\beta$ KT-2 equilibria in $l_i$-$q$ space remains stable region during current flattop period.

Fig. 14 shows the evolution of the central and edge safety factor, and of the radius of q=1 surface. q$_s$ reaches about 2.7 and q=1 radius is frozen at about 13 cm after heating is supplied.

PF coil currents waveforms obtained as an output from simulation are displayed in Fig. 15, which shows the PF system can provide the proper fields to maintain the high $\beta$ equilibrium within it's design specifications.

### III.5 The High Bootstrap Mode

The performance estimations indicated in Figs. 1 and 2 show that it is possible with FWCD power of 5 MW to achieve a high-$\beta$, high-bootstrap discharges with bootstrap fraction of more than 80%, if $H_\theta$ of about 2 is achieved and if profiles are properly controlled to sustain stability. In such bootstrap-dominant discharges, the "advanced tokamak" physics issues could be investigated including 100% non-inductive current drive and steady-state operation, and the second stability.

As shown in Fig. 16, the toroidal magnetic field is maintained 2.0 T for the entire simulation and plasma current is ramped from 20 kA at $t=0.0$ sec to the flattop value of 300 kA at 0.5 sec. At 0.6 sec 5 MW auxiliary heating is supplied to plasma and as a consequence, plasma $\beta$ is expected to reach 2.4 % and $\beta_0$=$l_i/a_B$ is 4.0.

The central electron density is programmed to increase linearly with time to the flattop value of $\langle n \rangle \equiv 5.5 \times 10^{19} \text{ m}^{-3}$ with peak to average value, $n_0 = 0.6$. Ratio of average density to Murakami limit $\langle n \rangle \equiv 0.6n_{MH}$ and effective charge $Z_{eff}$ is set to 2.0.

Flattop value of central electron and ion temperatures are 9.5 keV and 2.0 keV respectively as shown in Fig. 17. Flattop value of volume averaged $\beta \sim 2.4\%$, plasma internal inductance $l_i \sim 0.80$ and poloidal beta $\sim 3.3$ as shown in Fig. 18. Rise time for plasma $\beta$ is about 0.5 sec. A trajectory of KT-2 equilibrium in $l_i$-$q$ space remains stable region during entire current ramp up and flattop period.

Fig. 19 shows the evolution of current profile and safety factors. Due to higher bootstrap current fraction, the current profile develops to a hollow current profile, edge safety factor q$_s$
reaches about 3.1 and $q=1$ radius reduces to 6 cm after heating is supplied. When current profile reaches its steady state after heating is supplied, poloidal flux contour and current profile as a function of toroidal flux are shown in Fig. 20. Bootstrap current is seen to be 80 % of total toroidal plasma current and 90 % when volume average density increases to $8 \times 10^{19}$ m$^{-3}$.

PF coil currents waveforms obtained as an output from simulation are displayed in Fig. 21, which shows the PF system can provide the proper fields to maintain the high $\beta$, high bootstrap equilibria within it’s design specifications.

IV. Summary

The performance of KT-2 tokamak is estimated based on zero-dimensional power balance. It is shown that if energy confinement improvement of $H_f$ of about 2 is achieved and if profiles are properly controlled to sustain stability, it is possible with FWCD power of 5 MW to achieve a high-$\beta$, high-bootstrap discharges with bootstrap fraction of more than 80%.

Profiles evolution of KT-2 operation modes (OH mode, 5 MW heating mode and 5 MW high bootstrap mode) is addressed using TSC time-dependent transport code and performance estimation obtained from zero-dimensional studies are confirmed by transport simulation. Simulations indicate that trajectories of equilibria remain in stable region during entire current ramp up and flattop period and PF system can provide proper equilibria fields for KT-2 operation modes.

In this study we have developed "conventional" operation scenarios for KT-2 tokamak, in which safety factor profile has a minimum at magnetic axis. To demonstrate the possibility of 100 % non-inductive current drive, steady-state and second stability operation for which KT-2 tokamak is conceptualized, development of advanced operation scenarios with high $\beta$ bootstrap-dominant discharge is necessary. Negative magnetic shear configuration is promising since it gives higher $\beta_n(=l_B/a_{Br})$, higher $q$ values while maintaining larger total plasma current and total current profile aligns well with bootstrap current profile. Dynamic evolution of configuration is necessary to create an ideal MHD stable path from OH to negative magnetic shear state and demonstrate the compatibility with the PF system. Modelling of non-inductive current drive to realize this scenario and MHD stability to ballooning mode and $n=1$ external kink mode are presently under study.

V. References

Table 1  Basic specification of KT-2 tokamak

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
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<tbody>
<tr>
<td>Major radius</td>
<td>R = 1.4 m</td>
</tr>
<tr>
<td>Minor radius</td>
<td>a = 0.25 m</td>
</tr>
<tr>
<td>Aspect ratio</td>
<td>A = 5.6</td>
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<tr>
<td>Elongation</td>
<td>κ = 1.8</td>
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<tr>
<td>Triangularity</td>
<td>δ &gt; 0.5</td>
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<tr>
<td>Toroidal field</td>
<td>Bₜ = 3 Tesla max. (water-cooled)</td>
</tr>
<tr>
<td>Number of magnets</td>
<td>N = 16</td>
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<tr>
<td>Ripple</td>
<td>δBₚ = 2% at separatrix</td>
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<tr>
<td>Plasma current</td>
<td>Iₚ = 500+ kA</td>
</tr>
<tr>
<td>Flux swing</td>
<td>φ = 9.2 Wb (bipolar) (air-core)</td>
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<tr>
<td>Current flat-top</td>
<td>t = 4.2 sec @ Iₚmax</td>
</tr>
<tr>
<td>Density</td>
<td>nₑ = (0.5 ~ 1.0) x 10³⁰ m⁻³</td>
</tr>
<tr>
<td>Electron/ion temperature</td>
<td>Tₑ(Tᵢ) = 1 keV</td>
</tr>
<tr>
<td>FWCD/ICRH</td>
<td>1 MW</td>
</tr>
<tr>
<td>ECRH</td>
<td>5 MW (by 2001)</td>
</tr>
<tr>
<td></td>
<td>1 MW (by 1999)</td>
</tr>
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</table>
Figure 1. KT-2 operating regimes as function of the heating power. (a) $H_f=2.0$, (b) $H_f=3.0$. 
Figure 2 Bootstrap current fraction as function of the edge safety factor when $H_f = 3.0$
Figure 3 Schematic of OH operation mode
Figure 4  Plasma-vacuum interface during simulation
Figure 5  Assumed electron density evolution
Figure 6 Evolution of electron and ion temperatures
Figure 7 Evolution of $\beta$ and internal inductance
Figure 8 Evolution of central and edge safety factor, and radius of q=1 surface
Figure 9  PF coil current waveform
Figure 10 Scematic of high $\beta$ operation mode
Figure 11 Contours of poloidal flux and current densities
Figure 12 Evolution of electron and ion temperatures
Figure 13  Evolution of $\beta$ and internal inductance
Figure 14 Evolution of central and edge safety factor, and radius of q=1 surface
Figure 15 PF coil current waveform
Figure 16 Scematic of high bootstrap operation mode
Figure 17  Evolution of electron and ion temperatures
Figure 18 Evolution of β and internal inductance
Figure 19 Evolution of central and edge safety factor, and radius of $q=1$ surface
Figure 20 Contours of poloidal flux and current densities.
(a) $<n_e>=5.5 \times 10^{19} \, \text{m}^3$, (b) $<n_e>=8.0 \times 10^{19} \, \text{m}^3$. 
Figure 21  PF coil current waveform
Engineering R&D issues for steady-state tokamak fusion reactors

R. Andreani (ENEA-Frascati, Italy)
Main features of a fusion reactor

1. Continuous operation (shutdown for maintenance and repair only).

2. Reactor availability of 70-80%.

3. Electrical Recirculating Power ≤15-20%.

4. No plasma disruptions or major power transient excursions.

5. Neutron fluence tolerable without replacement of the main structural components corresponding to a material damage of 150-200 dpa.

6. Full maintainability by Remote Handling.
Advanced Tokamak Physics

- $Q \propto n \tau E T \propto H^2 \cdot A^2 \cdot I(2-a); \ a = 0 + 0.5$;

  H enhancement factor of $\tau E$ from L-type to H-type confinement regimes

- A = $R/a$ aspect ratio
- I plasma current
- Bootstrap current fraction
  - $I_{BS} \propto A^{1/2} \beta N \cdot k \cdot q_{circ}$

- $\beta = \beta N \frac{I}{aB}$
- k elongation

- $P_{fus} \propto f^2 \cdot \beta N \cdot B^2 \cdot I^2 \cdot R \cdot k$

  f dilution factor ($\alpha$ particles and impurities)

- B toroidal magnetic field
- R major radius

**Lines of tendency for the reactor parameters**

- $<n> \geq 10^{20} m^{-3}$
- I $\sim 10$ MA
- $I_{BS} \sim 80\% \sim 2$ MA current drive; I
- q$\Psi \sim 4$;
- k $\sim 1.6$;
- $\beta N \geq 3$; $\rightarrow \beta = 2 + 3\%$;

  $P_{fusion} \propto \beta^2 B^4 \sim 2-4$ MW/m$^3$;

  Volume

  $P_{neutron} \sim 2$ MW/m$^2$;

  Wall surface
1. **Magnetic Structure.** Superconducting materials and magnets.

2. **First wall and blanket structural materials.** Neutron damage resistant.

3. **First wall and divertor. Protective and armour materials.** Heat transfer Halo currents. Disruptions.

4. **Heating and Current Drive.**

5. **Remote Handling.**

---

### 1. Magnetic structures. Superconducting materials and magnets.

- Use of superconductors essential to reduce electrical power.

  CICC Cable in conduit:
  - Mechanical resistance: $\sigma_{0.2} \sim 1000 \text{MPa}@4.2\text{K}$
  - Average current density: $\sim 20\text{A/mm}^2$ (ITER CS Conductor)

- Three main possible materials:
  - NbTi, Nb$_3$Sn, Nb$_3$Al;

<table>
<thead>
<tr>
<th></th>
<th>NbTi</th>
<th>Nb$_3$Sn</th>
<th>Nb$_3$Al</th>
</tr>
</thead>
<tbody>
<tr>
<td>$B_c(T)@4.2\text{K}$</td>
<td>11.5 (1)</td>
<td>22 (2)-25 (3)</td>
<td>29.5</td>
</tr>
<tr>
<td>$T_c (\text{K})$</td>
<td>10.2</td>
<td>18.3</td>
<td>16.9</td>
</tr>
<tr>
<td>$J_c (\text{A/m}^2)$ @7T,4.2K</td>
<td>800 (4), 600 (5)</td>
<td>500-700 @12T,4.2K</td>
<td>@12T,4.2K</td>
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<tr>
<td>$J (\text{A/m}^2)$ @7T,4.2K</td>
<td>300 @7T,4.2K</td>
<td>150@12T,4.5K</td>
<td>100@12T,4.5K</td>
</tr>
<tr>
<td>Hysteresis losses $\text{(mJ/cm}^3)$</td>
<td>250</td>
<td>600 (3)</td>
<td>100-200 (4)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>1000</td>
</tr>
</tbody>
</table>

**Nb$_3$Al still under development.**

(1) @1.8 K  $\Delta B_c = 1.5\text{T}$

(2) Binary Nb$_3$Al or Nb$_3$Sn

(3) Ternary Nb$_3$Sn [Nb 7.5%Ta, Nb1%Ti]

(4) Internal tin Nb$_3$Sn

(5) Bronze route. Nb$_3$Sn
2. Materials

- Neutron damage.
- Heat transfer.
- Mechanical resistance

- Austenitic steel (Stainless steel).
- Martensitic steel.
- Vanadium alloys. (Titanium alloys)
- Ceramic. SiC - SiC Composites.
### Structural materials

<table>
<thead>
<tr>
<th>Operating temper.</th>
<th>Heat transfer [MW/m²mm]</th>
<th>Neutron fluence</th>
<th>Ferromagnetism</th>
<th>Influence of the Environment</th>
<th>Environmental impact from neutron induced radioact.</th>
<th>Dose rate from maintenance operations</th>
<th>Industrial development relevant to fusion</th>
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</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Neutrons and radioact.</td>
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<td></td>
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<tr>
<td></td>
<td>&lt;450°C</td>
<td>2.3</td>
<td>yes</td>
<td>no</td>
<td>Not negligible</td>
<td>low</td>
<td>very good</td>
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<tr>
<td></td>
<td>&lt;500-580°C</td>
<td>5-6</td>
<td>yes</td>
<td>yes</td>
<td>high</td>
<td>high</td>
<td>high</td>
</tr>
<tr>
<td></td>
<td>&lt;700°C</td>
<td>9-11</td>
<td>yes</td>
<td>no</td>
<td>low</td>
<td>very low</td>
<td>very low</td>
</tr>
<tr>
<td>SiC-SiC composites</td>
<td>very high</td>
<td>15-20</td>
<td>yes</td>
<td>no</td>
<td>no porosity</td>
<td>no</td>
<td>modest with respect to fusion</td>
</tr>
</tbody>
</table>

**Austenitic st. steel**
- <450°C
- Neutron fluence: 2.3 dpa large swelling
- No water corrosion
- No embrittlement
- Very good

**Martensitic Steel**
- <500-580°C
- Neutron fluence: 5-6 dpa DBTT welding
- Yes
- No pick up of interstitial impurities (coatings)
- High
- Very good

**Vanadium alloys**
- <700°C
- Neutron fluence: 9-11 dpa
- No
- No pick up of interstitial impurities (coatings)
- Low
- Very modest

**SiC-SiC composites**
- Very high
- Neutron fluence: 15-20 dpa
- No
- No porosity
- Very low
- Very low
- Modest with respect to fusion

---

**Neutron damage**

1 MWY/m² of 14 MeV neutrons → 1.4 x 10^25 n/m² produces in stainless steel:

- ~11 dpa
daughter, 130-170 appm H₂, 500-600 appm H²
- Variation in chemical composition due to transmutation of elements.
- Swelling and creep.
- Radiation hardening and embrittlement.
- Increased sensitivity to stress corrosion cracking.
- Loss of fracture toughness.
- Thermal conductivity changes.
- Corrosion-radiation fatigue.
- Induced radioactivity.
- Nuclear heat production.
2. Heat Transfer

Thermal Stress Factor (Surface heat load capability)

$$ M = \frac{2 \sigma_y \cdot k (1 - \nu)}{\alpha E} = \Phi \cdot \tau $$

$E$  Young's Modulus
$\sigma_y$  Yield strength
$k$  Thermal conductivity
$\nu$  Poisson's modulus
$\alpha$  Thermal expansion coefficient
$\Phi$  Thermal flow
$\tau$  Wall thickness

$M = 2 - 3 \, \text{MW/m}^2 \cdot \text{mm}$  Austenitic steels
5 - 6  Martensitic - ferritic steel
9 - 11  Vanadium alloys
15 - 20  SiC/SiC fiber
~70  Copper
~130  Isotropic graphite.

First wall - Divertor Heat load

Power to the walls by radiation and convection for 2 MW/m$^2$ average neutron wall loading: 0.5 MW/m$^2$ average heat load.

1000 MW electrical $\rightarrow$ 3000 MW fusion power

600 MW heat power ($\alpha$ particles)

400 MW  200 MW

divertor walls

ave. wall loading: 0.1-0.2 MW/m$^2$

Unfortunately: large dishomogeneity in power density distribution

Reasonable assumptions (for the time being):
peak loads: 5 MW/m$^2$  first wall
15 MW/m$^2$  divertor plate
**Technical solution**

Copper alloys as heat sink and water cooling for >5MW/m².
150°C - 300°C: operation range for copper.

- Neutron embrittlement
- Mechanical softening
- Water cooling

Be or graphite

Typical example: Monobloc solution for divertor plates.

**Test results:**

- 18 MW/m² 5 s 300 cycles
- Then 15 MW/m² 5 s 3000 cycles

**Tmax** on graphite: 1800°C

**Disruptions**

Three main causes:

1. Density, q, β limits.

2. Vertical Displacement Events (VDE) with elongated plasmas.

3. MHD instabilities.

**Remedies**

1. Use operating parameters away from the limits.

2. Passive structure plus feedback control.

3. Look for "precursors" and use additional heating to modify density and current density profiles.

**Halo Currents**
4. Heating & Current Drive

In a reactor heating will be needed to:

◊ Provide enough power through the separatrix to access H mode confinement. $P_{\text{threshold}} \approx n_e B_T$ (ASDEX Upgrade).

◊ Raise the temperature to ignition in initial burn conditions and/or

◊ Support driven burn scenarios (finite Q)

◊ Drive on axis current (~0.5 MA) to provide the seed current for high bootstrap steady state operation and

◊ Provide current drive to maintain the nominal plasma current.

◊ If needed, impart angular momentum to the plasma to avoid locked modes.

Experimental situation

1. Lower Hybrid Waves
   ◊ Long pulse:
     TRIAM: ~1 hour 30 kA $\tilde{n}_e \sim 0.2 \times 10^{19} \text{m}^{-3}$
     TORE SUPRA: 1 min 1 MA $\tilde{n}_e \sim 2 \times 10^{19} \text{m}^{-3}$
   ◊ High current:
     JT60-U: 3.6 MA $\gamma = 0.34 \times 10^{20} \text{A m}^{-2} / \text{W}$
     TORE SUPRA: 1.6 MA $\gamma = 0.15 - 0.2 \times 10^{20} \text{A m}^{-2} / \text{W}$
     JET (combined LH and ICRH):
       1.8 MA $\gamma = 0.42 \times 10^{20} \text{A m}^{-2} / \text{W}$

2. Fast Wave CD
   ◊ DIII-D: 0.18 MA $\gamma \sim 0.02 \times 10^{20} \text{A m}^{-2} / \text{W}
   ◊ TORE SUPRA: 0.08 MA

3. NBCD
   ◊ TFTR: $\gamma \sim 0.05 \times 10^{20} \text{A m}^{-2} / \text{W}$
   ◊ TEXTOR (NB combined with IC):
     $\gamma \sim 0.052 \times 10^{20} \text{A m}^{-2} / \text{W}$

4. ECCD
   ◊ DIII-D: 0.1 MA $\gamma \sim 0.016 \times 10^{20} \text{A m}^{-2} / \text{W}$

In general, good agreement with theory.

$$\gamma = \frac{<n> R \omega_p}{P_{\text{cd}}}$$

from: G. Tonon- Tokamak Concept Improvement- Int. School of Plasma Physics
P. Caldirola - Varenna 1994
Analytical values of $\gamma$

$<\text{Te}>_{\text{unit}} = 10 \text{ keV} \quad [\gamma] = [10^{20}] \text{A.m}^{-2}/\text{W}$

1. LHCD:

$$\gamma_{\text{LH}} = \frac{0.37}{(5 + \text{Z}_{\text{eff}})} \frac{B_T}{(n_e)^{0.7}} <\text{Te}>;$$

frequency $>5$ GHz on ITER to avoid a particle absorption.

2. Fast wave CD:

$$\gamma_{\text{FW}} = \frac{0.63 <\text{Te}>}{2 + \text{Z}_{\text{eff}}}$$

Frequency $<50$ MHz, $20$ MHz for ITER to avoid ion absorption.

3. NB CD:

$$\gamma_{\text{NB}} = 0.25 <\text{Te}> F(\text{Z}_{\text{eff}}, \text{Z}_b, \text{A})$$

$F(\text{Z}_{\text{eff}}, \text{Z}_b, \text{M}_b) \sim 1$;

Optimum beam energy:

$E_b \sim (30\% - 50\%) \text{ M}_b \text{Te}$;

$\text{Z}_b, \text{M}_b$ beam ion charge and mass

$A = R_{\text{aspect ratio}}$

On average: with $R \sim 8 \text{ m}; \quad \gamma = 0.3 \times 10^{20} \text{ A.m}^{-2}/\text{W}; \quad I_{\text{CD}} = 2 \text{ MA}; \quad P_{\text{CD}} \sim 50 \text{ MW}$
Table I - Parameters of conceptual reactor design compared with ITER outline design (EDA)

<table>
<thead>
<tr>
<th>Parameter</th>
<th>INTOR</th>
<th>ITER</th>
<th>ARIES I</th>
<th>SSTR</th>
<th>ARIES II</th>
<th>SEAEFP Baseline solution st.</th>
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<td>R (m)</td>
<td>5.0</td>
<td>7.7</td>
<td>6.75</td>
<td>7</td>
<td>5.8</td>
<td>9.4</td>
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<tr>
<td>a (m)</td>
<td>1.2</td>
<td>3</td>
<td>1.50</td>
<td>1.7</td>
<td>1.4</td>
<td>2.09</td>
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<tr>
<td>A</td>
<td>4.2</td>
<td>2.56</td>
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<td>k</td>
<td>1.6</td>
<td>1.6</td>
<td>1.8</td>
<td>1.85</td>
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<td>Plasma volume (m³)</td>
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<td>2200</td>
<td>546</td>
<td>688</td>
<td>445</td>
<td>1360</td>
</tr>
<tr>
<td>I (MA)</td>
<td>8</td>
<td>24</td>
<td>10.2</td>
<td>12</td>
<td>8.5</td>
<td>10.4</td>
</tr>
<tr>
<td>B (T)</td>
<td>5.5</td>
<td>6</td>
<td>11.3</td>
<td>9</td>
<td>8</td>
<td>7.8</td>
</tr>
<tr>
<td>q (95%)</td>
<td>3</td>
<td>4.75</td>
<td>4.5</td>
<td>12.2</td>
<td>3.9</td>
<td></td>
</tr>
<tr>
<td>&lt;J&gt; %</td>
<td>4.9</td>
<td>2.68</td>
<td>1.9</td>
<td>2.7</td>
<td>3.4</td>
<td>2.2</td>
</tr>
<tr>
<td>Pé (Troyes)</td>
<td>4.1</td>
<td>2</td>
<td>3.2</td>
<td>3.5</td>
<td>6</td>
<td>3.5</td>
</tr>
<tr>
<td>Pépol</td>
<td>1.37</td>
<td>0.59</td>
<td>2.18</td>
<td>2</td>
<td>5.4</td>
<td>2.31</td>
</tr>
<tr>
<td>fBS</td>
<td>0.46</td>
<td>0.15</td>
<td>0.68</td>
<td>0.75</td>
<td>0.9</td>
<td>0.84</td>
</tr>
<tr>
<td>Pé (GW)</td>
<td>0.6</td>
<td>1.5</td>
<td>1.925</td>
<td>3.0</td>
<td>2.63</td>
<td>3.0</td>
</tr>
<tr>
<td>Pé (GW)</td>
<td></td>
<td></td>
<td>1.000</td>
<td>1.080</td>
<td>1.000</td>
<td>1.000</td>
</tr>
<tr>
<td>M (kt)</td>
<td>34</td>
<td>10.1</td>
<td>25</td>
<td>10.3</td>
<td></td>
<td></td>
</tr>
<tr>
<td>E (GJ)</td>
<td>100</td>
<td>144</td>
<td>144</td>
<td>144</td>
<td></td>
<td></td>
</tr>
<tr>
<td>P [Pm] (MW/m³)</td>
<td>2.6</td>
<td>0.68</td>
<td>3.5</td>
<td>4.0</td>
<td>5.9</td>
<td>2.2</td>
</tr>
<tr>
<td>Pcp (MW)</td>
<td>50</td>
<td>100</td>
<td>60</td>
<td>75</td>
<td>75</td>
<td></td>
</tr>
<tr>
<td>Q</td>
<td>30</td>
<td>19</td>
<td>50</td>
<td>30</td>
<td>40</td>
<td></td>
</tr>
</tbody>
</table>
Design and R&D activities for fusion reactor program in China

J. Huang (Southwest Institute, China)
Fusion Reactor Design and R&D Activities in China

J.H. Huang
Southwestern Institute of Physics, P.O.Box 432,
Chengdu 610041, China

Abstract

This paper reviews the status of fusion reactor study in China since early 1970s. A fusion breeder program and a pure fusion reactor design program have been supported by the State Science and Technology Commission and the China National Nuclear Corporation, respectively. In addition, the National Nature Science Foundation has supported a fusion transmutation reactor design and a D-3He fueled reactor design studies. Among them, the fusion breeder program is the major one, which has a great impact on the development of fusion reactor study in China. The motivation of this program is explained. Then the fusion reactor design activities in the Southwestern Institute of Physics and the Institute of Plasma Physics, Academia Sinica, are introduced. The detailed conceptual design of a fusion experimental breeder, FEB, which has been jointly conducted by these two Institutes since 1993 is described, including some recent progress in the blanket design. It is the most developed design earned out so far in China. Finally, the fusion reactor R&D activities in China are reviewed, indicating some features of these activities and the progress in areas of materials, tritium technology, liquid metal technology, neutronics integral experiments and plasma engineering.

1. Introduction

The fusion reactor design study in China started in early 1970s by a group at Institute of Modern Physics, IMP, under the Atomic Energy Ministry of China. The group mainly worked on neutronics design of the reactor blanket. In 1973, academician, Professor Li, Zhengwu of Southwestern Institute of Physics, SWIP, presented a paper on the topic of fusion-fission symbiosis systems at a domestic conference on fusion research. Then, in 1974 a research group was set up at SWIP engaging in fusion reactor design study and at the end of 1970s, a similar group was organized at Institute of Plasma Physics. In 1985, a symposium on fusion reactor study was held in SWIP for the first time in China. In 1986, a fusion breeder program was proposed, the motivation of which was as follows. China has a large population and energy source relies heavily on coal. It is projected that a total annual energy consumption of 4 Gt equivalent standard coal will be needed by 2050. Though China has plenty of coal resources, pollution and transportation remain as big problems. A large scale development of nuclear energy will help to alleviate these problems. Fusion energy is the long term goal in China. However, it seems unlikely that commercial fusion plants will be available in China before 2050. Furthermore, it is not easy to maintain the expensive fusion research without an essential intermediate application at an earlier time. Fission energy is practical. It is obvious that the utilization of U-238 is needed for a large scale development of fission reactor plants. The fusion breeder has the potential to provide plenty of commercial fissile fuel and thus become an essential intermediate application of fusion energy. This potential is worth studying. Under this consideration a fusion breeder program was approved by State Science and Technology Commission, SSTC.

The first stage of this fusion breeder program was performed from 1987-1991. A nation-wide cooperation of domestic institutes was organized and coordinated by IPP. The second stage from 1993-1995 has been coordinated by SWIP. The third stage program, if approved, will extend to 2000 and conduct engineering design of a fusion experimental breeder, FEB, and relevant R&D activities. Due to the support by this program, we have made essential progress in fusion reactor study.

In addition to this major fusion reactor program,
some other smaller programs have been carried out: design study of a pure fusion reactor supported by China National Nuclear Corporation, CNNC; design studies of a D-3He fueled reactor and a fusion transmutation reactor for nuclear waste disposal were supported by the National Nature Science Foundation of China.

2. Fusion reactor design study

2.1 Design study at SWIP

Several designs which were performed in the past ten years are listed in Table 1.

The series design of Tokamak Engineering Test Breeder, TETB, accounted for the main effort of design study at SWIP up to 1991. After that, effort has been devoted to the design of a Fusion Experimental Breeder, jointly performed by SWIP and IPP.

<table>
<thead>
<tr>
<th>Table 1. Fusion reactor design studies at SWIP</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>I. Fusion-fission hybrid reactors</strong></td>
</tr>
<tr>
<td><strong>Name</strong></td>
</tr>
<tr>
<td>TETB</td>
</tr>
<tr>
<td>TETB-II</td>
</tr>
<tr>
<td>TCB</td>
</tr>
<tr>
<td>TETB-III</td>
</tr>
<tr>
<td>FEB</td>
</tr>
</tbody>
</table>

**Transmutation** 94---95 For Np-237 waste disposal

**II. Advanced fusion reactor**

Mooncity 92 --- 94 D - 3He fueled

**III. Pure fusion reactor**

STR 93 --- 95 commercial, \( P_{\text{fus}} = 3000 \text{ MW}, R/a = 900 \text{ cm} / 180 \text{ cm} \), long pulse(9 hours)

A fusion transmutation reactor design was made to examine a concept of burning high-level waste, HLW, utilizing the released energy and converting \(^{237}\text{Np}\) into \(^{239}\text{Pu}\). The main results of the study is given in Table 2.

The research of a D-3He fusion reactor, the Mooncity, included topics such as the investigation of \(^3\text{He}\) resource and mining possibility; calculation of energy payback factor with a result of \(-200\) for mining lunar \(^3\text{He}\) resource; scouting analysis of parameters for D-3He fueled tokamak reactor; Monte-Carlo simulation of helium ash removal, etc. The calculations of radioactivity, after heat and Biological Hazard Potential, BHP, show that reduction factors of 10-60 can be expected compared with a D-T reactor. The main features of the Mooncity design are listed in Table 3.

<table>
<thead>
<tr>
<th>Table 2. Fusion Transmutation Reactor</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Main features</strong></td>
</tr>
<tr>
<td>Plasma radius, ( R(m)/a(m) )</td>
</tr>
<tr>
<td>Fusion power, ( (\text{MW}) )</td>
</tr>
<tr>
<td>Neutron wall loading, ( (\text{MWm}^{-2}) )</td>
</tr>
<tr>
<td>Max. power density ( (\text{Wcm}^{-3}) )</td>
</tr>
<tr>
<td>(a) only (^{237}\text{Np})</td>
</tr>
<tr>
<td>(b) with Pu as neutron multiplier</td>
</tr>
<tr>
<td>Transmutation rate of Np ( (\text{kg yr}^{-1}) )</td>
</tr>
<tr>
<td>(a)</td>
</tr>
<tr>
<td>(b)</td>
</tr>
<tr>
<td>( k_{\text{eff}} )</td>
</tr>
<tr>
<td>(a)</td>
</tr>
<tr>
<td>(b)</td>
</tr>
<tr>
<td>Amount of (^{239}\text{Pu}) generated ( (\text{kg yr}^{-1}) )</td>
</tr>
<tr>
<td>Support ratio for 1GW PWR</td>
</tr>
<tr>
<td>(a)</td>
</tr>
<tr>
<td>(b)</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Table 3. D-3He Fusion Reactor, Mooncity</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Main features</strong>:</td>
</tr>
<tr>
<td>Major radius ( R(m) ),</td>
</tr>
<tr>
<td>Minor radius ( a(m) ),</td>
</tr>
<tr>
<td>Elongation k,</td>
</tr>
<tr>
<td>Magnetic field at axis ( B_r(T) )</td>
</tr>
<tr>
<td>Plasma current ( I_p(\text{MA}) )</td>
</tr>
<tr>
<td>Average ion density ( &lt;n_i&gt; ) ( (10^{20} \text{ m}^{-3}) )</td>
</tr>
<tr>
<td>Ion temperature ( &lt;T_i&gt; ), (keV)</td>
</tr>
<tr>
<td>Electron temperature ( &lt;T_e&gt; ), (keV)</td>
</tr>
<tr>
<td>Fusion power ( P_{\text{fus}}(\text{MW}) )</td>
</tr>
<tr>
<td>Net electrical power, (MW)</td>
</tr>
</tbody>
</table>

The design of a commercial fusion reactor named Slender Tokamak Reactor, STR, has been
carried out since 1993 and will completed at the end of 1995. It has a high aspect ratio and features a long pulsed operation mode. The parameters of the STR are shown in Table 4.

Table 4. Conceptual Design of a Slender Tokamak Reactor

<table>
<thead>
<tr>
<th>Main parameters:</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Major radius, m</td>
<td>9.0</td>
</tr>
<tr>
<td>Minor radius, m</td>
<td>1.8</td>
</tr>
<tr>
<td>Aspect ratio</td>
<td>5.0</td>
</tr>
<tr>
<td>Elongation</td>
<td>1.8</td>
</tr>
<tr>
<td>Plasma current Ip, MA</td>
<td>13.0</td>
</tr>
<tr>
<td>Magnetic field at axis, T</td>
<td>7.9</td>
</tr>
<tr>
<td>Average temperature, keV</td>
<td>15.7</td>
</tr>
<tr>
<td>Electron density, m⁻³</td>
<td>1.4×10²⁰</td>
</tr>
<tr>
<td>Bootstrap current fraction</td>
<td>46%</td>
</tr>
<tr>
<td>Energy confinement time, s</td>
<td>1.9</td>
</tr>
<tr>
<td>H-mode enhancement factor</td>
<td>2.0</td>
</tr>
<tr>
<td>Total magnetic flux, V·s</td>
<td>1590</td>
</tr>
</tbody>
</table>

Current design status:
- System study of reactor parameters (POPCON)
- IV²-D plasma transport simulation
- Auxiliary heating power investigation for ignition blanket/shield neutronics
- PFC and TFC system and reactor structure.

2.2 Design Study at IPP

Parallel design activities have been carried out at IPP, a brief description of which is given in Table 5.

2.3 Detailed conceptual design of the FEB

As was mentioned above, FEB is a joint design by SWIP and IPP since 1993. It is the most detailed fusion reactor design so far in China, and at next stage, the reactor R&D activities will be based on this design.

The objectives of the FEB are to demonstrate engineering feasibility of fusion breeder, to demonstrate an annual ²³⁵Pu production capability of ~100kg, and to function as a test reactor for materials and key components for the next step reactor, the DEMO.

Based on previous design studies accomplished in both Institutes, the parameters shown in Table 6 were chosen for the FEB.

Table 5. Fusion reactor Studies at ASIPP

<table>
<thead>
<tr>
<th>Name</th>
<th>Design period</th>
<th>Main features</th>
</tr>
</thead>
<tbody>
<tr>
<td>SSEHR</td>
<td>88–91</td>
<td>compact, R/a = 2.6m/0.8m, low Q of ~2, H₂-cooled solid tritium breeder</td>
</tr>
<tr>
<td>Tokamak transmutation reactor (TTR) for Actinides</td>
<td>91–93</td>
<td>moderate neutron wall load = 1.0M W/m²</td>
</tr>
<tr>
<td></td>
<td></td>
<td>R as neutron multiplier kₚ = 0.9, high power density in blanket 3.00 - 5.00 M W/m²</td>
</tr>
<tr>
<td></td>
<td></td>
<td>support ratio 20 - 30 PWR</td>
</tr>
<tr>
<td></td>
<td></td>
<td>for fission products D₂O as moderator</td>
</tr>
<tr>
<td></td>
<td></td>
<td>R as neutron multiplier kₚ = 0.97, Arranges Sr, Cs, Tc, I at different places poloidally</td>
</tr>
<tr>
<td>III. Small Tokamak (VNS) fusion power 200 MW</td>
<td>93–95</td>
<td>joint design with SWIP</td>
</tr>
<tr>
<td></td>
<td></td>
<td>wall load 1.0 MW/m²</td>
</tr>
<tr>
<td></td>
<td></td>
<td>test size 1.3 m x 1.3 m x 0.8 m</td>
</tr>
</tbody>
</table>

Table 6. Parameters of FEB

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Major radius, R (m)</td>
<td>4.0</td>
</tr>
<tr>
<td>Minor radius, a (m)</td>
<td>1.0</td>
</tr>
<tr>
<td>Elongation k</td>
<td>1.7</td>
</tr>
<tr>
<td>Plasma current Ip, (MA)</td>
<td>5.7</td>
</tr>
<tr>
<td>Magnetic field at axis, (T)</td>
<td>5.2</td>
</tr>
<tr>
<td>Global energy confinement time, (s)</td>
<td>0.9</td>
</tr>
<tr>
<td>Plasma temperature, (keV)</td>
<td>10.0</td>
</tr>
<tr>
<td>Average electron density, (10²⁰/m³)</td>
<td>1.1</td>
</tr>
<tr>
<td>Fusion power, (MW)</td>
<td>143.0</td>
</tr>
<tr>
<td>Neutron wall loading, (MW/m²)</td>
<td>0.45</td>
</tr>
<tr>
<td>Auxiliary heating power, (MW)</td>
<td>50.0</td>
</tr>
<tr>
<td>Blanket coolant, He (MPa)</td>
<td>5</td>
</tr>
<tr>
<td>Number of TFC</td>
<td>16</td>
</tr>
</tbody>
</table>

In plasma area, following topics have been studied to a certain depth:
- Rationale for the FEB plasma parameter design,
- Determination of main plasma parameters,
- Heating, current-drive and bootstrap current analyses,
- Energetic alpha-particle confinement,
- Fueling and alpha-ash exhausting,
- Sensitivities of the FEB plasma parameters to uncertainties,
• plasma burning control,
• 1 1/2 dimensional simulation of the plasma evolution process,
• divertor with gas puffing and electromagnetic plug,
• operation mode.

Since blanket is the key nuclear component and safety and reliability of the reactor strongly depend on the blanket, emphasis has been put on the blanket design. To accommodate the blanket structure to the complex tokamak geometry, a pebble bed structure is adopted. The selected compositions for the basic blanket are He/liquid lithium/ SS316/ uranium/ Be. LLI self-cooled blanket is attractive; however, self-healing technology for insulator to solve the MHD pressure drop problem has to be developed. So, helium coolant is selected and LLI is adopted as tritium breeder instead of solid breeder to improve the heat conduction in the pebble bed. As a result, the number of cooling channels is reduced considerably. Each of 16 sectors consists of 2 inboard and 3 outboard blanket modules for maintenance requirement. Two types of basic blankets have been designed with identical outer shape of module. Some results obtained recently for the design type A are given here as an example. This design features a fission-suppressed OB blanket with low energy multiplication so that the temperature evolution during LOCA will be moderate. The OB modules are long curved box along the poloidal direction. The cross section at middle plane is shown in Fig. 1. The modules are cooled by 2 independent He cooling systems. 4 panels with built-in cooling channels serve as primary cooling system; a shield cooling system is provided at the back of the modules. In addition, the liquid lithium can flow slowly in a loop for tritium extraction and provides an additional cooling measure in case of accident.

The neutronics calculations using different transport codes and nuclear data libraries were made by us and Russian colleague and compared. The agreement is satisfactory. The actual figure of the blanket, the large ports and nonuniform neutron source in the plasma were modelled in a detailed 3-D Monte-Carlo calculation. The nuclear data of Li-6, Li-7 and Be from the newest library FENDL were used in the calculation, the results of which is given in Table 7.

Fig.1. Cross section of FEB outboard blanket module (type A)

Table 7 Parameters from 3-D Calculation

<table>
<thead>
<tr>
<th>Parameter</th>
<th>IB</th>
<th>OB</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>T</td>
<td>0.431</td>
<td>0.367</td>
<td>0.798</td>
</tr>
<tr>
<td>F</td>
<td>---</td>
<td>0.397</td>
<td>0.397</td>
</tr>
<tr>
<td>T+F</td>
<td>0.425</td>
<td>0.678</td>
<td>1.195</td>
</tr>
<tr>
<td>U-235(n,f)</td>
<td>---</td>
<td>0.015</td>
<td>---</td>
</tr>
<tr>
<td>U-238(n,f)</td>
<td>---</td>
<td>0.064</td>
<td>---</td>
</tr>
<tr>
<td>Be(n,2n)</td>
<td>0.119</td>
<td>0.413</td>
<td>0.532</td>
</tr>
</tbody>
</table>

Thermal-mechanic analysis using the ANSYS code was performed taking all loads into account such as static weights of the materials, helium pressure, temperature difference throughout the module and electromagnetic force due to the eddy current induced in transient processes. It is shown that the module structure needs to be strengthened by stiffening ribs to reduce the stress and deformation.

The consequences of primary cooling system failure were examined in the safety analysis. Then, a preliminary failure sequence analysis was made with the failure rate data from available references. The results for LOCA is shown in Fig. 2. The failure sequence of the LOFA is similar, except that the failure probability of LOFA is 2 orders of magnitude higher. The detailed fault tree analyses for reactor shut-down system failure and LOFA are in progressing to find ways to reduce their failure probability and consequence.
3. Fusion Reactor R&D Activities

Regarding R&D activities, reference is given to an article "Fusion reactor design and technology program in China," presented in 1993. Here, a general description and some information of status since 1993 are given.

3.1 Some features of the R&D activities

The financial support goes mainly to research work but not to building new major experimental facilities. So, it is required that the activities make use of available facilities in China, such as a fission reactor and a neutronics integral experiment device at Southwestern Institute of Nuclear Physics and Chemistry, a tritium laboratory at Institute of Atomic Energy; a heavy ion research facility of Lanzhou, the HIRHL, at Institute of Modern Physics, IMP, etc. Only some small or medium size devices were built, such as a liquid metal loop at SWIP, a tritium production loop installed in the fission reactor at SWINPC, some devices for hydrogen isotope permeation barrier research, including ion driven and gas driven permeations, etc.

It is important to develop a core plasma for the fusion reactor. So, the fusion breeder program has supported some topics of plasma engineering. These are lower hybrid current drive, LHCD, ICRF and NBI heating and pellet injection. These researches account for a large part of the program funding and have made essential progress under this program. That explains why we put these researches into the reactor study category.

3.2 R&D activities

3.2.1 Materials

Various kinds of materials having fusion applications have been prepared and some of them tested. The materials prepared are austenitic stainless steel, 316L, and its modifications with Ti, Ni, Si, etc. for better irradiation resistance; ferritic steels, HT-9, HT-7 (oxide dispersion strengthening, ODS); vanadium based alloys; ceramics, and SiC/SiC composite for structural materials; low Z materials, TiC, SiC and diamond coating, and W-Cu alloy for plasma facing materials; LiAlO2, Li ceramics, PbLi and liquid lithium for tritium breeding materials. Some materials were tested in various devices such as an intensive D-T neutron source (3×10^{12} n/s at Lanzhou University), a high flux reactor up to 6.2×10^{14} n/(cm².s) at Nuclear Power Institute of China, the HIRHL at IMP using heavy ion from C to Ta and 3 MeV alpha-particles, 1MV electron microscopy and thermal shock test facility. The results of test experiments have been presented and published.

3.2.2 Tritium technology

To reduce the tritium leakage from fusion reactors, tritium permeation barrier materials have been studied. Most extensive experiments using tritium were conducted at CIAE and satisfactory results were obtained. To reduce experimental difficulty and provide independent check, hydrogen has been used instead of tritium at Institute of Solid State Physics to investigate gas driven permeation. Main results form these experiments are shown in Table 8 and 9.

In order to demonstrate the feasibility of the recovery of tritium generated from solid breeders and to study key operation factors affecting tritium release, an in-situ experiment (CITP) had been performed in a fission reactor at SWINPC. Solid breeder, LiAlO2 pellets were prepared and experimental results were obtained. After 1993, the experiment was halted due to the high cost to run the fission reactor.
Table 8 Permeability of Tritium through 316L ss and Its Coating Materials

<table>
<thead>
<tr>
<th>Specimen</th>
<th>Permeability ( \Phi = \Phi_0 \exp\left(-\frac{E_0}{RT}\right) )</th>
<th>( \Phi_0 )</th>
<th>( E_0 )</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pd+316ss+Pd</td>
<td>3.24x10^{-5}</td>
<td>60.075</td>
<td></td>
</tr>
<tr>
<td>Cr_2O_3+HR-1ss+Cr_2O_3</td>
<td>1.07x10^{-5}</td>
<td>76.788</td>
<td></td>
</tr>
<tr>
<td>TiC+TiN+3161ss+TiN+TC</td>
<td>3.30x10^{-5}</td>
<td>78.281</td>
<td></td>
</tr>
</tbody>
</table>

Table 9 Permeability of Hydrogen through HR-1 ss and Its Coating Materials

<table>
<thead>
<tr>
<th>Specimen</th>
<th>Permeability ( \Phi = \Phi_0 \exp\left(-\frac{E_0}{RT}\right) )</th>
<th>( \Phi_0 )</th>
<th>( E_0 )</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pd+HR-1ss+Pd</td>
<td>2.03x10^{-4}</td>
<td>66.00</td>
<td></td>
</tr>
<tr>
<td>Cr_2O_3+HR-1ss+Cr_2O_3</td>
<td>2.87x10^{-4}</td>
<td>84.00</td>
<td></td>
</tr>
<tr>
<td>TiN+HR-1ss+TiN</td>
<td>2.28x10^{-5}</td>
<td>92.569</td>
<td></td>
</tr>
<tr>
<td>TiN+TiC+HR-1ss+TiC+TiN</td>
<td>4.62x10^{-6}</td>
<td>99.306</td>
<td></td>
</tr>
<tr>
<td>TiN+Ti(C,N)+TiC+HR-1ss+Ti(C,N)+TiN</td>
<td>5.33x10^{-7}</td>
<td>94.738</td>
<td></td>
</tr>
</tbody>
</table>

3.2.3 Liquid metal loop experiments

LLL self-cooled blanket remains as an attractive concept to pursue. Test modules of this concept will be put into the FEB. MHD pressure drop experiments have been conducted on the liquid metal loop, LMEL, at SWIP, using Nak eutectic alloy flowing through circular duct under constant and quickly changing magnetic field. Circular duct with insulation layer coated from inside was prepared and experiment was conducted to study the insulation effect.

3.2.4 Neutronics integral experiments

A good neutronics performance is vital for a fusion breeder. To check the nuclear data and calculation methods, the program has supported neutronics integral experiments carried out at SWINPC. Useful results have been provided from these experiments. Another concern is the shield for magnets due to the deep penetration transport and the strong streaming through large ports. Experiments have been planned at SWINPC with the first series experiments using Fe and Fe-H_2O alternative layers.

3.2.5 Plasma engineering

Two series of tokamak experiments have been developed in China, i.e. HT-6M, HT-7 to HT-7U, and HL-1, HL-1M to HL-2A in IPP and SWIP respectively. These experiments have provided target plasmas for the development of plasma engineering supported by the fusion breeder program. During the first stage program, a LHCD
system was built at SWIP and experiment was performed in 1992 with a peak output of 200 kW from a single klystron and a pulse length of 0.1s; a single shot pellet injection experiment was performed in 1992 at SWIP with a speed up to 500 m/s, and a 2MW ICRH system was developed at IPP. At the end of second stage program, the following hardwares will be developed and preliminary experiments will be conducted at HT-7 and HL-1M: LHCD, 4×0.5MW klystrons with pulse length of 0.1-1.0s at SWIP, and 20×0.1MW DC klystrons at IPP. Pellet injection, 8 shots with a speed up to 1km/s at SWIP. 2MW ICRH experiments at IPP and 1MW 40kV NBI with a pulse length of 0.1-1.0 s at SWIP.

4. Summary

Fusion reactor study has been conducted in China since early 1970s. However, essential progress both in reactor design and R&D activities has been made in recent 10 years mainly due to the support of the fusion breeder program. Domestic and international cooperations has also been developed.

References:

Critical heat-flux issues in fusion reactor diverter design

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빈 관
Critical Heat Flux Issues in Fusion Reactor Divertor Design

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ABSTRACT

This paper discusses cooling aspects of the divertor in Tokamak fusion devices with primary emphasis on the critical heat flux (CHF) issues in water-cooled designs. General characteristics of four (4) coolant options for divertor cooling - gases, water, liquid metal, and organic liquid, are discussed first, including the comparison of advantages and disadvantages of those options. Then recent studies on the high-heat-flux CHF of water at subcooled high-velocity conditions are reviewed to provide a general idea on the feasibility of the water-cooled divertor concept for future Tokamak-type fusion reactors.

1. INTRODUCTION

The main functions of the divertor system in a D-T Tokamak fusion reactor are (i) the exhaust of helium ash, (ii) impurity control, and (iii) the removal of heat transported by particles to the plasma edge. Plasma particles reaching the scrape-off layer either from the main plasma or from the first wall move along the magnetic field lines until they strike the divertor target plate where they are neutralized and are then extracted by the vacuum pumping system [Thompson & Worraker 1990].

The particles striking the divertor target are at very high energy and intensity; so two basic engineering problems are brought in divertor design. First, the high-energy particle beams cause the erosion of the target material by sputtering (material problem). Second, the peak heat flux can be of order 10 MW/m², which is higher than that encountered in water-cooled fission reactors by an order of magnitude (cooling problem).

A combination of materials of high erosion-resistant, high temperature-resistant, high irradiation-resistant, and excellent heat transfer characteristics is required together with a novel cooling system design. At present, the divertor system represents one of the most difficult tasks in design of future Tokamaks such as International Thermonuclear Experimental Reactor (ITER) [Rebut et al. 1993]. Some of divertor plate design concepts for ITER are illustrated in Fig. 1.

It is found that four kinds of coolant materials used presently or in the past for fission reactors can be considered for cooling of the divertor system: water, gases, liquid metals and organic liquids. As in the case for fission reactors, there is no absolute choice for the divertor coolant; instead, the coolant material should be selected by considering the overall design and operating characteristics of individual reactors. General requirements for the divertor coolant and characteristics of each coolant option are reviewed in Section 2.
Among the coolant options, water is very attractive in many aspects: good heat transfer properties, easy and safe handling, abundance, economics, etc. Therefore it has been adopted as the coolant for many experimental Tokamak reactors. One of the critical problems related to water cooling is the existence of the critical heat flux (CHF) above which the cooling capability is drastically degraded. Though the CHF is a thermal-hydraulic phenomenon that has been investigated extensively with the development of water-cooled fission reactors, the conditions imposed on the divertor cooling system is unique due to its very high peak heat fluxes and non-uniformity in heat flux distributions. It requires very high water velocity and sometimes heat transfer enhancement devices to remove the unusually high heat fluxes of the divertor plate. During the last 10 years, there have been significant contributions on the high-heat-flux CHF for water flow relevant to fusion reactors by several workers. Their achievement and the state-of-the-art knowledge on the high-heat-flux CHF related to fusion applications are summarized in Section 3.

The overall objective of this paper is to provide a general understanding of the CHF issues relevant to divertor cooling, so detailed description of the specific phenomenon and modeling aspects are not included.

2. COOLANT OPTIONS FOR DIVERTOR COOLING

2.1 Overall

In choosing a coolant for either fission or fusion reactor components, the following criteria can be applied [Marshall 1983]:

(a) the coolant must be chemically compatible with the structural materials over the desired temperature range;

(b) The coolant must be commercially available in large quantities at an acceptable price;

(c) the pumping power required to remove heat at a given rate must be a minimum;

(d) the coolant pressure must be acceptable in terms of reactor safety and structural material costs; and

(e) The coolant must be stable under intense neutron and gamma-ray irradiation in the core.

In the above, the third criterion is directly related to the heat transfer properties. Additional desirable characteristics include: low melting point (for liquid coolants); high boiling point (for liquid coolants); high operating temperature for high power conversion efficiency; high thermal stability; low induced radioactivity; easy and safe handling characteristics, etc. [Ma 1983].

Five coolant materials are used in existing fission reactors: (a) light water (H₂O) for pressurized water reactors (PWRs) and boiling water reactors (BWRs), (b) heavy water (D₂O) for pressurized heavy water reactors (PHWRs), (c) carbon dioxide (CO₂) gas for British-type gas cooled reactors (GCRs), (d) helium (He) gas for high-temperature gas-cooled reactors (HTGRs), and (e) liquid sodium (Na) for liquid-metal fast breeder reactors (LMFBRs). Organic liquids and other liquid metal coolants were also investigated but have not attracted much attention. Major characteristics of coolant materials are well discussed in books by Ma (1983) and by Gupta (1989).

Several coolant options also exist for divertor cooling. It is difficult to predict at present what will prevail as the fusion reactor coolant in the future, because it also depends on the progress of other fusion reactor design and analysis technologies. For instance, the peak heat flux on the ITER divertor plate is
assessed to be less than 10 MW/m$^2$ at the present engineering design activity (EDA), which is considerably smaller than 15-30 MW/m$^2$ assumed during the conceptual design activity (CDA). For the heat fluxes of 15-30 MW/m$^2$, high-velocity subcooled flow boiling of water would be the only practical method for reliable cooling. For heat fluxes of several MW/m$^2$, however, other coolant options could be effective, although water is still the best when only the cooling capability is considered.

Boyd et al. (1985) assessed three high-heat-flux heat removal techniques for fusion applications: (a) subcooled flow boiling (SFB) with water, (b) high-velocity helium gas convection, and (c) liquid lithium cooling. They concluded that subcooled flow boiling of water would be the most efficient heat removal technique, and several aspects of related thermalhydraulics, in particular the high-heat-flux CHF, should be further investigated. Thompson & Worraker (1990) presented the results of some engineering studies for the divertor target plate. Considering four limits related to cooling (surface temperature limit, coolant velocity limit, pressure drop limit, and pumping power limit), they suggested two cooling options: (a) cooling by helium at 400 bar with $T_{\text{inlet}} = 325^\circ C$ and $T_{\text{outlet}} = 375^\circ C$ in tubes of diameter 9 mm, and (b) cooling by supercritical water at 350 bar with $T_{\text{inlet}} = 320^\circ C$ and $T_{\text{outlet}} = 380^\circ C$ in tubes of diameter 2.7 mm. Recently Merola & Matera (1994) surveyed coolant options for a monolithic CFC divertor that was proposed by themselves. They excluded water from viable coolant options and advocated helium gas, HB-40 organic liquid and some liquid metals.

2.2 Comparison of Coolant Materials

Table 1 summarizes some thermophysical properties of important coolant materials. Please note that the properties are evaluated at different conditions between different materials. Table 2 compares relative advantages and disadvantages of four coolant options.

Water has excellent heat transfer characteristics due to its favorable properties such as high specific heat, moderate density, low viscosity, etc. High heat transfer coefficient allows only a small temperature rise of the surface over the liquid saturation temperature. High specific heat together with high heat transfer coefficient result in reasonable pressure drop and required pumping power that are considerably smaller than those for gases or liquid metals. There are no effects of magnetic field on the pressure drop; instead, electric fields produced by the plasma can enhance the boiling heat transfer. Furthermore, there exists extensive operating experience which is uncomparable to other coolant options. The existence of the CHF condition, however, brings an important design concern. Relatively low operating temperature of water can be another major disadvantage in fusion reactors for electricity generation.

Gas coolants are not subjected to magneto-hydrodynamic forces and CHF problems; however, high pumping power requirements and low heat transfer properties can be significant disadvantages. Helium (He) gas is excellent in the aspects chemical inertness, high operating temperature, no induced radioactivity, easy tritium extraction, mild leakage problem, etc. Main disadvantages of helium would be the high pumping power requirement and high operating pressure. The peak heat flux should be lowered to several MW/m$^2$ to make helium cooling very attractive.

There are many liquid metal coolant options: sodium (Na), lithium (Li), potassium (K), sodium-potassium (Na-K), bismuth (Bi), lead (Pb), lead-bismuth (Pb-Bi), tin (Sn), lead-lithium (Pb-Li), gallium (Ga),
mercury (Hg), etc. They generally have good heat transfer characteristics primarily owing to high thermal conductivity. But they are subjected to magnetohydrodynamic forces, resulting in high pumping power requirements. Lithium or lead-lithium can be considered for both breeder and coolant; however, poor compatibility of lithium with graphite and relatively high melting points (179°C for Li; 235°C for Pb-Li) bring significant troubles. Bismuth, lead and tin are not appropriate due to their high melting points (> 200°C). Mercury has too low boiling temperature and toxicity. This leads to the conclusion that gallium, sodium, and lead-bismuth can be possible liquid metal coolants. If the liquid lithium or lead-lithium are used for the breeder blanket, they should also be seriously considered as the divertor coolant.

HB-40 is an organic liquid used in a Canadian research reactor WR-1 [Turner 1974]. It shows relatively good characteristics as a coolant material; however, decomposition and flammability at high temperature can be problematic.

There are several other coolant options and cooling methodologies. However, it can be concluded as follows:

(a) Water in subcooled boiling conditions shows the best cooling capability with many other favorable characteristics; so it can serve as a reliable coolant of experimental fusion reactors for which high coolant temperature for efficient power conversion is not a primary concern. It would be the only coolant that could remove heat fluxes far above 10 MW/m².

(b) Helium is an excellent coolant in the aspects of high operating temperature, chemical and radiological inertness, easy cleanup and tritium extraction, etc. It would be attractive for fusion power reactors if the heat flux level can be lowered to several MW/m².

3. CHF OF WATER IN DIVERTOR-RELEVANT CONDITIONS

3.1 General Discussion

The CHF condition is characterized by a sharp reduction of the local heat transfer coefficient which results from the replacement of liquid by vapor adjacent to the heat transfer surface [Collier 1982]. It has been extensively investigated over the last 40 years with the development of water cooled reactors. However, the CHF problem related to divertor cooling is unique at least in two aspects: (a) high heat flux and (b) non-uniform heat flux distribution.

One of reliable CHF prediction methods over a wide range is the look-up table approach developed by USSR Academy of Sciences (1976) and improved by Groeneveld et al. (1986). Both tables present the CHF for a uniformly-heated 8 mm-ID vertical tube at various values of local conditions (pressures, mass fluxes, and local qualities or subcooling temperatures). In the Groeneveld et al. (1986) table for mass fluxes < 7500 kg/m²s, the highest CHF of ~17 MW/m² is given at pressure 45-70 bar, mass flux of 7,500 kg/m²s and local quality of -0.50. As the CHF can be further increased by increasing the mass flux and/or by decreasing the diameter, one can expect that substantially higher heat flux can be removed through water subcooled boiling. In addition, the CHF can be further increased by introducing various CHF enhancement techniques. However, it should be noted that those tables were constructed based on very limited number of high heat flux data.

There were limited number of available literatures on the high-heat-flux CHF until
Boyd (1985a, b) first published a comprehensive review on this subject relevant to fusion components cooling. Since then, there have been an increasing effort to investigate the CHF phenomenon both experimentally and theoretically, and findings of subsequent investigations are well summarized in the review papers of Boyd (1988), Inasaka & Nariai (1993) and Celata (1993). It is found that the highest CHF value ever measured is 227.9 MW/m² at very high flow, highly subcooled conditions in a 0.4mm-ID tube. This highest heat flux may not be practical due to other design constraints; however, it demonstrates the effectiveness of water in high-heat-flux component cooling.

The following sections summarize the parametric trends, enhancement techniques tested, and some prediction correlations for the high-heat-flux CHF.

3.2 Important Parametric Trends

Basically there are five important parameters affecting the CHF in uniformly heated vertical round tubes: (a) pressure (P), (b) mass flux (G), (c) inlet subcooling (ΔT₀) or exit quality (X), (d) tube diameter (D), and (e) tube heated length (Lₜ). Other flow channel geometries and orientations also affect the CHF magnitude. The parametric effects of the CHF are schematically shown in Fig. 2 and discussed below. Discussions are primarily based on the review by Celata (1993). It should be noted that parametric trends are discussed based on fixed exit conditions instead of fixed inlet conditions.

(a) Influence of Exit Quality
The CHF is a decreasing function of the exit (or local) quality, showing almost linear relationship in highly subcooled conditions. Low exit quality can be achieved by lowering the inlet temperature, i.e., by increasing the inlet subcooling.

(b) Influence of Mass Flux
The CHF is an increasing function of the mass flux (or fluid velocity) in the subcooled region.

(c) Influence of Pressure
Groeneveld et al. (1986) table suggests that the CHF decreases sharply in the very low pressure region, then increases until 3-7 MPa, and then slowly decreases again, in the case that quality is considered as the local parameter[Chang et al. 1991; Moon et al. 1995].

(d) Influence of Channel Diameter
The CHF generally increases as the channel diameter decreases at constant exit conditions, showing large effects in small diameters and high mass fluxes. This is the reason that extremely high CHFs were measured in some tests with tubes of diameter less than 1 mm. However, the effect of diameter is rather small for diameters of the order 1 cm, which would be more practical for divertor cooling channels.

(e) Influence of Channel Length
For small Lₜ/D, the CHF is generally a decreasing function of Lₜ/D or the heated length. However, the length effect becomes negligible for larger Lₜ/D, say >30-40.

(f) Influence of Channel Orientation
The flow orientation (horizontal or vertical) does not have significant effect on the high-heat-flux CHF under high velocity conditions. Therefore, most data measured for vertical orientation can be applied to divertor design with horizontal flow paths.

(g) Influence of Non-uniform Heating
The effect of non-uniform heating has not been analyzed satisfactorily. However, it is expected that nonuniformity in heat flux distribution would not significantly affect the CHF if actual local flow conditions are considered, since the subcooled boiling...
CHF at high velocities are basically a local phenomenon.

(h) Other Influences
Some investigators have questioned the effects of wall material or dissolved gas content in water. Most studies reveal negligible effects of those factors.

3.3 CHF Enhancement Techniques

Very high heat fluxes on the divertor plate (2-60 MW/m²) can be removed with subcooled water flow in straight smooth tubes. However, small diameter channels and/or high velocity conditions bring severe engineering problems as the heat duty becomes higher. Passive techniques for enhancing the CHF would alleviate this situation and facilitate practical cooling loop design. Three approaches of CHF enhancement are discussed below.

(a) Twisted Tapes Inside Circular Tubes
Twisted tape inserts inside circular tubes increase the heat transfer coefficient and the CHF by inducing swirl flow in the coolant. Disagreement in the degree of CHF enhancement is found among investigators; however, it could increase the CHF up to three times of that for smooth tubes at certain conditions with an increase of the pressure drop by several to several ten times.

(b) Helically Coiled Wires Inside Circular Tubes
Helically coiled wires have been tested as turbulence promoters to enhance the heat transfer and the CHF. They generally improve the CHF but degree of enhancement is smaller than the case of the twisted tape.

(c) Hypervapotron
A channel for the hypervapotron technique is characterized by the presence of fins of high thermal conductivity material, placed perpendicular to the flow of subcooled water [Cattadori et al. 1993]. Steam formation, rapid condensation of the steam and quenching of the heated wall are repeated in the spaces between adjacent fins and heated wall, thereby increase the heat transfer and the CHF. This technique is known to be suitable for the removal of high heat flux (up to about 30 MW/m²) where high values of fluid velocities or subcooling are not practical.

3.4 CHF Prediction Methods

More than 500 CHF correlations and models are available in the literature; however, only a limited number of them is really applicable to the high-heat-flux CHF relevant to divertor design. Recently assessment of correlations for the subcooled water CHF at high heat fluxes have been conducted by Yin et al. (1992), Celata et al. (1994) and Inasaka & Nariai (1993). For smooth tubes, the following correlation by Tong (1968) is widely used as a basis for further improvements:

\[
\frac{q_c}{h_{fg}} = C_{Tong} \left( \frac{G \mu}{D} \right)^{0.6}
\]

where

\[
C_{Tong} = 1.76 - 7.433 X - 12.222 X^2
\]

Inasaka modified the expression for C as follows:

\[
\frac{C_{Inasaka}}{C_{Tong}} = 1 - \frac{52.3 + 80X - 50X^2}{60.5 + (P/10^5)^{1.4}}
\]

On the other hand, Celata et al. (1993b) modified the Tong correlation as follows:

\[
\frac{q_c}{h_{fg}} = C_{Celata} \left( \frac{G \mu}{D} \right)^{0.5}
\]

where

\[
C_{Celata} = (0.216 + 0.00474P) \Psi
\]

\[
\Psi = \begin{cases} 
1 & \text{if } X < 0.1 \\
0.825 + 0.986X & \text{if } -0.1 < X < 0 \\
1/(2 + 30X) & \text{if } X > 0 
\end{cases}
\]

Both modified-Tong correlations are reported to show reasonable prediction capability with r.m.s. errors of about 20%.
Several other correlations show similar prediction capability; however, they are not included here.

Among the various mechanistic models of Weisman & Ileslamlou (1988), Chang & Lee (1989), Lee & Mudawwar (1988), Katto (1992) model is known to show the most promising prediction capability with an r.m.s. error of about 25% [Celata 1994, Inasaka & Nariai 1993].

For tubes with CHF enhancement devices, the present prediction correlations such as that of Inasaka & Nariai (1993) still require further refinement.

3.5 Discussion

Survey of CHF studies for subcooled flow of water indicates that water can effectively remove the heat imposed on the divertor plate. Understanding of the parametric trends and development of prediction methods have been progressed to the level that design of practical divertor cooling system is feasible.

However, much more experimental and theoretical work is necessary to find optimum design parameters. In particular, the CHF enhancement techniques should be studied more systematically to provide more friendly designs of the divertor cooling system.

It should be noted that the divertor cooling system cannot be designed only by considering normal operating conditions. Transient and accident conditions, e.g. loss-of-flow accident or loss-of-coolant accident, should be considered in design. Probably the minimum value of the CHF ratio (the ratio of the expected CHF to the actual heat flux) in cooling channels should be around 3 during normal operation, in order to allow transients in power and flow without occurrence of the CHF. An appropriate value of the minimum required CHF ratio can be established through transient and accident analyses. The coolant flow during those transients and accidents will be highly time-dependent; so transient effects on the CHF should also be investigated.

4. CONCLUSIONS

The coolant options of the fusion reactor divertors and the CHF characteristics of subcooled water flow under high-heat-flux conditions have so far been discussed. Important aspects can be summarized as below.

(a) The cooling requirements for the divertor plate are much severe than those for fission reactors due to very high peak heat fluxes.

(b) Forced convective subcooled boiling of water and forced convective cooling by helium gas would be two competing cooling methods of divertors for future Tokamak reactors.

(c) Water is an excellent coolant in the aspects of heat removal capability and required pumping power, easy & safe handling, abundance & economics, etc. Subcooled boiling of water would be the only cooling method for divertors with peak heat fluxes above 10 MW/m², and would be the easiest method of cooling for diverters of lower heat fluxes.

(d) High-velocity subcooled boiling of water under moderate pressures would be the most viable and practical method of divertor cooling for experimental Tokamaks where power generation is not a primary concern.

(e) The CHF is a parameter of paramount importance in determining the limits of water cooling. Understanding of the high-heat-flux CHF has been greatly progressed during the last 10 years; reasonable prediction methods are now
available for smooth tubes.

(f) Further investigation of the CHF is required for more reliable and optimum design of the divertor cooling system. Important topics include more sound understanding of physical mechanisms, improvement of prediction methods, CHF enhancement techniques, and transient effects.

5. REFERENCES


USSR Academy of Sciences (1976), Tabular data for calculating burnout when boiling water in uniformly heated round tubes, Teploenergtika 23, 90-92.


Table 1. Some Physical and Thermal Properties of Coolant Materials

<table>
<thead>
<tr>
<th>Coolant Materials</th>
<th>Water</th>
<th>Gas</th>
<th>Liquid Metal</th>
<th>Organic</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>H₂O</td>
<td>He</td>
<td>CO₂</td>
<td>Na</td>
</tr>
<tr>
<td>T_melt (1 atm), °C</td>
<td>0</td>
<td>-</td>
<td>-</td>
<td>98</td>
</tr>
<tr>
<td>T_bziel (1 atm), °C</td>
<td>100</td>
<td>-</td>
<td>-</td>
<td>883</td>
</tr>
<tr>
<td>Density, kg/m³</td>
<td>990</td>
<td>5.11</td>
<td>58.6</td>
<td>903</td>
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<tr>
<td>Specific Heat, J/kg-K</td>
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<td>5.195</td>
<td>1.084</td>
<td>1,330</td>
</tr>
<tr>
<td>Viscosity, 10⁴ N-s/m²</td>
<td>5.44</td>
<td>0.268</td>
<td>0.23</td>
<td>4.57</td>
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<tr>
<td>Th. Cond., W/m-K</td>
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<td>0.214</td>
<td>0.033</td>
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</tr>
<tr>
<td>Prandtl No.</td>
<td>3.515</td>
<td>0.655</td>
<td>0.76</td>
<td>0.0074</td>
</tr>
<tr>
<td>Elec. Resist., μΩ-m</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>0.0042</td>
</tr>
<tr>
<td>Induced Radioactivity</td>
<td>³¹⁷N</td>
<td>None</td>
<td>¹³C,¹⁶N</td>
<td>¹⁸Na</td>
</tr>
</tbody>
</table>

Note: Properties are evaluated at the following conditions: 3.5 MPa & 50°C for water; 5 MPa & 200°C for helium and carbon-dioxide; 0.1 MPa & 200°C for sodium, lithium, and gallium; 0.1 MPa & 250°C for lead bismuth; 0.1 MPa & 250°C for mercury; and 2 MPa and 200°C for HB-40.
Table 2. Comparison of Coolant Options Applicable to Diverter Cooling

<table>
<thead>
<tr>
<th>Characteristics</th>
<th>Water</th>
<th>Helium Gas</th>
<th>Liquid Metal</th>
<th>Organic Liquid</th>
</tr>
</thead>
<tbody>
<tr>
<td>Heat removal capability</td>
<td>Best (≤ -30 MW/m²)</td>
<td>Reasonable (≤ -5 MW/m²)</td>
<td>Good (≤ -10 MW/m²)</td>
<td>Good</td>
</tr>
<tr>
<td>Operating pressure</td>
<td>Medium to high</td>
<td>High</td>
<td>Atmospheric</td>
<td>Medium</td>
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<tr>
<td>Available temperature</td>
<td>Low</td>
<td>High</td>
<td>Highest</td>
<td>Medium</td>
</tr>
<tr>
<td>Required Pumping power</td>
<td>Small</td>
<td>High</td>
<td>Medium to high</td>
<td>High</td>
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<tr>
<td>Thermal &amp; radiational stability</td>
<td>Good</td>
<td>Best</td>
<td>Good</td>
<td>Reasonable at low temperature</td>
</tr>
<tr>
<td>Induced radioactivity</td>
<td>Low</td>
<td>None</td>
<td>Low</td>
<td>Low</td>
</tr>
<tr>
<td>Compatibility with structural materials</td>
<td>Good</td>
<td>Best</td>
<td>Reasonable</td>
<td>Good</td>
</tr>
<tr>
<td>Easy &amp; safe handling</td>
<td>Easy</td>
<td>Easy</td>
<td>Difficult</td>
<td>Reasonable</td>
</tr>
<tr>
<td>Well-known properties</td>
<td>Good</td>
<td>Good</td>
<td>Reasonable</td>
<td>Reasonable</td>
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<tr>
<td>Operating experiences</td>
<td>Extensive</td>
<td>Reasonable</td>
<td>Reasonable</td>
<td>Least</td>
</tr>
<tr>
<td>Abundance &amp; economics</td>
<td>Excellent</td>
<td>Reasonable</td>
<td>Reasonable</td>
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<tr>
<td>Magnetohydrodynamic effects</td>
<td>No</td>
<td>No</td>
<td>Yes</td>
<td>No</td>
</tr>
</tbody>
</table>

Fig. 1 Diverter Design Concepts for the ITER (Vieider at al. 1991; Merola & Matera 1994)
Fig. 2 Parametric Trends of Subcooled-Boiling Water Flow CHF

(a) Local Quality

(b) Mass Flux

(c) Tube Diameter

(d) Heated Length

(e) Pressure

Fig. 3 Examples of CHF Enhancement Devices

(a) Twisted Tape Inside a Tube

(b) Hypervaporton Technique
Progresses in high-power RF transmitter systems
development for tokamak fusion reactor application

J. Wyss (Thomcast)
RF systems technology used for fusion reactors has made significant progress during the last ten years. This progress concerns ICRH, LH and ECRH systems. In all of this RF systems circuit components and also the associated circuitry has been further developed resulting in higher performance, availability and efficiency. In this paper, some new components like RF tubes are presented and different solutions for RF generation are described.

1. INTRODUCTION

RF systems are nowadays required with very high power and increasingly higher pulse duration. Future fusion machines may even work with continuous RF power. The reduction of the power losses in the RF sources will thus become more and more important. In principle tetrodes for ICRH can work with very low losses if used with a well matched load and if working in class C operation. In fusion machines however, there is no constant load because of plasma changes. To overcome this constraint special circuit techniques have to be used.

In LH and ECRH the development of RF sources has also made remarkable progress concerning power, reliability and efficiency.

2. ICRH SYSTEM

RF generation is done in separate amplifier systems with powers up to 2 MW or more per unit. One system normally consists of a multistage amplifier with preliminary amplifier, driver and final power amplifier. For flexibility each amplifier should have its own control system and power supplies. Figure 1 shows a typical 2 MW ICRH amplifier system.
2.1. RF AMPLIFIER CIRCUIT DESCRIPTION

This amplifier circuit is an example of a three tube set-up using water cooled ceramic tetrodes in all stages. The first stage, the preliminary amplifier, is driven by a solid state wide band amplifier. It can provide up to 5.5 kW and is in grounded cathode configuration. The tube is a Siemens SK 1054 K. This tube does not require any neutralisation because of its small interelectrode capacitances. This preliminary stage is followed by the driver stage equipped with an Eimac 4 CW 150000 E tube. This tube is connected in grounded grid and can deliver up to 170 kW to the final power amplifier. The final power amplifier uses a THOMSON-CSF CQK 650-2 and is also configured as a grounded grid stage. Driver and final power amplifier are built as coaxial circuits. The preliminary amplifier consists of concentrated elements. All three stages are tuneable over the whole frequency range and are equipped with tuning motors. A preset system allows fast tuning to a number of frequencies.

2.2. POWER SUPPLIES

The plate power supply for the preliminary amplifier and also all the supplies for the control and screen grids are built into the amplifier casing. The plate power supplies for the driver and the final power amplifier are set up as separate units and can be conventional SCR controlled units or PSM type DC sources.

2.3. FINAL POWER AMPLIFIER TUBE LOSS

The tube loss of a tetrode operating in class B is greatly dependent on the sustaining voltage (Minimum plate voltage at conducting state of the tube). It is a well known fact that operating a tetrode with a too low sustaining voltage will result in excessive screen grid current and hence overloading in terms of screen grid dissipation. If the load is not constant, the plate voltage has to be large enough to operate the tube at the highest occurring load impedance with the required minimum sustaining voltage. This is to prevent any overloading of the screen grid.

The plate loss with a pure resistive load becomes:

\[ P_{\text{dam}} = \frac{P_{\text{out}} \cdot 4}{\pi} \left( \frac{U_p}{U_{de}} \right) - P_{\text{out}} \]

\[ U_p/U_{de} \] is the voltage utilising factor \( X_i \)

For minimum loss \( X_i \) must be as high as possible. This implies that the plate supply voltage \( U_{de} \) must be chosen carefully to obtain good efficiency.

2.3.1. Tube DC loss with constant plate voltage

Figure 2 shows the calculated plate loss for a typical tetrode operated with a constant plate voltage at 2500 kW output power. The plate voltage has been chosen so that the tube can deliver the requested power into the load with a SWR of 1 : 1.5. Because of the increasing sustaining voltage with decreasing load impedance, the tube plate loss is much higher at lower impedances (on the left side of the Smith chart). Assuming a maximum permissible plate loss of 1300 kW for this CQK 650-2 tetrode, it can be seen from Figure 2 that the tube can only operate safely in a small part of the area with a SWR of 1:1.5.

2.3.2. Tube DC loss with adapted plate voltage

If the plate voltage is regulated so that the tube works with a constant sustaining voltage, the plate loss is greatly reduced. This adaptation of the plate voltage can be done by operating the tube with a regulated supply. The criteria for controlling the plate voltage may be the screen current [2]. Figure 3 shows the plate loss at constant sustaining voltage for a tetrode delivering an output power of 2500 kW into a load with a SWR of 1:1.5. It can be seen that in the whole area with this SWR the loss does not exceed 1300 kW. The loss is highest at the points with a high imaginary part because of the inevitable phase shift between RF output voltage and the plate current.

2.3.3. RF losses

In addition to the DC tube losses considered above, it is important to take into account the RF losses, which lead to additional limitations when frequency, power and/or duty increase. Such limitations have also been investigated as shown in chapter 4.2.
2.4. SYSTEM WITH FAST REGULATING PLATE POWER SUPPLY

The use of a regulated plate power supply allows a tetrode to deliver much more power into a changing load because of reduced plate loss. If full power is to be delivered also during short pulses or during fast load changes then a fast adaption of the plate voltage is required. A fast regulating power supply is best realised as a PSM (Pulse Step Modulator) type DC source which is described in chapter 3.
2.5. FINAL POWER AMPLIFIER IN CLASS C

In the past ICRF amplifiers have for several reasons been mostly operated in class B. Class B amplifiers have linear transfer characteristics and therefore the regulation of the output power can easily be done by varying the input power. With class B the necessary cathode current the tube must deliver is smaller than with class C. Also the tube electrode loading is relatively small. On the other side class C offers a much better efficiency. The use of a regulated power supply with fast controllable output voltage would allow class C operation of the RF final power amplifier. The regulation of the output power would have to be done by varying the plate voltage. Better stability of the amplifier in case of increased reflection (due to ELM's and other spontaneous changes in the plasma) can be expected. The demands on the tetrode in respect to electrode loading and peak cathode current are however higher in class C.

3. PSM POWER SUPPLY

The PSM type DC source consists of a number of independent pulsed series connected DC supplies. Each contains IGBT's as switching devices and delivers an output voltage of about 1000 V. The control of this switching modules is done by means of a DSP through optical fibers. A typical supply can deliver up to 30 kV at 150 A. The voltage can be switched on and off within a few microseconds and may be modulated with frequencies of up to 10 kHz. A crowbar for the protection of the load is not required. Increasing the output voltage to 100 kV with more modules is possible. Systems with 80 kV output voltage have already been built [3]. Figure 4 shows a block diagram.

3.1. PSM MAIN FEATURES

The PSM system offers some interesting features, such as:
- inbuilt redundancy; in case of a missing module only the peak voltage capability is limited but operation can continue.
- fast switch off capability; no crowbar system is required to protect the load from excessive stored energy in case of a load short circuit.

- DC amplification; the output voltage of the PSM supply follows any reference voltage from DC up to a frequency of several kHz. The PSM supply functions as a MW DC amplifier.

3.2. APPLICATION OF THE PSM SUPPLY

The PSM supply has initially been developed for the broadcast industry. More than 100 amplitude modulated high power transmitters with carrier powers up to 1 MW (4 MW peak power) are in operation world-wide. But because of the special features a PSM supply offers also many advantages if used to supply DC power to tetrode amplifiers, gyratrons, klystrons or neutral beam sources.
4. THOMSON GRIDDED TUBES IN FUSION

4.1. INTRODUCTION

Thomson tetrodes are already fully operative in a number of ICRF heating systems of world leading machines, such as JET, TORE SUPRA, TEXTOR, ASDEX, TFTR, etc. Other systems using this tetrodes are presently being commissioned.

As the scientists require for the new projects higher and higher performances, THOMSON TUBES ELECTRONIQUES has invested in a full test facility, which allows a better knowledge of the products behavior and control of the technologies. This facility includes:

- a 6 MW HV power supply
- one RF amplifier, similar to those delivered by TTE for TORE SUPRA, able to deliver over 2 MW RF power, covering the 35-80 MHz frequency range. It can also be tuned at 120 MHz.

Until now, ICRF heating systems have been operating in long pulses (several seconds to several tens of seconds) at relatively moderate duty rates, typically $\leq 12\%$. In order to prepare for the future cw RF power requirements, TTE has explored the limits of the tetrodes in such cw conditions. Tests done at 57 MHz have evidenced an operational limit around 1.5 MW at that frequency; above this limit, these tests show that the reliability would not be sufficient for operational use, and that the key factor in this limitation is RF losses.

4.2. RF LOSSES

Considering a resonant coaxial line loaded by a capacitance, in the present case the tube end capacitance (see figure 5), the reactive current value depends on this capacitance value : as a consequence, losses in the lower part of the screen grid (base side) are higher if the capacitance is higher.
In a conventional tetrode, we will try to minimize this capacitance, but it shall remain impossible to avoid it completely. The reactive currents may also be reduced by decreasing the anode load impedance: but this has a limitation with the peak cathode current necessary to obtain the full output power. This problem gets even worse if it is also required to sustain some VSWR and to operate in class C for a better efficiency. Increasing the cathode size to produce the higher currents results either in a higher tube end capacitance or a higher tube length : in both cases, we again increase the RF losses. From figure 5C, we may infer that if the maximum voltage (antinode) is situated at the center of the active zone of the tube, the amplitude of the reactive current, and consequently the RF losses, shall be lower. This remark leads to the concept of the DIACRODE® : anode and screen grid having external connections at both ends of the tube, an appropriate resonant circuit allows to place the voltage antinode at the center of the active zone (figure 6) : the reactive currents amplitude being lower, the RF losses at each end of the screen grid are also lower. With this DIACRODE, the extra margin obtained on RF losses allows to increase the tube length for higher currents required.

4.3. TUBE ANODE DISSIPATION

In order to widen the possible operating range for full power with high VSWR at various phases TTE has developed a new anode design, which allows much higher anode dissipation. The TH 525A and TH 526A versions of fusion tetrodes fitted with this new anode offer a 2 MW cw rated dissipation without a significant increase in the water flow. They have been fully tested, and tubes have been already delivered to the fusion community.

4.4. CONCLUSION

The conventional tetrodes have now reached their limits. Tests at 57 MHz conducted by TTE have shown that, around that frequency, 1.5 MW will be the maximum cw RF power which can be obtained while keeping the reliability needed for operational use.
- This limitation is mainly due to RF losses.
- TTE, in order to achieve higher performances, is working on a different design, the DIACRODE®.

This concept has already been implemented in an 80 kW peak (vision) UHF TV tube. For scientific applications, after technological tests with a preliminary model, TTE develops presently a high power diacrode aimed to the particle accelerators 200 MHz RF requirement of the very near future. Full validation of the technology at that frequency shall allow extrapolation to a higher power tube in the ICR frequencies range.

4.5 STATUS OF THOMSON 2 MW CLASS TETRODES

Table 1 shows the proven performance of today available 2 MW class tetrodes. The maximum ratings shown on the left side are the design values of this devices. Tested values are measured results in the TTE test facility. Field operation values cited are the most demanding ones among those presently achieved with this tubes operating fusion machines. All the tested and field operation values are sets of data. This means that power, frequency, pulse length actually have been reached together. The data with a * are obtained with a VSWR of 1.5 : 1. For instance, the 80 MHz operational data (TORE SUPRA) are obtained with the following parameters:

<table>
<thead>
<tr>
<th>Tube type</th>
<th>TH 526</th>
<th>TH 535</th>
<th>TH 561</th>
</tr>
</thead>
<tbody>
<tr>
<td>RF output power</td>
<td>2 MW(*)</td>
<td>60 kW</td>
<td>3 kW</td>
</tr>
<tr>
<td>Anode dc voltage</td>
<td>23 kV</td>
<td>11 kV</td>
<td>4 kV</td>
</tr>
<tr>
<td>Anode dc current</td>
<td>124 A</td>
<td>9.5 A</td>
<td>2.5 A</td>
</tr>
<tr>
<td>G2 dc voltage</td>
<td>1500 V</td>
<td>800 V</td>
<td>600 V</td>
</tr>
<tr>
<td>G1 dc voltage</td>
<td>-290 V</td>
<td>-110 V</td>
<td>-70 V</td>
</tr>
<tr>
<td>RF input power</td>
<td>60 kW</td>
<td>3 kW</td>
<td>100 W</td>
</tr>
</tbody>
</table>

*: with VSWR = 1.5 : 1

5. MICROWAVE TUBES

5.1 LOWER HYBRID FREQUENCIES

The first experiments to study plasma heating at lower hybrid frequencies and current drive phenomena were performed with klystrons derived from tubes developed for other purposes (radar, TV,...), such as the TH 2086 (1.3 GHz - 600 kW - 0.5 s). To increase the energy transmitted to the plasma and for extensive use of the RF current drive, the development and the manufacturing of...
## 2 MW CLASS THOMSON TETRODES FOR FUSION

### NEW HIGH ANODE DISSIPATION VERSIONS OF TH 525/TH 526

<table>
<thead>
<tr>
<th>Model</th>
<th>Max. Rating (Continuous)</th>
<th>Tested Values</th>
</tr>
</thead>
<tbody>
<tr>
<td>TH 525A</td>
<td>2.0 MW</td>
<td>2.5 MW (15 MIN)</td>
</tr>
<tr>
<td>TH 525A</td>
<td>2.0 MW</td>
<td>2 MW (30 MIN)</td>
</tr>
</tbody>
</table>

### MAXIMUM RATINGS

<table>
<thead>
<tr>
<th>Frequency</th>
<th>Power (Peak)</th>
<th>Pulses</th>
<th>Duty</th>
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<td>2.5 MW</td>
<td>30 s</td>
<td>15%</td>
</tr>
<tr>
<td>80 MHz</td>
<td>2.2 MW</td>
<td>30 s</td>
<td>15%</td>
</tr>
<tr>
<td>120 MHz</td>
<td>1.5 MW</td>
<td>30 s</td>
<td>15%</td>
</tr>
<tr>
<td>120 MHz</td>
<td>1.8 MW</td>
<td>5 s</td>
<td>15%</td>
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### TESTED VALUES

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<th>Power (Peak)</th>
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<th>Duty</th>
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<tr>
<td>55 MHz</td>
<td>2.7 MW</td>
<td>20 s</td>
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</tr>
<tr>
<td>80 MHz</td>
<td>2.2 MW</td>
<td>50 s</td>
<td>10%</td>
</tr>
<tr>
<td>80 MHz</td>
<td>2 MW</td>
<td>210 s</td>
<td>-</td>
</tr>
<tr>
<td>120 MHz</td>
<td>1.2 MW</td>
<td>30 s</td>
<td>12%</td>
</tr>
<tr>
<td>120 MHz</td>
<td>1.4 MW</td>
<td>30 s</td>
<td>-</td>
</tr>
<tr>
<td>80 MHz</td>
<td>2 MW</td>
<td>10 s</td>
<td>-</td>
</tr>
<tr>
<td>120 MHz</td>
<td>1.2 MW</td>
<td>10 s</td>
<td>-</td>
</tr>
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</table>

### PRESENT FIELD OPERATION VALUES

<table>
<thead>
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<th>Frequency</th>
<th>Power (Peak)</th>
<th>Pulses</th>
<th>Duty</th>
</tr>
</thead>
<tbody>
<tr>
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<td>12%</td>
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<tr>
<td>80 MHz</td>
<td>2 MW</td>
<td>30 s</td>
<td>12%</td>
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</table>

### CW

<table>
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<th>Frequency</th>
<th>Power (Peak)</th>
<th>Pulses</th>
<th>Duty</th>
</tr>
</thead>
<tbody>
<tr>
<td>40 MHz</td>
<td>1.8 MW</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>70 MHz</td>
<td>1.5 MW</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>55 MHz</td>
<td>950 kW</td>
<td>20 min</td>
<td>-</td>
</tr>
<tr>
<td>57 MHz</td>
<td>1.6 MW</td>
<td>30 min</td>
<td>-</td>
</tr>
</tbody>
</table>

### ANODE DISSIPATION CAPABILITY

<table>
<thead>
<tr>
<th>Model</th>
<th>Max. Rating (Continuous)</th>
<th>Tested Values</th>
</tr>
</thead>
<tbody>
<tr>
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</tr>
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<td>2.0 MW</td>
<td>2 MW (30 MIN)</td>
</tr>
</tbody>
</table>

Table 1
specific klystrons were necessary with the following requirements: RF frequency in upper S-band or above; large RF unit power per source; quasi-steady-state or cw operation.

The TH 2103 [8], which has been specially developed for the plasma heating at lower hybrid frequencies, responds to these objectives with the following main characteristics: 3.7 GHz - 500 kW (beam 60 kV x 18.5 A) - 60 s or cw. In another version, it operates at 650 kW (beam 60 kV x 24 A) - 10 s. An overcoupled output cavity is required for operation on a mismatched load. Two beryllium oxide pillbox windows (half wavelength) offer safety operating margins: max. temperature rise at center of ceramic is 23°C at 500 kW cw. A recombining or magic tee feeds the user port through a single output flange.

Moreover, a recent redesign, which mainly eliminates the modulating anode, should allow very soon a new TH 2103C to deliver 700 kW in quasi-steady-state on a load having a VSWR = 1.4 : 1, with 70 kV and 24 A.

Still in the LHRF range, but at 8 GHz [9], six complete systems have been designed, which include each: one TH 1504 gyrotron, its focusing magnet, all special RF components (TE51 bend, TE51 to TE01 mode converter, two mode filters, a bidirectional TE01 selective coupler and finally a coupler-divider ending by 12 rectangular WR 137 waveguides). All have been manufactured, tested and delivered for FTU. The purpose of this experiment is to drive a high current in the thermonuclear plasma.

These systems are now under implementation on the tokamak. A power higher than 850 kW has been measured at the output of the 12 arm coupler, the power generated by the gyrotron being then higher than 1.0 MW. The overall efficiency is about 41% (available RF power at the output of the final coupler over the gyrotron beam power), while the gyrotron interaction efficiency is greater than 46%. On a VSWR in the order of 2 : 1, the output power is still above 800 kW. Runs with a VSWR of 4 : 1 are still possible.

The TH 1504 gyrotron is characterized by a 82 kV x 26 A electron gun which includes a modulating anode at Vak = 50 kV. The room temperature electromagnet, fed by several power supplies, provides a 3100 gauss magnetic induction.

5.2. ELECTRON CYCLOTRON RESONANCE FREQUENCIES

In the range of the electron cyclotron frequencies TTE has developed several gyrotrons such as:

- TH 1503: 100 GHz - 200 kW in TE34 -beam 80 kV x 9 A - 100 ms
- TH 1505 [10]: 110 GHz - 350 kW in TE 64 -beam 85 kV x 17 A - 60 s

Another model is now under development:

- TH 1506 [6][7]: 118 GHz - 500 kW in TE 22.6 -beam 82 kV x 21 A - 210 s

This gyrotron has a lateral output in HE11 mode.

All these fast wave tubes use cryogenic magnets and the electron guns are fitted with a modulating anode (Vak approx. 20 to 30 kV). A sweeping coil (10 Hz) around the collector helps the power dissipation by using all the internal collector surface.

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Superconducting magnet development at KAERI

G. W. Hong (KAERI)
빈 편
Abstract

Superconducting magnet have been considered as the best tool to realize the fusion reactor having the concept of energy generation. KAERI is proceeding the basic research about replacing the PF-6 coil in KT-2 by a superconducting coil. As the first step, a precise superconducting magnet of 4 Tesla was designed and fabricated. In spite of small scale, many kind of technics were needed such as a design of field, a shimming of field, and a jointing of superconducting wire. After correcting the field, the maximum field inhomogeneity along the z-axis within 5 cm was below 10 ppm, and the field instability was lower than 0.1 ppm/hour.

1. Introduction

The confinement of high temperature plasma by high magnetic field is one of the key technology in developing nuclear fusion, and the necessity of superconducting magnet increases as the feasibility and applicability of the nuclear fusion increases. The use of the normal electric magnet using normal conductor such as copper is limited because of its high electric power consumption exceeding generating electricity and impossibility of continuous operation. The tokamak type thermonuclear fusion reactor is regarded as the most feasible reactor type and the effectiveness of superconducting magnet is increased as the size of the reactor increases.

Superconductor shows perfect electrical conductivity below its critical transition temperature(Tc). The Tc of the conventional metal-base superconductor is in the range of 2 - 20 K and it is very difficult to keep the system at such a low temperature. Therefore the application technology of the superconductor were closely related to the development of cryogenic engineering technology. The most popular superconductors used for fabricating high field superconducting magnet are NbTi and Nb3Sn. In order to generate very high magnetic field using electromagnet made with normal conducting material such as copper, it is needed to supply very high electricity and large amount of coolant water to dissipate the resistive heat generation in normal conductor. Also it is very difficult to obtain stable magnetic field. But in case of superconducting magnet, relatively small electricity is required to cool the magnet below its critical temperature and for initial magnetization of the coil. Therefore the superconducting magnet is regarded as only choice for future commercial fusion reactor which can produce more electric power output than input electricity.

Korea atomic energy research institute has been performing basic study to develop the
design and fabrication technics of poloidal magnet coil which can be used in KT-2 reactor and verifying the effectiveness of superconducting poloidal coil. Fourteen PF coils are designed to be used in KT-2 reactor for plasma generation and magnetic field regulation. The first target of this study is to prepare PF-6 coil of Fig.1 with superconducting magnet and checking the feasibility of replacing normal PF-6 coil with prepared superconducting coil [1].

A small superconducting magnet with the central field intensity of 4 T was fabricated by superconductivity research team of KAERI for getting fundamental technologies of design and fabrication of superconducting magnet. The techniques of magnetic field compensation, quench protection, superconductor jointing and basic cryogenic operation were established through the fabrication of that magnet.

2. Study on the superconducting magnet in Korea

The research on development of device applying superconducting magnet was initiated in late eighties. A research group in Seoul National University (SNU) fabricated a small scale superconducting magnetic energy storage system under the financial support from Korea Electric Power Company (KEPCO). But the discovery of high Tc superconductor activated the ministry’s interest on the superconductivity and government supporting program on HTSC was organized. Such a research effort on HTSC encourages the research on the low temperature superconductor application also.

KAERI has the experiences of designing and fabricating the low critical temperature superconducting magnet for NMR (Nuclear Magnetic Resonance). Both size and capacity of the magnet are chosen to be suitable for laboratory purpose. During the fabrication magnet, the techniques of the field shimming and the superconducting jointing improving the field homogeneity and stability were developed. KEPCO has carried out the project of developing the 0.5 MJ SMES (Superconducting Magnetic Energy Storage) for power stabilization, whose main subjects were the evaluation of the magnet coil and high speed power converter. KEPCO has also made a base of developing a electric generator using the superconducting coil. KERI (Korea Electrotechnology Research Institute) has been fabricating a NbTi superconducting wire for NMR system, which is considered as a fundamental field in superconducting industry. Furthermore, study on superconducting transformer and high field magnet is being performed. KERI has a lot of experiences about many kinds of superconducting magnets. The upper mentioned projects and researches are in the fundamental stage, and the dependence on foreign technics are rather high compared with other conventional industry. Therefore, an international co-work structure for technologies and informations exchanges is necessary.

Recently, the design concepts and fabrication technics for small scale experimental fusion reactors employing the superconducting magnet are developed in a few countries. But in case of large scale of system, technics for whose systems fabrication are not completed yet, and the design of wire and the cryogenic system are still the major research items. Since the coils developed in domestic institutes were rather small scale, the liquid helium dipping bath type cryostat was enough to cool the coils. And commercial ready-made superconducting wires were more intensive in winding the coils. If the magnet system is large such as the poloidal field coils for KT-2, studies on the conservative design of wires for considering the a.c. loss and the effective cooling method of superconducting are quite needed.
3. Status of superconducting poloidal field coil development

The superconducting toroidal coils for experimental fusion reactors (ITER, FER) is about 12 Tesla of magnetic field intensity, and that of the poloidal coil is about 7-13 Tesla, and the diameters are about 3-20 m. Refrigerators of 30KW capacity are expected to cool the coils. Figure 2 shows the status of development of pulsed poloidal coils fabricated for experimental reactors. A lot of studies are being performed in Japan and U.S.A [2,3]. Japan had many experiences and reported useful results through carrying out the project of Demo Poloidal Coil-U1 and U2. Those results are very useful for designing and fabricating the superconducting poloidal coil in KT-2. The results of US-DPC and the DPC-EX using Nb$_3$Sn coils are also important. But, until now there seems no experience to install them into a fusion reactor, but evaluation of coils in test cryostat were carried out. The conduit type of conductors are generally used for poloidal coils, and a forced circulation of liquid helium through the conductor is used to cool the coils. Field intensity of the poloidal field coil planned to be fixed outside of the KT-2 reactor is about a few hundred gauss. Since the diameter of the coil is as large as about 4-5m, intensive studies on coil cooling will be necessary in spite of low field intensity.

4. Fabrication of precise field magnet

A laboratory scale superconducting magnet was developed in KAERI to prepare the basis of developing the coils for fusion reactor. Technics for the design and correction of the magnetic field were established through the fabrication of the magnet. Though the scale was small compared with the experimental fusion reactor, many kind of technics were necessary for design and fabrication, whose concepts are considered almost similar with the large scale experimental fusion reactor.

The technical specification of fabricated superconducting magnet is listed in table 1. Figure 3 shows the schematic structure of the magnet, which is composed of three main coils, movable two mechanical shim coils, and magnetic shimming coil assemblies. The mechanical shim coils are fixed with screws, then the position can be adjusted to make precise field distribution after preliminary test of the magnet. A cylindrical bobbin of z and z$^2$ term field shimming coils is located outside of the mechanical shim coils. The plate shape of x and y term shim coils were assembled to the outside of the z and z$^2$ term shim coil correcting the radial direction field of magnet. Specifications of used superconducting wires are listed in table 2.

4.1 Fabrication and evaluation of the superconducting coils

The superconducting wires were wound round the bobbins made of glass fiber reinforced plastic by a winding machine modified with a commercial lathe. The size and shape of bobbins were designed with the analytical solution formulas [4,5]. During the winding, epoxy impregnated cloth were inserted on each layer for the electrical insulation. This is known as a proper method in case of laboratory scale coils, because the mistakes which may be happened during the winding can be corrected.
The electrical circuit excepting the shim coils of the magnet system is shown in Fig 4. In the preliminary test, 4.01 Tesla was obtained at the center of magnet with operating current of 139 A. The field inhomogeneity measured by hall probe are shown in Fig 5. The maximum inhomogeneity was about 180 ppm, and the second order term of the field governed the shape of field inhomogeneity. The preliminary test was performed without low resistance superconducting joint in order to adjust the mechanical shim coils and to assemble the magnetic shim coils after the test.

4.2 Assembling and evaluation with shim coils

After preliminary test, radial and axial field shim coils were designed and fabricated to obtain the required field homogeneity. The design concepts of main coils and shim coils may be almost similar with the toroidal coil and poloidal coils, respectively. Axial field shim coils (z and z² term) were wound round cylindrical bobbin calculated with analytical formular. And radial field shim coils (x and y term) were wound round the plate bobbin as shown in Fig 6 [6].

Superconducting switches were fabricated for each shim coil for persistent current mode operation. And superconducting jointing of wire was performed to improve the field stability in persistent current mode. Figure 7 shows the completely assembled magnet coil. Magnetic field of the assembled superconducting coil was measured with a NMR probe at room temperature using inverted cyostat, and the transport current for each shim coil was adjusted to make high magnetic field homogeneity. The sensitivity of the NMR probe was about 0.3 ppm. The persistent current mode of magnet was checked by operating the each superconducting coil in sequence. First, main coil set was energized and made as the persistent current mode. The current of each shim coil was adjusted to have high field homogeneity with monitoring the field change. The finally corrected field inhomogeneity along the z-axis within 5cm was below 10 ppm as shown in Fig. 8. The maximum field inhomogeneity along the circumference of 1.7 cm radius was also below 10ppm as shown in Fig.9. Figure 10 shows that the field instability was lower than 0.1ppm/hour, which means that the low resistance superconducting joint of wire had been performed successfully.

5. Conclusion

Superconducting magnet have been considered as the best tool to realize the fusion reactor having the concept of energy generation. KAERI is proceeding the basic research about replacing the PF-6 coil in KT-2 by a superconducting coil. As the first step, a precise superconducting magnet of 4 Tesla was designed and fabricated. In spite of small scale, many kind of technics were needed such as a design of field, a shimming of field, and a jointing of superconducting wire. After correcting the field, the maximum field inhomogeneity along the z-axis within 5 cm was below 10 ppm, and the field instability was lower than 0.1 ppm/hour.

The plan of design and fabrication of PF coil is a good opportunity to enhance a vitality of related cryogenic industry, since the large scale application needs the interests of many industrialists. After this project, the possibility of fabricating the next fusion reactor employing superconducting field generating coils will increase.
References

Table 1. Technical specifications for precise superconducting magnet.

<table>
<thead>
<tr>
<th>Specification</th>
<th>Value</th>
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<tr>
<td>Center field</td>
<td>4 Tesla</td>
</tr>
<tr>
<td>Field inhomogeneity</td>
<td>10 ppm at 50 mm dsv</td>
</tr>
<tr>
<td>Field instability</td>
<td>0.1 ppm/hour</td>
</tr>
<tr>
<td>Operating current</td>
<td>130 Ampere</td>
</tr>
<tr>
<td>Shim coils</td>
<td>S/C coil 4 set, Z, Z^2, X, Y term</td>
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</table>
Table 2 Specification of superconducting wire used in precise superconducting magnet.

<table>
<thead>
<tr>
<th>location</th>
<th>wire diameter</th>
<th>No of filaments</th>
<th>critical amperes at 7, 6, 5 T</th>
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<tbody>
<tr>
<td>main coil (3 parts)</td>
<td>1 mm</td>
<td>60</td>
<td>408 518 630</td>
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<tr>
<td></td>
<td></td>
<td>36</td>
<td>230 282 342</td>
</tr>
<tr>
<td>end compensate coil</td>
<td>0.85 mm</td>
<td>210</td>
<td>330 430 530</td>
</tr>
<tr>
<td>shim coil</td>
<td>0.3 mm</td>
<td>18</td>
<td>36 47 57</td>
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</table>
Fig 1 The KT-2 Poloidal field magnet system configuration.
Fig 2 Pulsed coils for development of superconducting poloidal coils.
Fig 3 Schematic drawing of field generating main coils and end-side compensate coils of precise superconducting magnet system.
Connection of wires.

Fig 4 Schematic circuit diagram of main magnet system without shim coils.
Fig 5 The magnetic field deviations measured by hall probe along the Z axis in 4.02 T.
Fig 6 X term shim coils and a superconducting switch.
Fig 7 Assembled magnet system for the operating test.
Fig 8 The field deviations measured by NMR probe along the Z axis in persistent mode of 4.04T corrected by shim coils.
Fig 9 The field deviations measured by NMR probe along circumferential direction in persistent mode of 4.04T corrected by shim coils. Measuring point are on the line of Zc=10cm, R=1.7cm.
Fig 10 The change of magnetic field deviation with time.
<table>
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<th>INIS 주제코드</th>
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<td>KAERI/TR-515/95</td>
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제목 / 부제 | 토카막 최적설계 및 성능예측을 위한 방법론 연구

연구책임자 및 부서명 | 김 성 규 (핵융합로 연구 분야)

연구 자 및 부서명 | 김성규, 이강현, 인상렬, 황철규, 홍봉근, 홍계원(핵융합로 연구 분야)

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보고서종류 | 기술보고서

연구위탁기관 |

계약 번호 |

초록 (300단어 내외)

이 논문집은 차세대 토카막연구를 위한 제1회 대덕 국제핵융합 심포지움에서 발표된 16편의 연구논문을 수록하였다.

주제명 키워드 (10단어 내외)

대덕 국제핵융합 심포지움, 차세대 토카막, 핵융합, KT-2 토카막
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