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Annual Report of Naka Fusion Research Establishment
from April 1, 2004 to March 31, 2005

Naka Fusion Research Establishment
Japan Atomic Energy Research Institute
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This annual report provides an overview of research and development (R&D) activities at Naka Fusion Research Establishment during the period from 1 April, 2004 to 31 March, 2005, including those performed in collaboration with other research establishments of JAERI, research institutes, and universities.

In the JT-60 research program, the pulse length of the tokamak discharge was extended successfully up to 65 s in FY 2003. In FY 2004, following the successful results, optimization of long pulse discharges was continued in order to explore the boundaries of facility capabilities for the long pulse operation. The pulse length of the negative-ion based neutral beam injection system has reached up to 25 s with an injection power of 1MW. In the electron cyclotron wave system, the pulse length has also extended up to 45 s with an RF power of 0.35 MW by using four gyrotrons in a series operation. Sustainment of higher normalized $\beta$ of $\beta_n > 2.3$ for 22.3 s, or $\beta_n > 2.5$ for 15.5 s has been achieved by exploiting available plasma heating systems. This discharge exhibits not only the high $\beta_n$, but also high confinement improvement with the H factor of $H_{99} = 1.9 - 2.3$ and high normalized fusion performance of $G = H_{99} \beta_n / q_{95} = 0.4 - 0.5$ during the sustainment, where $q_{95}$ is a safety factor at the edge. $G > 0.4$ corresponds to the fusion energy gain of $Q = 10$ for the ITER standard scenario. The H-mode plasma with $H_{99}, n_{e} > 1.4$ has been maintained for about 30 s, although degradation of the performance was observed at the later half of the discharge. In the reversed shear plasmas, the operation regime was successfully extended to the density higher than Greenwald density limit, while maintaining high confinement and high radiation loss fraction by tailoring the internal transport barriers of the density and temperature. Demonstration of neoclassical tearing mode stabilization and improvement of plasma performance in the high beta region ($\beta_n > 3$) has been performed using local current drive by the second harmonic electron cyclotron waves. In addition, a real-time control system of safety factor profile has been developed. This system enables spatial control of driven current by adjusting the parallel refractive index of lower-hybrid waves through the change of phase difference between multi-junction launcher modules.

The design of National Centralized Tokamak (NCT), which is the superconducting modification of JT-60, progressed both in physics and engineering. Machine has been designed to have a wide-range capability of operation in aspect ratio and plasma shape. Engineering design of the main components of superconducting toroidal and poloidal magnetic field coils, vacuum vessel, in-vessel components, and cryostat has been performed to investigate the structure optimization from viewpoints of manufacturing processes, operation and maintenance feasibility.

A series of the experimental programs on the JFT-2M were completed in FY 2003. In FY 2004, experimental data on the Advanced Material Tokamak Experiment (AMTEX) using the reduced
activation ferritic steel (F82H), high performance experiment, characteristics of SOL and divertor plasma and compact toroid injection for fueling have been analyzed and evaluated. Concerning the AMTEX, analysis of high-β experiments with the Ferritic Inside Wall (FIW) facing close to the plasma have shown a wall stabilization effect. By using an MHD equilibrium code, it has been confirmed that the plasma with $\beta_N \approx 3.5$ is compatible with FIW.

In the theoretical and analytical researches, significant progress was made in the studies of transport simulation of current hole plasma, role of low order rational $q$-values in the ITB events, the theory of Alfvén eigenmodes in tokamaks, current spike behavior of disruptive plasma, and stability of external MHD modes. In the project of numerical experiment of tokamak (NEXT), the studies of the structure formations in toroidal electron temperature gradient driven turbulence, control of the zonal flow, and formation of current hole also progressed.

R&Ds of fusion reactor technologies have been carried out both to further improve technologies necessary for ITER construction, and to accumulate technological database to assure the design of fusion DEMO plants. For the design optimization of ITER superconducting magnets, degradation of critical current performances of the Nb$_3$Sn conductors has been experimentally and numerically examined and a new simulation model has been developed to predict degradation behavior in a large current superconductor. For ITER Neutral Beam Injector, MeV-range accelerator R&D is being in progress and the current density has been extended to 100 A/m$^2$. For the further pulse extension and power increase of 170 GHz gyrotron, a built-in radiator at the mode converter has been optimized to improve the efficiency of gyrotron output power and to reduce stray radiation, and pre-program controls of a cathode heater power has been employed to stabilize the beam current and the output power. In the R&Ds on Plasma Facing Components, a screw tube has been developed as a possible option for the ITER divertor. For the design of ITER Test Blanket Module (TBM), two candidates, namely Water Cooled Solid Breeder TBM and Helium Cooled Solid Breeder TBM have been proposed, and elementary technology R&Ds have been progressed for fabrication of the TBM, thermo-mechanical properties of the packed bed, and irradiation technologies. An outline design of an electrochemical hydrogen pump has been carried out as a candidate of the advanced Blanket Tritium Recovery system. Using DT neutrons, neutronics integral experiments have been performed with a blanket mockup at FNS facility to predict the tritium breeding ratio with an error less than 5%. As one of the most promising structural materials for the ITER TBM and DEMO blankets, F82H has been investigated with its neutron irradiation effects using HFIR, JMTR, and so on. In the IFMIF program, transitional activities have been continued.

In the ITER Program, along the work plan approved on June 2004 under the framework of the ITER Transitional Arrangements, the Design and R&D Tasks have been carried out by the Participant Teams. In FY 2004 JAERI has been in charge of fifty-five Design Tasks that make the implementation of preparing the procurement documents for facilities and equipments that are scheduled to be ordered at an early stage of ITER construction. The site issues have been continuously discussed among the delegations of six parties/area and through the bilateral negotiation between Japan and the EU based on a viewpoint of “a broader approach” concept.

Finally, in the fusion reactor design studies, the conceptual design of the fusion DEMO plant which is placed beyond ITER has progressed. Three options with different capabilities of center solenoid (CS) coil are studied. Researches on the physics related to the ramp up CS-less reactor, and waste management have progressed.

Keywords; JAERI, Fusion Research, JT-60, JFT-2M, Fusion Technology, ITER, Fusion Power Demonstration Plants, Fusion Reactor
日本原子力研究所（平成16年度）

（2005年8月12日 受理）

日本原子力研究所（原研）那珂研究所における平成16年度（2004年4月1日～2005年3月31日）の研究開発活動について、原研所内他研究所および所外の研究機関並びに大学との協力により実施された研究開発を含めて報告する。

JT-60による研究開発では、平成15年度に65秒の長時間放電に成功し、その成果を受けて、平成16年度は、長時間運転における機器性能の限界を探求するため、長時間放電の最適化を進めた。負イオン中性粒子入射装置では1MWで25秒までパルス幅を伸長し、電子サイクロトロン波装置では4本のジャイロトンを連続に用いて0.35MWのパワーを45秒まで入射することができた。加熱装置のパルス幅の伸長により、規格化ベータ値、β_n>2.3の場合22.3秒、またはβ_n>2.5の場合15.5秒という高い規格化ベータ値を長時間維持することに成功した。この放電では、高いβ_nだけでなく、高い閉じ込め改善度（H_{bo}=1.9-2.3）および高い規格化核融合性能（G=H_{bo}/β_n/q_{95}^2=0.4-0.5）を達成した。q_{95}はプラズマ端の安全係数である。G～0.4の条件はITERの標準運転である核融合エネルギー増倍率Q=10に相当する。更に、閉じ込め改善度がH_{bo}～1.4で、30秒の長時間放電が達成された。但し、プラズマ性能の劣化が放電の後半に観測された。また、内部輸送障壁をもつ負磁気シア先進運動では、電子温度および密度の内部輸送障壁を工夫して生成し、グリーンラウドの密度限界以上の高密度で、高閉じ込めと高放射率を実現した。高い規格化ベータ値（β_n〜3）の領域で、電子サイクロトロン波の2倍高調波による局所電流駆動を用いて、新古典テアリングモードの抑制とプラズマ性能の改善を実証した。更に、低域領域の波の多分岐ランナーの位相差を変化させ、磁場と平行方向の屈折率を調節することにより駆動電流の空間位置を制御する、安全係数の実時間制御システムを開発した。

JT-60を超伝導化改造で国内重点化装置の物理と工学設計を進めた。装置はアスペクト比とプラズマ形状に対して幅広い運転領域を可能とする設計とした。製作過程、運転、保守の観点から、超伝導コイル、真空容器、クライオスタットの主要コンポネントの最適化検討のための工学設計を実施した。

JFT-2Mによる一連の実験研究を平成15年度に完了し、平成16年度では、低放射化フェライト鋼（HELL）を用いた先進材料プラズマ試験（AMTEK）、高性能試験、ダイバータ/SOLおよび燃料注入のためのコンクレートトロイド入射に関する実験データの解析と評価を行ってみた。AMTEKに関して、フェライト鋼壁をプラズマに近接した場合における高ベータ実験の解析結果は、壁によるプラズマの安定化を示した。改良したMHD平衡コードを用いて、プラズマと壁の距離が近接している場合、β_n〜3.5までの高ベータプラズマが得られることを確認した。

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編集者：山本 巧、佐藤 正泰、工藤 祐介、洲 亘、吉田 英俊
理論および解析研究では、電流ホールおよびアルファバーイ固有モードの理論、ディスラプションプラズマの電流スパイク現象の振舞などについての研究を進めた。トマク数値実験プロジェクト(NEXT)では、トロイダルモード形成の解明や帯状流の制御と電流ホール形成の解明を進めた。

核融合炉工学の研究開発は、ITER の建設に必要な技術の更なる改良、および発電実証プラントの設計に必要な技術データベースの蓄積という二つの目的に向けて推進された。ITER 用超伝導マグネット設計の最適化に向けて、Nb₃Sn 導体における臨界電流密度の低下現象を実験および解析の画面に調べ、それを含む解析モデルを開発した。ITER 用中性子注入における技術開発では、MeV 級の高電流密度ビーム加速器の研究開発を着実に推進し、電流密度を 100 A/m² まで拡張した。170 GHz ジャイロトロンのパルス化および高出力化に向けて、モード変換器における放射器の改良を進め、出力効率の向上および不要放射の削減を図った。更に、長パルス運転時におけるビーム電流の安定化を図るため、陰極ヒーターの電力調整にプログラム制御を採用した。プラズマ対向機器の開発においては、ITER ダイバータ冷却用スクリュー管を開発し、高温、高圧条件における限界熱流束特性データを取得した。ITER テストプラント・モジュールの二つの候補として、水冷却式固体増殖テストプラント・モジュールとヘリウム冷却式液体増殖テストプラント・モジュールを提案し、その設計作業を進めるとともに、製作性、熱機械特性、照射技術に関する要件技術開発を進めめた。プラントで生成されたトリチウムを回収するための進先技術としての電気化学ポンプの概念設計を行った。更に、5%未満の誤差でトリチウム増殖比を評価するために、プラントに関する中性子工学実験を進めめた。また、ITER テストプラント・モジュールおよび発電実証プラント用プランケットの構造材料として最有力な F82H の耐中性子照射試験を HFIR, JMTR などの施設で実施した。IFMIF 計画に関しては移行期活動を継続した。

ITER 計画においては、ITER 移行措置の枠組みのもとで 2004 年 6 月に承認された作業計画に沿って、参加国チーム合同で設計および R&D タスク作業が本格化した。原研は 2004 年度、建設の早い段階において発注が予定される設備と機器に関して、日本が担当して調達書類の準備を行う 55 件の設計タスク作業を進めた。この間引き続き、サイト合意に向けた協議が、「幅広いアプローチ」という考え方に基づいて、六国／地域および日欧間で進められた。

最後に、核融合炉設計研究では、ITER の後の発電実証プラントの概念設計を進めた。中心ソレノイド(CS)コイルの機能に応じた三つのオプションについて設計検討を行った。CS なし炉の電流立ち上げに関する物理研究および廃棄物の評価を進めた。
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FOREWORD

This report presents the results from research and development activities at Naka Fusion Research Establishment of Japan Atomic Energy Research Institute (JAERI) during the period from 1 April 2004 to 31 March 2005.

Research on the JT-60 has made a remarkable progress in expanding the plasma discharge regimes, in which higher confinement and higher beta were simultaneously attained at longer pulses. Extended capability of plasma heating systems enabled to accomplish these discharge regimes. H-mode discharges were also extended up to 30 s. In the long pulse H-mode discharges, degradation of the plasma performance was observed in the later half of the discharge time. The degradation is attributable to the increase in the wall recycling. The design studies on the modification of JT-60 for a superconducting machine has progressed in physics and engineering in collaboration with universities and industries. With respect to the research on JFT-2M, significant outcomes resulted from analyzing and evaluating the experimental data obtained in FY 2003. Steady progress was also made in theoretical and analytical researches and the project of numerical experiment of tokamak, NEXT.

R&Ds on fusion reactor technologies have been carried out to further improve technologies necessary for ITER construction, and to accumulate technological database to assure the design of fusion DEMO plants. For the design optimization of ITER superconducting magnets, the critical current performance of the Nb3Sn conductors was carefully re-examined. With respect to ITER Test Blanket Module (TBM), technology R&Ds have progressed for their fabrication, thermo-mechanical properties of the packed bed and irradiation technologies. Reduced activation ferritic steel F82H is one of the most promising structural materials for the ITER TBM and DEMO blankets. Neutron irradiation on F82H has been continued using HFIR and JMTR. JAERI has been participating in the transition activities on the International Fusion Materials Irradiation Facilities, with the expectation of starting the Engineering Validation and Engineering Design Activities in the near future.

As the Japanese implementing institute of ITER Transitional Arrangements, JAERI has performed fifty-five Design Tasks in FY 2004 that contribute to the preparation of the procurement documents for facilities and equipments.

It should be noted that the collaborations with other research establishments of JAERI, research institutes, and universities have played an important role in performing the fusion R&Ds in Naka Fusion Research Establishment. In view of these encouraging outcomes, I am expecting that the fusion research and developments will make a new and powerful take off towards the energy source for mankind.

It is our pleasure that the ITER construction site was finally decided in Cadarache. We will do our best for the success of the Project and we are prepared for that.

関昌弘
Masahiro Seki
Director General
Naka Fusion Research Establishment, JAERI
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I. JT-60 PROGRAM

Objectives of the JT-60 project are to contribute to physics R&D of the International Thermonuclear Experimental Reactor (ITER), and to establish the physics basis for the steady state tokamak fusion reactor.

In the fiscal year of 2004, the reactor-relevant performance progressed much with the collaboration with the universities and institutes.

This part is divided into the three chapters; Experimental Results and Analyses (Chap.1), Operation and Machine Improvements (Chap.2) and Design Progress of the National Centralized Tokamak Facility (Chap.3).

1. Experimental Results and Analyses

1.1 Long Pulse Operation and Extended Plasma Regimes

Following the successful results in the long pulse experiments in FY2003, in order to explore the boundaries in the long pulse operation capabilities of the facility, optimization of long pulse discharges was continued. Extension of the heating and current drive pulses has contributed to the long pulse experiments. The pulse length of N-NB has reached up to 25 s with injection power of 1 MW. The ECRF pulse has reached 44.6 s. The input energy reached 15 MJ using four Gyrotrons in series (44.6 s) or three Gyrotrons in parallel (14.8 s). A grill mouth made of carbon was newly placed at the LHRF launcher. Conditioning was undergoing, with the maximum injection power 1.3 MW (about 60% of the previous value) and the input energy of 16.3 MJ (before carbon mouth placed it was 10.9 MJ).

1.1.1 Sustainment of High Normalized Beta Value $\beta_n$

For economical fusion reactors, steady sustainment of high $\beta_n$ is required, since fusion power density is proportional to the square of $\beta_n$. We reported, in FY2003, that $\beta_n=1.9$ was sustained for 24s and that the current profile reached in steady state. In 2004, we have achieved sustainment of much higher $\beta_n > 2.3$ for 22.3s, or $\beta_n > 2.5$ for 15.5s [1.1-1], exploiting available P-NB, N-NB, and ECRF systems. Figure 1.1.1-1 shows waveforms of a high $\beta_p$ ELMy H-mode discharge (E043903) at $I_p=0.9MA$ and $B_t=1.7T$ ($q_95=3.4$) where $\beta_n=2.5$ was sustained for 15.5s. This discharge exhibits not only the high $\beta_n$, but also high confinement improvement $H_{99}=1.9-2.3$ and high value of $G=H_{99} q_95^2 = 0.4-0.5$ during the sustainment. The index G is a measure of the fusion gain [1.1-2]; G=0.4 corresponds to ITER standard scenario of Q=10.

Current and pressure profiles at initial phase were optimized, adjusting NB deposition profile, in order to avoid a neo-classical tearing mode (NTM) that limits attainable $\beta_n$. There was no significant electromagnetic instability ($n=0-3$) observed by saddle loops.

The current density profile reached a steady state (Fig. 1.1.1-1). In the discharge, the resistive diffusion time, defined by $\tau_R=(1/12) c_0 \alpha_{\rho 0}^{-1} n_0^{-1}$, was 1.6s, therefore the sustained period (15.5s) corresponds to 9.5$\tau_R$. Here, we used a formulation of $\tau_R$ by Mikkelsen [1.1-3].

![Fig. 1.1.1-1. Temporal evolutions of $\beta_n$, injection power (P-NB, N-NB, EC), line averaged electron density, $D_e$ intensity, H-factor, and $H_{99} q_95^2$. Sustained duration of $\beta_n=2.5$ reached 15.5s, during which $H_{99}$ and $H_{99} q_95^2$ were 1.9-2.3 and 0.4-0.5, respectively. Contour plot of current density profile (each 0.1MA/m$^2$ interval) evaluated using equilibrium reconstructions with MSE is shown in the bottom, showing the current profile reached steady state.](image)
the above equation, \( a \) is averaged minor radius and \( \sigma_{Na} \) is the neoclassical conductivity. Although we intended to increase particle exhaust rate by adjusting strike points near the pumping slot at the divertor, the intensity of the \( D_a \) emission continuously increased (Fig. 1.1.1-1) even without gas-puffing. The increased recycling raised line averaged electron density continuously from \( 1.6 \times 10^{19} \text{m}^{-3} \) to \( 2.1 \times 10^{19} \text{m}^{-3} \), corresponding to Greenwald density fraction of 46% and 58%, respectively. During \( t = 7 \text{ to } 20 \text{s} \), the electron density increased by 22% and \( H_{99\%} \) degraded by 15%.

1.1.2 Long Pulse High Recycling H-mode

Research in JT-60U has been expanded to never-explored-regime towards the steady-state operation owing to extension of the pulse length to 65 s and NB heating pulse length to 30 s \([1.1-4]\). In 30s-ELMy-H-mode discharges, wall saturation, which is defined as zero wall-pumping rate, or constant wall inventory, was observed by particle balance analysis \([1.1-5]\).

Figure 1.1.1-2 shows waveforms of the 30s-ELMy H-mode discharge in which the wall saturation was observed. At a toroidal magnetic field of 2.6 T, the flat-top of a plasma current (1.0 MA) was maintained for 34 s and the divertor configuration was kept for 36 s. The positive ion source based neutral beam (P-NB) was injected for 30 s with a heating power of 7-12 MW and the negative one for 25 s with 0.6-1.5 MW. The line-averaged electron density was controlled at 66% of the Greenwald density by the feedback control system of a gas-puffing rate. As Figure 1.1.1-2(d) shows, the wall inventory, invoked from the following particle balance equation,

\[
\int_{0}^{t} \left[ \Gamma_{\text{rev}}(\rho) + \Gamma_{\text{wet}}(\rho) \right] d\tau = N_{\text{plasma}}(\rho) + \int_{0}^{t} \left[ \Gamma_{\text{pump}}(\rho) + \Gamma_{\text{wall}}(\rho) \right] d\tau
\]

increases until \( t = 19 \text{ s} \). This increase means that the wall-pumping was effective because particles continued to be retained in the wall during the increase. After \( t = 19 \text{ s} \), on the contrary, the wall inventory is constant. Since the local particle-releasing and wall-pumping cannot be investigated from the above particle balance equation, this constant wall inventory does not directly indicate that the inventory of all the tiles is saturated but indicates that the wall neither pumps nor releases particles on balance. The outer divertor tiles are considered to release particles because of the increase in the surface temperature as shown in Fig.

Fig. 1.1.1-2 Waveforms of a long pulse, ELM/ H-mode plasma. (a) the plasma current \( I_p \), the line-averaged electron density \( < n_e > \), (b) the positive and the negative ion source based neutral beam heating power \( P_{NB} \) as well as the particle-fueling rate, \( \Gamma_p \) and \( \Gamma_{\text{NB}} \) respectively, (c) the gas-puffing rate \( \Gamma_{\text{gas}} \), the divertor-pumping rate \( \Gamma_{\text{wall}} \), (d) the numbers of injected particles, plasma particles, pumped particles and retained in the surface materials, (e) the temperature around the outer and the inner strike point measured by thermocouples, (f) \( D_a \) emission intensity from the divertor plasma, (g) plasma effective charge \( Z_{\text{eff}} \), and H-factor \( H_{99\%} \). The periods of H-mode, the wall saturation and MARFE are shown at the top of this figure. The shaded area indicates the H-mode period under the condition of the wall saturation. 1.1.1-2(e) while other tiles with the surface temperature still low are considered to continue pumping particles. Hence, this situation is interpreted as net wall saturation. This wall saturation was sustained until \( t = 28 \text{ s} \). Even under the condition of the wall saturation, the ELMy H-mode plasma, indicated by the ELM activity of \( D_a \) shown in Fig. 1.1.1-2(f), with a constant plasma effective charge \( Z_{\text{eff}} \sim 3 \) and a constant H-factor \( H_{99\%} \sim 1.7 \) was sustained as shown in Fig. 1.1.1-2(g).

At \( t = 28 \text{ s} \), the outer divertor plasma detached, resulting in an X-point MARFE. At this detachment, the particle flux to the outer divertor tiles decreased, and the wall inventory started to decrease as shown in Fig. 1.1.1-2 (d). From this observation, this decrease of
the particle flux to the outer divertor tiles is considered to result in the particle release (probably from the outer divertor tiles), or the decrease in the wall inventory. This situation is similar to the dynamic retention process: particles that retained in the wall during a plasma exposure are released when the plasma exposure is ended.

1.1.3 Extension of JT-60U Pulse Length
The 30 s H-mode plasma has been optimized up to \( I_p = 1.4 \) MA as shown in Fig.I.1.1-3 [1.1-4]. Although, H-mode was maintained throughout the heating period (\(-30\) s), the performance was weakened at the later half of the discharge, due to some beam faults including N-NB. However, \( H_{\text{NB}} \sim 1.4 \) was maintained for about 30 s. It should be noted that similar to the other long pulse discharges, the wall recycling increased in the later half of the discharge and might affected the performance. Before the beam faults, \( H_{\text{NB}} \sim 1.8 \) was maintained for about 14-15 s. Following the results in the last year, these new results have shown progress towards standard H-mode operation in ITER.

Fig.I.1.1-3, Typical waveforms of a 1.4 MA 30 s ELM My H-mode discharge.

1.2 Enhanced Performance and Steady State Research
1.2.1 Sustainment of weak shear plasma in nearly full CD [1.2-1]
The high \( \beta_p \) H-mode plasmas in JT-60U are characterized by a monotonic safety factor (\( q \)) profile with weak magnetic shear owing to the bootstrap current based on the internal transport barrier formation. Such a weak shear configuration is compatible with the ITER steady-state operation with \( Q > 5 \). The high \( \beta_p \) H-mode plasmas were optimized towards ITER steady-state operation scenario [1.2-2]. One of the key issues for obtaining a high-performance high \( \beta_p \) H-mode plasma is suppression of NTMs. Since the NTMs are destabilized at rational surfaces, the scenario for the avoidance of 3/2 NTM is considered that the \( q = 1.5 \) surface is removed from whole plasma region.

Fig.I.1.2-1 Typical waveforms of weak shear plasma with \( q_{\text{min}} \sim 1.5 \): (a) injection power of P-NB and N-NB, (b) normalized \( \beta_p \) (solid curve) and poloidal beta (\( \beta_p^* \) dotted curve), (c) line averaged electron density, (d) loop voltage, (e) deuterium recycling emission at the divertor. (f) Time evolution of \( q \) profile.
Typical waveforms of such a scenario are shown in Fig.1.1.2-1, where \( I_p=1 \) MA, \( B_t=2.4 \) T, \( R=3.35 \) m, \( a=0.8 \) m, \( \kappa=1.44 \), \( \delta=0.5 \), \( q_{95}=4.5 \). In this plasma, \( \beta_N \sim 2.4 \) (\( \beta_p \sim 1.7 \)) has been sustained for 5.8s. This duration corresponds to \( \sim 26\tau_0 \) and \( \sim 2.8\tau_0 \), which was limited by the pulse length of N-NB (\( \sim 4 \) MW, \( \sim 6.5 \) s). The \( H_{95}=2.2 \) and \( H_{H95y}=1.0 \) were obtained at \( t=8.3s \). The electron density was almost kept constant at 54% of the Greenwald density. Loop voltage was reduced to near zero (\( \sim 0.075V \)), which indicates the nearly full non-inductive current drive condition. It should be emphasized that no NTM was observed in this discharge by the optimization of \( q \) profile. For avoidance of NTM, the alignment of the local pressure gradient and rational surfaces (such as \( q=1.5, 2 \)) is important. The temporal evolution of \( q \) profile is shown in Fig.1.1.2-1(f). Pressure and \( q \) profiles were optimized as follows by feedback control of the stored energy and the injection timing of NBs. P-NB was injected at \( t=5.6s \) before the full penetration of inductive current when the shape of \( q \) profile is monotonic and \( q > 1.5 \) in the whole plasma region. During the initial phase of P-NB heating (\( t=5.6-7.0s \)), \( \beta_N \) was gradually raised to \( \sim 2.0 \) by feedback control of stored energy in order to expand the location of \( q=2 \) surface by the evolution of bootstrap current. At \( t=7.0s \), the shape of \( q \) profile was flattened and \( q=2 \) surface moved outward and \( q>1.5 \) in the whole plasma region due to the evolution of bootstrap current around off-axis region, where the \( q \) profile in the core region becomes slightly reversed. After the flattening of \( q \) profile, N-NB was injected to enhance the non-inductive current drive and to increase \( \beta_N \) from \( \sim 2.0 \) to \( \sim 2.4 \). At the later phase of the discharge, \( q \) profile was similar to that at \( t=7.0s \), but slightly decreased. The minimum value of \( q \) was kept \( \sim 1.5 \) and \( q=2 \) surface located at small temperature gradient region, then no NTM was observed. The change in the shape of \( q \) profile is small by N-NB injection, which indicates that the inductive current before N-NB injection could be replaced with the beam driven current by N-NB. The analysis of non-inductive current drive indicates that \( f_{95}=50-43\% \) and \( f_{90}=52-47\% \) were obtained, which indicates nearly full non-inductive current drive condition. The values of \( \beta_N, f_N \) and \( q_{95} \) are close to requirements for the ITER steady-state operation scenario.

### 1.2.2 Compatibility of an Advanced Tokamak Plasma with High Density and High Radiation Loss Operation [1.2-3]

Advanced tokamak plasmas with an internal transport barrier (ITB) have advantages of compatibility with high bootstrap current fraction and high confinement, which are essential for the steady-state operation. In order to apply these plasmas to fusion reactors, compatibility with high density and high radiation loss is also required for attaining high fusion power and reducing heat load localized onto the divertor plates. In a fusion reactor, high density operation above the Greenwald density \( n_{GW} \) is preferable and the radiation loss fraction of about 0.9 is necessary to reduce the heat load onto the divertor plates sufficiently.

![Fig. 1.1.2-2 (a) HH_{95} and (b) radiation loss fraction as a function of n_i/n_{GW}. Squares : RS plasma. Circles : high \( \beta_p \) H-mode plasma. Diamonds : ELMy H-mode plasma. Closed and open symbols show new (during 2003-2004) and old (before 2002) data. Double lines show the data with impurity seeding.](image)

In the reversed shear (RS) plasmas, the operation regime was successfully extended to high density above \( n_{GW} \) with high confinement (\( HH_{95y}>1 \)) and high radiation loss fraction (\( f_{95}>0.9 \)) by tailoring the density and temperature ITBs as shown in Fig. 1.1.2-2. In these plasmas, a large volume configuration (\( V_p=75-80 \) m\(^3\)) with a small outer gap (\( \Delta=0.08-0.16 \) m) between the plasma and the outside wall was used with the plasma current of \( I_p=1.0 \) MA, the toroidal magnetic field of \( B_t=2.5-2.9 \) T and the safety factor at the 95% flux surface of \( q_{95}=5.8-6.5 \). The high confinement of \( HH_{95y}=1.3 \) was obtained in the high density region above \( n_{GW} \) with NB fuelling only. In this plasma, the
high $\bar{n}_e/\bar{n}_{GW}$ was obtained owing to the peaked density profile inside the ITB, although the pedestal density was smaller than 0.4$\bar{n}_{GW}$. With Ne seeding, the total radiation loss was enhanced to a level greater than 90% of the net heating power with high confinement of $HH_{H}=1.1$ at $\bar{n}_e/\bar{n}_{GW}=1.1$. Without Ne seeding, high radiation loss fraction was also obtained. However, in these discharges, the radiation from the main plasma was enhanced.

In the large volume RS plasmas, Cu XXVI line (111.20 Å) emission was observed, and its intensity decreased with increasing the outer gap. The fast ion loss induced by a large toroidal ripple could be related to generation of Cu. The Cu line intensity was almost zero at $\Delta=0.4$ m and the radiation in the core plasma was much smaller than that with small outer gap ($\Delta=0.08-0.15$ m). The main plasma radiation loss was changed by scanning the outer gap for the investigation of effects of metal impurity accumulation on the confinement. With $\Delta=0.08$ m, the strong impurity accumulation was observed and radiation in the main plasma reached up to 80% of the heating power. Even with such a large radiation in the main plasma, high confinement of $HH_{H}=1.2$ was sustained at $\bar{n}_e/\bar{n}_{GW}=1$. The confinement degradation with the large radiation loss in the main plasma was not observed in the RS plasmas. The ITB seems to be robust for radiative cooling in the core plasma.

For the understanding of the mechanism responsible for the sustainment of high confinement with high main plasma radiation at high density, the relationship between central density and central temperature was examined. With the small outer gap, the central density increased rather than the central temperature during the confinement improvement. The central temperature was higher with the large outer gap ($\Delta=0.4$ m) than with the small outer gap ($\Delta=0.15$ m) at the same central density. The reduction in the central temperature was compensated with the increase in the central density. The NB heating profile became off-axis with the small outer gap. Off-axis heating, radiative cooling in the core plasma and NB fueling could be responsible for relationship between central density and central temperature.

In order to form the dense divertor, heat flux to the divertor plasma is necessary and the radiation should be enhanced in the divertor plasma. In the RS plasma, Ar accumulation inside the ITB has been observed [1.2-2]. Thus, the Ne seeding was applied to enhance the divertor radiation. The plasma configuration with $\Delta=0.15$ m was used to increase the heat flux to the divertor by reducing the main plasma radiation from Cu. Ne was puffed from the divertor region together with D$_2$ gas-puffing from the plasma top. The ratio of the divertor radiation to the total radiation increased from 20% without Ne seeding to 40% with Ne seeding. However, the radiation from the main plasma was still larger than that from the divertor plasma, although the edge density was increased with D$_2$ gas-puffing for suppressing the impurity penetration. The total radiation reached up to 90% of the absorbed heating power. The Ne radiation profile estimated using 1-D impurity transport code indicated the small contribution of Ne to the main plasma radiation. Metal impurity Cu could largely contribute to the radiation loss in the core plasma. However, the radiation loss from the main plasma and neutron yield rate were almost kept constant, indicating metal impurity did not lead catastrophic confinement degradation and fuel dilution.

1.2.3 Comparison of Electron Transport in Helical and Tokamak Devices [1.2-5, 1.2-6]

The plasmas with an electron internal transport barrier (ITB), which is characterized by peaked electron temperature profiles, are obtained in the JT-60U tokamak and in the Large Helical Device (LHD), when the ECH is focused on the magnetic axis. The maximum values of R/L$_{Te}$, where R is the major radius and L$_{Te}$ is the scale length of the electron temperature gradient, are similar ($R/L_{Te}$ is 3 -5 in the L-mode plasmas and 20 - 30 in the electron ITB plasmas both in LHD and JT-60U). However, there are differences in the mechanism that trigger the ITB formation between LHD and JT-60U plasmas. The transition of the radial electric field triggers the formation of an ITB in LHD, while the negative magnetic shear is an important parameter in the formation of the electron ITB in JT-60U. The difference in the trigger mechanism results in the difference in the dependence of the temperature gradient, R/L$_{Te}$, on the heating power as shown in Fig. 1.1.2-3. There is a clear transition of the R/L$_{Te}$ in the formation of the LHD electron ITB and the transition is associated with the transition from ion root (weak
negative radial electric field) to electron root (large positive electric field) in the collisionless regime $v_T < 0.3$, without a change in magnetic shear. On the other hand, the formation of the electron ITB is gradual in JT-60U, since the electron ITB needs the change in magnetic shear (positive or negative shear), which changes on the time scale of current diffusion.

Fig. 1.1.2-3 The normalized electron temperature gradient, $R/L_{De}$, as a function of ECH power normalized by electron density, $P_{ECH}/n_e$, in LHD (closed circles and solid line) and JT-60U (open circles and dashed line).

In order to obtain a comprehensive understanding of the non-linearity of heat transport in toroidal devices, transient transport experiments (cold or heat pulse) are performed in plasmas on LHD and JT-60U without ITB. The dependence of electron heat diffusivity, $\chi_e$, on electron temperature, $T_e$, and its gradient, $\nabla T_e$, is analyzed by an empirical non-linear heat transport model ($\chi_e \propto T_e^{\alpha} |\nabla T_e|^\beta$). The heat diffusivity, $\chi_{eb}$, obtained in LHD with $R_{eq}=3.5$ m from the transient analysis based on the empirical non-linear transport model is shown in Fig. 1.1.2-4(a). The heat diffusivity estimated by power balance analysis, $\chi_{pb}$, is also shown in Fig. 1.1.2-4(a). The small difference between $\chi_{eb}$ and $\chi_{pb}$ indicates a weak $\nabla T_e$ dependence of $\chi_e$ ($\beta<<1$) in LHD. On the contrary, a gyro-Bohm like $T_e$ dependence ($\alpha=3/2-5/2$) is obtained. In JT-60U, the short pulse ECH is injected at $\rho=0.6$ for transient transport analysis. In order to compare the $\nabla T_e$ dependence of $\chi_e$ with the critical gradient length model, both $\chi_e$ and $\chi_{pb}$ normalized by $T_e^{3/2}$ are plotted as a function of $R/L_{Te}$ in Fig. 1.1.2-4(b). The dependence of $\chi_{eb}$ on $R/L_{Te}$ seems to be changed at $R/L_{Te}=6-8$ i.e. the temperature gradient-driven mode may be switched on above this value. The value of $\chi_e$ has the $\nabla T_e$ dependence and $\chi_{pb}$ seems to be enhanced from $\chi_{pb}$ at $R/L_{Te}=6-8$. The $\nabla T_e$ dependence factor $\beta$ decreases from 3 to 1.6 with the increase in $R/L_{Te}$ while the $T_e$ dependence factor $\alpha=0.5-2$ is not different from that obtained in the LHD plasma. For the stabilization of microturbulence, the local shear is a critical parameter. The influence of local and global shear on turbulence might be one of the candidates to explain the difference in the non-linearity between LHD and JT-60U plasmas.

Fig. 1.1.2-4 (a) Radial profiles of $\chi_e$ and $\chi_{pb}$ in LHD. (b) $R/L_{Te}$ dependence of $\chi_e$ and $\chi_{pb}$ normalized by $T_e^{3/2}$ in JT-60U.

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1.2-3 Takenaga, H. et al., in Proc. 20th Int. Conf. on Fusion Energy 2004 (Vilamoura, 2004).
1.2-6 Inagaki, S., et al., in Proc. 20th Int. Conf. on Fusion Energy 2004 (Vilamoura, 2004).
1.3 MHD Instabilities and Control
1.3.1 Stabilization of the Neoclassical Tearing Mode
In obtaining a stationary high-beta plasma with positive magnetic shear, suppression of neoclassical tearing modes (NTMs), which appear below the ideal MHD limit, is the most critical issue. In JT-60U, two scenarios have been developed for NTM suppression: (a) NTM avoidance by modification of pressure and current profiles, and (b) NTM stabilization with electron cyclotron current drive (ECCD) / electron cyclotron heating (ECH). In 2004, these scenarios have been applied to a high-beta plasmas with $\beta_n$=2.9-3 [1.3-1, 2, 3].

(1) NTM avoidance with low $q_{95}$
Experiments in JT-60U demonstrated that NTM can be avoided by adjusting the mode rational surface (typically $q$=1.5 and 2 surfaces) at the location with low pressure gradient, that is, low local bootstrap current. It can be considered that the mode rational surfaces can be located at peripheral region with low pressure gradient if the safety factor is reduced. To demonstrate the feasibility of this scenario, operations at very low-q regime were performed.

![Fig. 1.1.3-1](image)

Fig. 1.1.3-1 Typical discharge of a stationary high-beta plasma with NTM avoidance. (a) $\beta_N$ and NB power, (b) $q_{95}$ and (c) frequency spectrum of magnetic perturbations.

Typical discharge is shown in Fig.1.1.3-1, where $I_p=1$ MA and $B_t=1.7$ T. In this discharge, safety factor is decreased to ~2.2 during NB injection by lowering the plasma height and as a result decreasing plasma minor radius. The value of $\beta_N$ is kept at 3 without NTMs for 6.2 s (4.2$\tau_T$). Owing to the high-$\beta_N$ operation at low-$q$ regime, high value of $\beta_p$=2.4% is kept stationary. It is notable that no sawtooth is observed by any available diagnostics. MSE diagnostic also shows that the central factor is not below 1, and that very flat $q$-profile is sustained for ~6 s. Although internal inductance $\ell$ decreases to ~0.7 due to the flat $q$-profile resulting in $\beta_n/\ell$~4, no major disruption is observed.

(2) NTM stabilization with ECCD in $\beta_n$~3 regime
Another approach to suppress NTMs is to stabilize them by localized current drive and heating at the magnetic island with EC wave. In 2004, demonstration of NTM stabilization and improvement of plasma performance in high beta region ($\beta_n$=3) has been performed using the second harmonic X-mode ECCD. Typical discharge is shown in Fig.1.1.3-2, where plasma parameters are as follows: $I_p=0.85$ MA, $B_t=1.7$ T, $q_{95}=3.5$. In this discharges, beta value is gradually increased by feedback control on NB power to avoid a 2/1 mode which degrades the plasma performance significantly. At t=4 s, $\beta_N$ reaches 2.9, and a 3/2 mode appears. As the mode grows, NB power is increased to sustain a given beta value. At t=5.8 s, beta value begins to decrease due to further degradation of confinement. After the unmodulated EC wave injection of ~2.4 MW from t=6 s, beta value begins to increase until it reaches the target value of the feedback. At t=7.8 s, $\beta_N$~2.9 is sustained with smaller amount of NB power, which shows improvement in confinement. Actually, $H_{HPL}$ increases to 1.8 after the stabilization, resulting in successful sustainment of a high-beta and high-confinement plasma with $\beta_n H_{HPL}/q_{95}^2$~0.4 (Fig.1.1.3-2(c)).

![Fig. 1.1.3-2](image)

Fig. 1.1.3-2 Typical discharge of a stationary high-beta plasma with NTM stabilization. (a) $\beta_N$ and neutron emission rate, (b) NB and EC wave power and (c) $\beta_n H_{HPL}/q_{95}^2$.

1.3.2 MHD Activities in a Low Beta RS Discharge
A reversed shear (RS) plasma is expected as a
discharge of advanced scenario of ITER because it has good confinement and a large bootstrap current fraction. It is understood that the disruption at \( q_{\text{surf}} \geq 2 \), \( \beta_N > 2 \) is caused by stability limit of \( n=1 \) ideal kink ballooning mode. However RS plasmas with a strong ITB disrupt frequently even at lower \( \beta_N \). By now, low beta disruptions are explained by double tearing mode [1.3-4] or resistive interchange mode [1.3-5], and these are MHD instabilities at the \( q_{\text{min}} \) surface and around the ITB. However these cannot explain all of the observed low beta disruptions. To understand the cause of the low beta disruption, we investigated the MHD instabilities of RS plasmas with a strong ITB and a central flat pressure by measuring plasma current profiles and MHD fluctuations. We observed two types of disruptions. One is the disruption without precursor at \( q_{\text{surf}} > \) integer. The other is the disruption with \( n = 1 \) precursor. The poloidal mode number of the \( n=1 \) mode is equal to the \( q \) value of the outermost rational surface \( (q=\text{integer}) \). The \( n=1 \) modes exist continuously from the peripheral region to the ITB layer or separately at peripheral region and at ITB and the phase difference is 180 degree between them as shown in Fig. I.1.3-3.

Fig. I.1.3-3 Coherence and phase difference of the \( n=1 \) modes.

From the MSE measurement \( q_{\text{eff}} \) is not below 4 in any case and \( q_{\text{min}} \) scatters and is significantly below 2 at disruptions \( q_{\text{surf}} \leq 4 \) as shown in Fig. I.1.3-4. Moreover, disruption frequently occurs around \( q_{\text{surf}} = 5 \) and 4. These imply that disruptions depend on \( q_{\text{surf}} \) rather than \( q_{\text{min}} \). To explain these characteristics of disruption, we introduce a simple model that, disruption occurs when the both MHD instabilities at the plasma surface and at an inner rational surface with the safety factor being equal to the surface mode are unstable [1.3-6]. This simple model can explain almost all observed disruptions by two processes. One is that the surface mode triggers disruption, which occurs when \( q_{\text{surf}} \) changes, and the corresponding \( q \) surface at ITB layer changes discretely. The other is that the internal mode triggered disruption, which occurs when the mode at an inner rational surface become unstable gradually. In the former case, the mode number of surface mode changes to a next integer as surface \( q \) changes and relative location of ITB and the rational surface in the RS region changes discretely. For instance, when the surface mode changes form \( m=5 \) to \( m=4 \), the corresponding internal rational surface changes discretely. Disruptions take place when the pressure gradient around \( q=4 \) is very large because of ITB, while instability of \( q=5 \) is stable due to central flat pressure profile. If the ITB is far from the internal \( q=4 \) surface, disruption does not occur. The \( m/n=4/1 \) mode changes to \( 3/1 \) mode, when the surface \( q \) decreases below 4. The \( q=3 \) surface in the RS region is always in the large \(-dT/d\rho \) region, therefore surface \( q \) cannot be below 4 any longer. In the latter case, disruption occurs when the pressure gradient increases or relative position of internal rational surface and ITB changes gradually. For example, the plasma with \( 4 < q_{\text{eff}} < 5 \) is stable when the \( q=4 \) surface in the RS region exists in central \( T_1 \) plateau region. After the \( q=4 \) surface moves to the large \(-dT/d\rho \) region, internal mode becomes unstable and leads to disruption. This disruption triggered by the internal mode can explain disruptions observed when \( \beta_N \) is decreasing.

Fig. I.1.3-4 Surface safety factor \( (q_{\text{surf}} and q_{\text{surf}}) \) and \( q_{\text{min}} \) at disruption of various \( \beta_N \).

I.3.3 Disruption study
(1) Fast Discharge Shutdown
It is shown that disruption deleterious effects on plasma facing components of a tokamak device can be greatly reduced or avoided by simultaneous puffing of small
amounts of high-Z noble gases, particularly, krypton and large amounts of hydrogen gas. A high electron density caused by the intense hydrogen gas puffing amplified the radiation of High-Z atoms. In turn the stored energy was radiated and plasma was terminated quickly. The high electron density and high effective charge made by high-Z species prevents runaway electron generation. [1.3-7]

(2) Mitigation of Post-Disruption Runaways
A clear deposition of impurity neon ice pellets in a post-disruption runaway plasma was observed, where most of plasma current was driven by runaway electrons. A high normalized electron density was stably obtained with $n_e^\text{nor}/n_{e0}^\text{GW} \leq 2.2$. Effects of prompt exhaust of runaway electrons and reduction of runaway plasma current were found (Fig.1.1.3-5). One possible explanation for the basic behavior of runaway plasma current is that it follows the balance of avalanche generation of runaway electrons and slowing down predicted by the Andersson-Helander model, including the combined effect of collisional pitch angle scattering and synchrotron radiation. It was suggested that the impurity pellet injection reduced the energy of runaway electrons in a step wise manner. [1.3-8]

References

1.4 H-mode and Pedestal Research
1.4.1 Dimensionless Pedestal Identity Experiments in JT-60U and JET in ELM My H-Mode Plasmas [1.4-1]

Dimensionless identity experiments in JT-60U and JET, aimed at the comparison of the plasma pedestal characteristics and ELM behavior in the two devices, were continued in this year. The method chosen for this study is the "dimensionless identity technique", based on the invariance of plasma physics to changes of dimensional parameters, e.g. $n_e$, $T_e$ at the pedestal, when the dimensionless plasma parameters are conserved (normalised plasma pressure $\beta$, safety factor $q$, Larmor radius $\rho^*$ and collisionality $v^*$). In contrast to other dimensionless comparison experiments, the similar size of JET and JT-60U results in dimensionless matched plasmas that are also very similar in their dimensional parameters, with the exception of the major radius.

The comparison of the pedestal profiles of dimensionless matched JET and JT-60U H-modes may help to gain further insight in the underlying physics mechanisms determining the similarities and difference in the H-mode characteristics. Figure 1.1.4-1 shows a
Fig. 1.1.4-1. Pedestal profiles comparison for the two $q_{95}=5.1$ nearly matched JET/JP-60U discharge pairs. (a) Experimental $T_e$ profiles for JET/JP-60U, E43065 (JP-60U, PNB) and JET #60849, while Fig. 1.1.4-1(b) shows the pedestal $n_e$ profiles for the same pair in Fig. 1.1.4-1(a). As for the $T_e$ profile, a reasonable match is obtained for $T_e$ at the pedestal top, pedestal width and gradients. The picture is quite different for the pedestal densities as shown in Fig. 1.1.4-1(b). The density pedestal of JET is much higher and wider than that of JP-60U. A comparison of the edge density gradient $V_{ne}$ is not straightforward, since $n_e(r)$ of these JET discharges is measured with good space resolution only over a part of the pedestal density gradient region.

A major difference between JET and JP-60U identity configuration is the $B_t$ ripple, $-0.1$% in JET compared to $-1.2$% in JP-60U at outer midplane. The associated fast ion losses in JP-60U are substantial, of the order of several MW for the plasma investigated. The resulting edge electric field may provide a counter-rotation source at the plasma edge sufficient for JP-60U plasmas to counter-rotate even for net positive parallel momentum injection. Experiments where a large fraction of perpendicular PNB were substituted by co-NNB gave conflicting results: at low $q$, no significant increase of the pedestal pressure was observed, in contrast to the high $q$ plasmas, where the use of NNB resulted in the highest pedestal pressures. In this case, the pedestal density obtained in JET and JP-60U are quite similar (within 15%) as shown in figure 1.1.4-1(c). In both $q$ cases, though, the plasma toroidal rotation changed in a similar way ($V_T$ is less negative). The reasons for the different behavior of the pedestal pressure at low and high $q$, as well as that for the improved performance at high $q$, are not yet understood.

As for the reason of smaller pedestal performance in JP-60U, two possible mechanisms have been considered. One is the effect of toroidal rotation on the pedestal performance. The other is that the ripple may have a direct effect on the thermal ion transport. In order to separate both effects in the experiments, further experiments are planned in both devices, an installation of ferritic steel in JP-60U to reduce the ripple and ripple enhancement experiment on JET by changing toroidal field coil currents for every other coil.

1.4.2 Impact of Toroidal Rotation on ELM Behavior [1.4-2]

The ability to actively change the plasma rotation using a combination of the tangential and perpendicular NBs in JP-60U has aided the efforts to determine the effects of counter rotation on the ELM characteristics in grassy ELM regime ($q_{95}=-4.9$ and $\delta=-0.59$) and type I ELM regime ($q_{95}=-4.1$ and $\delta=-0.28$), in terms of accessibility and controllability of small ELM regimes.

In JP-60U, the edge plasma near the top of the temperature pedestal rotated in the counter (CTR) direction even when CO-NBIs were applied. One possible reason is that the ripple-induced fast ion loss may cause a negative $E_T$. Therefore, replacing perpendicular (PERP) NBs with tangential CO NBNBs leads to less CTR toroidal rotation. Figure 1.1.4-2 shows the response of the divertor $D_e$ signal during these rotation scans with fixed plasma shape. As can be seen, the ELM type was clearly changed from type I ELMs to grassy ELMs with higher frequency up to
1500 Hz. As the CTR rotation was increased, the ELM frequency gradually increased together with a reduction in the ELM amplitude.

![Graphs showing ELM behavior with CTR rotation](image)

Fig. I.1.4-2 Time evolution of $D_\alpha$ signal during plasma toroidal rotation scan at $\phi_5=4.9$ and $\delta=0.59$. Plasma rotation profiles were changed by using different combinations of NBIs: (a) 2CO+2PERP +2N-NB, (b) 2CO+3PERP+1N-NB, (c) 2CO+5PERP and (d) 1CO+1CTR+5PERP.

All the plasmas shown in Fig. I.1.4-2 satisfied the typical parameters (e.g. $q_95$, $\delta$ and $\beta_p$) for access to the grassy ELM regime, but the ELM characteristics were clearly different. Therefore, the toroidal rotation can be considered as an important parameter for access to the grassy ELM regime. One can question whether the most important parameter affecting the ELM type is the absolute value or the direction of the toroidal rotation. Since we cannot obtain a larger CO rotation as yet, further experiments are required to resolve this issue.

In the type I ELMMy H-mode plasmas, the change of plasma rotation (or direction of the momentum input) affects the ELM frequency and amplitude, but the plasma usually remains in the type I ELMMy phase. When the plasma position was carefully optimized, a steady ELM free phase (QH-mode) with stationary pedestal parameters was obtained as shown in Fig. I.1.4-3. After the LH transition at $t=3.45s$, an intermittent ELMing phase is observed. Then, the $D_\alpha$ signal remains at a high level from $t=4.56s$, which is concurrent with clear coherent temperature fluctuations ($T_e$ fluctuation) with frequencies of $-9$ kHz and $-18$ kHz. The mode was localized at the edge (R-R$_{sep}$=2cm), and edge density and ion-saturation current at divertor target were also modulated with the same frequency as the edge temperature fluctuation. This edge fluctuation may cause the reduction of pedestal pressure by $\sim 18\%$.

Basically, the QH-mode seems to be easily reproducible with CTR-NBIs. However, we have also observed partial QH phase during CO-NB injection phase with almost no net toroidal rotation at the plasma edge and during BAL-NB injection phase. The occurrence of $T_e$ fluctuations together with enhancement of the $D_\alpha$ signal was also observed in all QH phases.

![Graphs showing QH-mode plasma behavior](image)

Fig. I.1.4-3 Typical waveforms of QH-mode plasma in JT-60U. (a) plasma current and divertor $D_\alpha$ signal. (b) pedestal $T_e$ (black) and $T_i$ (gray). (c) line-averaged density and pedestal density. (d) Solid (dashed) line show total (CTR) NBI power together with radiation power shown by dotted line.

### 1.4.3 Reduced Heat Transport during the Inter-ELM Phase [I.4-3]

For the H-mode physics, the characteristics of the heat transport in a time scale longer than ELM events has been studied to predict the performance in future reactor. In addition, the energy pulse expelled by ELMs has also been intensively studied for the interest of predicting the peak heat load onto plasma facing components. However, little is known about heat transport that occurs during the phase between ELMs (or ‘inter-ELM phase’). In fact, the modelling of ELMMy H-mode involving MHD stabilities and anomalous transport process cannot be developed without the transport process responsible for the inter-ELM phase. In this study, the reduction of electron heat diffusivity during the inter-ELM phase to the level of ion...
neoclassical transport was found in the plasma edge region affected by an ELM burst.

Understanding the completed system of self-regulating dynamics of ELMing cycle is realized by separating the recovery phase between ELMs from the instantaneous ELM burst phase. In JT-60U, we have investigated the characteristics of the heat transport during the inter-ELM phase for the first time in the world. In a steady-state phase in a time scale much longer than an ELM event, the energy balance near the plasma boundary can be expressed as: \( P_{\text{heat}} = P_{\text{int}} + P_{\text{ELM}} \) where \( P_{\text{int}} \) and \( P_{\text{ELM}} \) are the loss powers of the inter-ELM transport and ELMs, respectively. The source heating power crossing the separatrix is assigned to these two loss channels. In this study, it has been found that, as \( v^* \) is increased, the inter-ELM transport is enhanced and the ELM heat loss is reduced. Since the reduced heat transport is always accompanied by a large ELM loss power, one can find that it is not simple to achieve the high energy confinement simultaneously with small ELM heat loss particularly at low \( v^* \).

In order to understand the process responsible for heat transport during the inter-ELM phase, we have examined the dependence of the inter-ELM energy confinement time \( \tau_{\text{E, int}} \) on \( v^* \) as shown in Fig. 1.1.4-4(a). It is obviously seen that \( \tau_{\text{E, int}} \) is significantly improved as \( v^* \) is reduced, satisfying the relation of \( \tau_{\text{E, int}} \propto v^{-0.6} \). Since the energy confinement in a global time scale shows weaker \( v^* \) dependence given as \( \tau_{\text{E}} \propto \langle v^* \rangle^{-0.35} \), this result is indicative of a more collision-based heat transport during the inter-ELM phase. We have also examined the \( \rho_{\text{pol}}^* \) dependence. In the peripheral plasma region, it is hard to conduct the pure \( \rho_{\text{pol}}^* \) scan because of the existing edge stability boundary. Thus, by adopting the knowledge of the \( v^* \) dependence obtained former, we have obtained the relation of \( \tau_{\text{E, int}} \propto \rho_{\text{pol}}^{-0.7} \). During the inter-ELM phase, this collapsed profile is replenished by the heat flux from the plasma core. Form this point of view, the electron heat diffusivity during the inter-ELM phase \( \chi_{\text{E, int}}^* \) (\( \propto \chi_{\text{E}} \)) has been calculated using the relative perturbations of \( T_e \) profiles due to an ELM. (see Fig. 1.1.4-4(b)). In the plasma edge region which is affected by ELM burst, the electron heat diffusivity is reduced significantly to the level of the ion neoclassical transport.

References

1.5 Current Drive Research
1.5.1 Real-Time Control of Safety Factor Profile [1.5-1]
Active control of safety factor profile is essential in sustaining a high performance plasma that is optimized in stability and confinement. A real-time control system of safety factor (q) profile has now been developed in
JT-60. This system, for the first time, enables 1) real-time evaluation of q profile using local magnetic pitch angle measurement by motional Stark effect (MSE) diagnostic and 2) control of current drive (CD) location (pcd) by adjusting the parallel refractive index n_{pcd} of lower-hybrid (LH) waves through the change of phase difference (Δφ) between multi-junction launcher modules.

A newly developed method for the q profile evaluation realized q profile calculation within every 0.01s, which is much faster than current relaxation time, typically order of 1s. Safety factor profile by the real-time calculation agreed well with that by equilibrium reconstruction with MSE. See section 1.2.6.2 for comparison of q(r) by the two methods. From temporal evolution of q (or current) profile, the system also evaluates p_{CD} in real-time. The location p_{CD} is where the rate of the current enclosed between two magnetic surfaces on which MSE channels are viewing are increasing. The control system changes p_{CD} through n_{pcd} (or directly Δφ) in such a way to minimize the largest residual between the real-time q profile and its reference profile. Since we had a dataset showing increase of n_{pcd} or Δφ shifts p_{CD} outward in minor radius, we employed an algorithm to control Δφ as follows: 
\[ d(Δφ)/dt = -α(p_{CD} - p_{CD_{ref}}) \] ; α is a positive constant. The system determines p_{CD_{ref}} comparing real-time q(r) and a give reference q(r); p_{CD_{ref}} is where the system wants to drive current. Thus, when CD location p_{CD} is smaller than the reference CD location p_{CD_{ref}} the system increase Δφ in order to shift CD location outward.

The real-time control system was applied to positive shear plasmas having q(0)=1 at I_p=0.6MA, B_t=2.3T, and n_e=0.5x10^{19}m^{-3}. The reference q profile was set to q(0)=1.3. In order to keep good coupling of LH waves to the plasma, gap between the launcher and the plasma surface was controlled to about 0.1m. Figure I.1.5.1 (a) shows waveforms of the discharge. When the LH power was stably injected (>10s), the loop voltage dropped down to 0V, q=1 surface vanished and q=1.25 surface shrank. The largest residual decreased close to its error level. The real-time q profile (Fig. I.1.5.1 (b)) approached to its reference profile after application of the real-time control; the controlled q profile was sustained for 3s (t=13-16s), which was limited by injected LH power.

1.5.2 Validation of Beam Particle Self Interaction
Circulating fast ions generated by NBI are predicted to affect the beam stopping cross-section of the neutral beam itself through the interaction between the neutrals and the fast ions. This "beam-particle self-interaction (BPSI)" can be notable especially for a high-energy beam injected into a low density plasma [1.5-2,3].

One of the beam lines of N-NB, N-NB (U), was used for this experiment. The beam power was 1.5 MW with an energy of 350 keV. The major plasma parameters have been maintained nearly in constant; I_p=1MA, B_t=2.5T, <n_e>=0.92x10^{19} m^{-3}, <T_e> = 1.2 keV. The beam pulse duration was 1.5 sec, which is much longer than the beam slowing down time. The shine-through power is evaluated using temperature increment of N-NB facing tiles measured by an infra-
red (IR) camera. Heat transport analysis of the facing tile has shown that the time evolution of the tile temperature is not reproduced by assuming the shine-through power in proportion to beam power, but it has been well reproduced assuming that the shine-through fraction decreases exponentially about by 35% within several hundred msec.

Fig.1.1.5-2 (a) An estimated waveform for shine-through and (b) temperatures of the facing tiles. The measured temperature is denoted by a dotted curve while the numerical calculation is denoted by a solid curve.

This time scale is close to the build-up time of fast ion component (~200 msec). The reduction of shine-through is also consistent to the estimated range by the BPSI theory[1.5-4].

1.5.3 Current Clamp in the Current Hole
A stable tokamak plasma with nearly zero toroidal current in the central region (a “current hole”) is sustained for several seconds in the JT-60U tokamak [1.5-5]. However, it has not been clear whether the current drive source such as inductive toroidal electric field $E_\phi$ and non-inductively driven current $j_{ni}$ remains at zero level during the sustainment of current hole or some mechanism works to clamp the current density at zero level against the current drive source.

Two kinds of experiments were performed to investigate responses to $E_\phi$ and $j_{ni}$ separately [1.5-6, 1.5-7]. In the first experiment, $E_\phi$ was changed transiently by variation of $j_{ni}$ outside the current hole, keeping $j_{ni}$ inside the current hole as small as possible. The electron cyclotron wave (ECW) power of 2.6 MW at the frequency of 110 GHz was injected outside the current hole at $t = 5.2$ s during the current flat top in a plasma of $I_p = 1$ MA and $B_t = 3.6$ T, to increase $j_{ni}$ (the EC-driven current and the bootstrap current) in the direction of the main plasma current. In the same discharge, at $t = 6.0$ s or 0.8 s after the start of ECW injection, ECW power and almost all NB power were turned off, to decrease $j_{ni}$ rapidly. Radial profiles of loop voltage $V_{\text{loop}}(\rho)$ are shown in Fig. 1.1.5-3(a), where $\rho$ denotes the normalized minor radius. The $V_{\text{loop}}(\rho)$ is obtained from the time-derivative of the poloidal magnetic flux $\Psi(\rho)$ which is determined by the equilibrium reconstruction using the MSE data. The $V_{\text{loop}}(0)$ was negative ($\sim -0.3$ V) during EC injection ($t = 5.4-5.6$ s) while it was positive ($\sim +0.4$ V) after the stop of EC injection ($t = 5.98-6.4$s). Note that uncertainties in $V_{\text{loop}}(\rho)$ are relatively small near the axis though they are large around $\rho = 0.5$ where $\Psi(\rho)$ has a large gradient. In Fig. 1.1.5-3, the sum of calculated $j_{OH}$, $j_{EC}$, $j_{NB}$ and measured $j_{ni}$ are

Fig. 1.1.5-3. Radial profiles of loop voltage and current density in a discharge with $E_\phi$ changed transiently. (a) Loop voltage $V_{\text{loop}}$ during $t = 5.4-5.6$ s and $t = 5.98-6.4$ s. (b), (c) Current densities at $t = 5.5$ s and $t = 6.2$ s. In (b) and (c), $j_{ni}$ (solid line with a shaded belt) denotes the measured current density, while the summation of $j_{OH} + j_{NB} + j_{EC}$ is shown by the dotted line with the error bars.
compared for (b) negative $E_d(0)$ and (c) positive $E_d(0)$ cases. Here $j_{\text{inj}}$, $j_{\text{EC}}$, $j_{\text{abs}}$, and $j_{\text{BS}}$ denote the calculated inductive, EC-driven, beam-driven and bootstrap current densities, respectively. In both cases, the calculated current density is dominated by the inductive current, and is largely negative in (b) and is largely positive in (c). The measured current density, however, remained nearly zero. In the second experiment, EC current drive inside the current hole was attempted in the same and opposite directions to the plasma current in a plasma of $I_p = 1$ MA and $B_t = 3.7$ T during the quasi-stationary period with small $E_d(0)$. In neither direction did the EC current drive change the current inside the current hole, and the current hole was maintained.

From these results, it has been shown experimentally for the first time that although the current drive source exists in the current hole, some mechanism works to clamp the current density at zero level once it becomes at zero level in the central region. Simulation results show that resistive MHD instabilities take place in the current hole, leading to the current clamp. In our experiments, however, no MHD instabilities with a high frequency (1-100 kHz range) were observed. Though small collapses with longer intervals (~0.1s) were observed in some discharges, no clear change in the current density inside the current hole was observed between or at these collapses, indicating that these collapses are not the cause of the current clamp in the current hole.

1.6 Divertor/SOL Plasmas and Plasma-Wall Interaction

1.6.1 SOL Transport of ELM Plasma and Fluctuations in L- and H-Modes

Transient heat and particle loading caused by ELM is crucial for determining the lifetime of ITER divertor materials. At the same time, study of the ELM radial propagation was recently focused to evaluate the heat and particle loadings to the first wall. Determination of the perpendicular propagation during ELM deposition (such as a few 100µs) was improved with increasing sampling rate from 200kHz to 500kHz for the Mach probes (at outer midplane and X-point) and magnetic pick-up coils [1.6-1]. Fluctuation level was compared in L-mode and ELM H-mode plasmas, and analysis of the fluctuation characteristics using Probability Distribution Function (PDF) was initiated [1.6-2].

(1) Characteristics of ELM propagation

During ELM deposition, large multi-peaks appeared in ion saturation current at midplane Mach probe, $j_{\text{mid}}$, during the base level of $j_{\text{mid}}$ increasing, just after large magnetic turbulences due to MHD activities (Fig.1.1.6-1). Time lags of the first $j_{\text{mid}}$ peak and the maximum base-level were defined as $\tau_{\text{perp mid}}(\text{peak})$ and $\tau_{\text{perp mid}}(\text{base})$.

References
1.5-2 Okano, K., Nuclear Fusion 31, 1349 (1991).
1.5-4 Okano, K. et al., Submitted to Journal of Plasma and Fusion Research
$\tau_{\text{peak}}^{\text{mid}}(\text{base})$, respectively. Here, $\tau_{\text{peak}}^{\text{mid}}(\text{peak})$ was a delay due to the SOL plasma transport across the magnetic field, and duration of large $j_s^{\text{mid}}$ peaks, $\delta t^{\text{peak}}$, was short: 8-24$\mu$s. Foreexample, $\tau_{\text{peak}}^{\text{mid}}(\text{peak})$-40$\mu$s, and $\tau_{\text{peak}}^{\text{mid}}(\text{base})$-160$\mu$s at the midplane distance from separatrix ($\Delta x^{\text{mid}}$) of 4.8 cm. On the other hand, the delay of the increase in divertor $j_s^{\text{div}}$, $\tau_{\text{on}}^{\text{div}}$, was 70-130$\mu$s, and that of the maximum $j_s^{\text{div}}$ base level, $\tau_{\text{off}}^{\text{div}}$, was 130-200$\mu$s. These time lags were consistent with parallel convective transport time along the magnetic field, $\tau_{\text{para}}^{\text{SOL-DIV}}=140$ $\mu$s. As a result, the radial transport of the peak $j_s^{\text{mid}}$ was faster than $\tau_{\text{off}}^{\text{div}}$, whereas parallel and radial propagation of the base-level were comparable.

The large multi-peaks in $j_s^{\text{mid}}$ were observed over all radii ($\Delta x^{\text{mid}}<15$ cm) with large radial delay length. Since $\tau_{\text{peak}}^{\text{mid}}(\text{peak})$ increased with $\Delta x^{\text{mid}}$, the radial velocity, $V_{\text{peak}}^{\text{mid}}(\text{peak})$, ranged between 1.3 and 2.5 km/s. From $V_{\text{peak}}^{\text{mid}}$-2 km/s, characteristic radial scale of the $j_s^{\text{mid}}$ peak was estimated to $\delta t^{\text{peak}} V_{\text{peak}}^{\text{mid}}=1.5$ - 4 cm, which may locally deposit the heat and particle loading to the first wall. Enhancement of $j_s^{\text{mid}}$ base-level occurred globally and simultaneously in far SOL and divertor. Extension of the $j_s^{\text{mid}}$ base-level was within smaller SOL radii ($\Delta x^{\text{mid}}<10$ cm), which may only influence particle and heat load to outer baffle.

(2) Characteristics of SOL Plasma Fluctuations

For ELMy H- and L-modes, fluctuation levels of the midplane SOL plasma, $\delta j_s^{\text{mid}}/j_s^{\text{mid}}$, were generally 4-5 times larger than those near X-point. At the same time, far SOL $\delta j_s^{\text{mid}}/j_s^{\text{mid}}$ in H-mode was increased with $\Delta x^{\text{mid}}$, which became 5-10 times larger than L-mode. Here, e-folding lengths of $j_s^{\text{mid}}$ profiles were comparable for H- and L-modes, thus the decay length was not influenced by the fluctuation level.

Statistical analysis of the $j_s^{\text{mid}}$ signals using PDF showed that density bursts became remarkable around $\Delta x^{\text{mid}}=6$-7 cm. Since the direction of the parallel SOL flow changed from the inner divertor to the outer divertor near $\Delta x^{\text{mid}}=7$ cm, enhancement of the density bursts may be related to the stagnation of the SOL flow. The fast fluctuations and ELM bursts are under investigation using Wavelet analysis, which shows transient characteristics of turbulences.

1.6.2 Spectroscopic Study of H$_2$ Molecules in Divertor

Understanding behavior of H$_2$ molecules in divertor plasmas is important for control of divertor plasmas and diagnostics of neutral particles [1.6-3]. In JT-60U, H$_2$ molecule behavior in attached and detached divertor plasmas has been studied by observation of H$_2$ Fulcher line emission [1.6-4].

![Fig. 1.1.6-2. Spatial profiles in the detached divertor plasma as functions of the distance from the outer divertor plates. (a) Calculated electron temperature and density along the separatrix, (b) observed (points) and calculated (line) He line intensity, (c) observed (points) and calculated (continuous line: with dissociative attachment from the n=3 state, broken line: without dissociative attachment) Fulcher v=1-1 Q3 line intensity, (d) calculated MAR (continuous line) and H$^-$e recombination (broken line) rate. In both (c) - (d), the thick and thin lines indicate the results obtained by assuming that the vibrational temperatures of the ground state were 0.5 eV and 1 eV, respectively.](image-url)
radiative model code. Spatial profiles of various measured and modeled plasma parameters in the detached divertor plasma are shown as functions of the distance from the outer divertor plates in Fig. 1.1.6-2. A detached divertor plasma solution was obtained from calculation using a two-dimensional fluid code. The H₂ Fulcher line intensity profiles calculated with and without considering the dissociative attachment from the n=3 state are compared with the observed profile. With the ground-state vibrational temperature of 0.5 eV, the calculation reproduced the logarithmic slopes in the observed intensity profile. Molecular assisted recombination (MAR) was estimated to be as important as H⁻⁻ recombination in the detached divertor plasma.

1.6.3 Modelling of Impurity Transport [1.6-5]
Impurity transport has been modeled using the 2-D fluid code UEDGE [1.6-6] in the divertor plasma for the high βₚ H-mode plasma with highly enhanced radiation by injecting seed impurity Ar. The impurity diffusivity was set to be 1.0 m²/s without the convection velocity. The carbon yield rate was set to be a Haasz yield [1.6-7] for both physical and chemical sputtering. The ratio of total Ar density to the electron density at the core_edge boundary (96% flux surface) (n_Å/nₐ)core_edge was scanned in the range of 0.14-1%.

![Comparison of calculated and measured radiation profiles](image)

Fig. 1.1.6-3. Calculated (lines) and measured (circles) radiation profiles. Dashed, solid and dotted lines show the radiation profile calculated using UEDGE results with (n_Å/nₐ)core_edge=1%, 0.75% and 0.14%, respectively.

Figure 1.1.6-3 shows comparison of radiation profile in the divertor plasma between measurement and calculation. The calculated divertor radiation had peaks at both strike points (ch 11 for inner and ch 19 for outer divertor) as well as the measurement. The peak at the inner strike point was almost constant even when (n_Å/nₐ)core_edge was increased. The peak at the outer strike point was largely enhanced with (n_Å/nₐ)core_edge=1% and was larger than the peak at the inner strike point. The calculated radiation with (n_Å/nₐ)core_edge=0.75%, which was consistent with edge Ar density estimated from the radiation in the main plasma, was consistent with the measurement at the inner strike point within the ambiguity of the measurement. At the outer strike point, the effect of misalignment of the sight lines was large due to strong localization of the radiation in front of the outer divertor plate. Therefore, when possible misalignment was considered, the radiation loss at the outer strike point increased to the same value as the measurement. On the other hand, the radiation in the divertor region (ch 9&10 for inner and ch 17&18 for outer divertor) was smaller than the measurement, indicating that the calculated radiation was localized around the strike points compared with the measurement.

1.6.4 Retention Characteristics of Hydrogen Particles
In order to study the tritium retention in different conditions (divertor geometry and operation temperature) compared with other tokamaks, we have investigated the erosion/deposition distribution and the hydrogen isotopes (H,D,T) behavior in the JT-60U plasma-facing wall (carbon-based). Fig. 1.1.6-4 shows schematic views of the JT-60U W-shaped divertor with an inner pumping slot and sample locations for the analyses. The operation temperature of the JT-60U
vacuum vessel was ~570 K. The base temperature of the divertor tiles were also ~570 K, since the divertor tiles were inertially cooled. The sample tiles were exposed to plasma from June 1997 to October 1998. The total number of deuterium discharges during this periods was ~3600 shots. Following the deuterium discharges, ~700 hydrogen shots were performed in a clean-up operation.

Deposition was found to be dominant on the inner divertor target, whereas erosion was dominant on the outer divertor target. No continuous deposition layer was obviously observed in the dome top tile [1.6-8]. Such in/out asymmetry of the erosion/deposition has been observed also in many tokamaks. In JT-60U, however, distributions of the hydrogen isotopes were not obviously correlated with the deposition distribution as described later.

to be (H+D)/C ~0.07 at the outer dome wing, which is much less than that observed in other tokamaks. Such low (H+D)/C must be attributed to high surface temperature of the dome tiles (~800 K). For the deuterium retention, at least two retention processes (ion-implantation and co-deposition) were distinguished on the dome region. [1.6-10, 11].

Form these hydrogen isotope analyses, it was found that the behavior of the hydrogen isotopes in the plasma-facing wall should be also considered with the ion implantation as well as the co-deposition for the detailed estimation of the tritium retention in ITER.

References
1.6-5 Takenaga, H., et al., to be published in Journal of Plasma and Fusion Research SERIES 7.
1.6-11 Hayashi, T et al., J. Nucl. Mater., in press.

Fig. I.1.6-5 Tritium and deuterium distributions
Tritium intensity and D/C ratio were obtained by imaging plate technique and nuclear reaction analysis, respectively

Distribution of the tritium, which was produced by D-D nuclear reaction, in the plasma-facing wall reflected the distribution of high-energy tritium ion implantation due to ripple loss and a slight modification owing to high surface temperature of the divertor target tiles. According to OFMC simulation, ~50% of the produced tritium were lost and implanted into the wall with high energy of up to ~1 MeV[1.6-9].

Deuterium distribution in the JT-60U divertor region was slightly different from the tritium distribution as shown in Fig. I.1.6-5. The highest concentration of the hydrogen isotopes was estimated
2. Operation and Machine Improvements

Two cycles of the JT-60 operation were implemented in FY 2004, which includes 902 shots of plasma pulse discharge, 43 shots of commissioning pulse sequence, 15 hours of Taylor-type discharge cleaning and 239 hours of glow discharge cleaning.

In the operation of the JT-60 facilities, the motor generator for toroidal magnetic field coil (T-MG) was used tentatively in place of the motor generator for plasma heating systems which had been out of order in FY 2003. Therefore the electric power feeder lines and control system of the T-MG were modified. On the other hand, the toroidal magnetic field was activated only by the grid power line. With careful operation of these electric power systems, JT-60 experiments were successfully implemented as plane.

2.1 Tokamak Machine

2.1.1 Operation without a Center Solenoid Coil

Innovative plasma build-up operations without a center solenoid, which would simplify the structure of future tokamak fusion reactors, were planned in this experimental campaign. Since poloidal magnetic field coils would receive different forces from usual operations in these operations, electromagnetic force analysis was done to confirm the safety of these coils. The analysis result showed that the electromagnetic forces on the coils were smaller than 20% of the allowable limits respectively, and these coils would be mechanically safe in this experiment.

2.1.2 Transportation of Solid Radioactive Wastes

Solid radioactive wastes are taken out from JT-60 facilities and stored in metal drums in the radioactive waste storage building in every maintenance period. To reduce the combustible wastes by incineration, forty metal drums were transported from Naka site to Tokai Site, in which the facility of Department of Decommissioning and Waste Management treated with them in Tokai Site. The transportation was initiated in 1999 and a total of 240 drums were removed.

2.1.3 Fabrication of Ferritic Steel Plate and Installation into the First Wall

Installation of ferritic steel tiles was proposed in JT-60U to reduce the toroidal magnetic field ripple and to improve the fast ion loss, which decreases plasma heating efficiency and increases heat load on plasma facing components in the operations with plasmas with a large volume. Candidate materials with a high saturated magnetization, SUS430 (18Cr), STBA26 (9Cr-1Mo) and F82H (8Cr-2W-0.2V-0.04Ta), were compared. Taking into account the cost-effectiveness to obtain a saturated magnetization required for experiments and the moderate neutron generation level in JT-60U, 8Cr-2W-0.2V ferritic steel whose activation element concentrations are a little higher than those of F82H was selected. The expected saturated magnetization is 1.8 Tesla at operational temperature of around 570 K.

Fabrication procedure of the ferritic steel mostly followed that of F82H. The steel more than 20 tons was melted in a vacuum induction furnace and cast into eight ingots. Seven ingots were used to obtain steel plates through the processes of forging and hot-rolling. These plates underwent normalizing at 1273 K for 30 min, air-cooling and then tempering at 1023 K for 90 min followed by air-cooling. The fabricated ferritic steel has clear tempered martensitic microstructures, and sufficient magnetic and mechanical properties. The saturated magnetization measured was over 1.7 Tesla at 573 K. Although it was lower than the expected value, it was confirmed by a numerical calculation that the saturated magnetization of 1.7 Tesla was sufficient for the JT-60 experiment.

2.1.4 Study of the Plasma-Surface Interaction

The cooperative research program between JAERI and universities using the JT-60 first wall tile was initiated in 2001. Under the program, various studies on the plasma facing materials have progressed [2.1-1], [2.1-2], [2.1-3]. Major research activities conducted in FY 2004 are as follows:


Thermal properties of the redeposition layer on the inner plate of the W-shaped divertor of JT-60U were measured with laser flash method in order to estimate transient heat load such as ELMs onto the divertor. Morphology analysis of the redeposition layer was conducted with a scanning electron microscope.

Redeposition layers of more than 200 μm thick were observed near the most frequent striking point.
Hydrogen isotopes were released from the graphite tiles used in JT-60U by the thermal desorption method. When the first wall tile was left under helium atmosphere at 600°C for 8 hours, about 40% of total amount of hydrogen and deuterium were released, while the amount of released tritium was only about 20%. At high temperature of 1000°C, the release rate of deuterium and tritium was enhanced. It was found that the amount of hydrogen retained in the graphite tile was much larger than that of deuterium. This indicates that a large amount of deuterium trapped in the tiles during deuterium discharge experiments was replaced with hydrogen during hydrogen discharge experiments.

(4) Tritium Release Behavior During Air Exposure and Gas Purge Conditions [2.1-8]

Exhaust gas from the JT-60U tokamak was analyzed to understand the behavior of fuel and impurity elements in the vacuum vessel. The behavior of tritium release by an isotope exchange reaction during air exposure and gas purging phases has been investigated. For the air exposure with water vapor concentrations of 40ppm, 300ppm, 680ppm and 3400ppm, tritium concentration in the air was measured. It was confirmed that water vapor enhanced release of tritium from the vessel. Tritium concentration initially increased with time and then became constant finally at each concentration level. The total amount of tritium released from the vacuum vessel was 13MBq for 3400ppm, which is almost the same as that removed by 5 hours' H₂-GDC that has been most effective detritiation method in JT-60U [2.1-9]. This suggests that tritium can be easily removed by water vapor.

Tritium release during the gas purging was measured. The various gases (H₂, He and Ar) were introduced into the vacuum vessel at constant pressure by controlling the gas flow rate. Tritium concentration was about 0.1Bq/cm³ at room temperature and was independent of gas species within pressure from 0.05 to 0.3 Pa., indicating that isotope exchange of tritium with hydrogen molecules was not so active under these purge conditions.

A trace of Oxygen purge was also examined. In the tokamak discharge, various hydrocarbons such as CD₄, C₂D₄, C₂D₆ and C₆D₂ were detected and possible relation with formation of codeposit is suggested because hydrocarbon could be produced by a shift.
reaction related with oxidized elements, such as water, carbon oxide and carbon dioxide. A gas purging and a glow discharge cleaning with 0.1% oxygen contained He showed a possibility of enhance the carbon removal. No effect of oxygen onto tokamak discharge was observed.

References
2.1-2 Tanabe, T., et al., ibid.

2.2 Control System

2.2.1 Development of an Innovative Integrator Resistant to Plasma Instabilities

A new integrator for magnetic measurements aiming at long pulse operation have been developed and tested in JT-60U [2.2-1]. Although most of the technical issues have been resolved, an amplifier saturation caused by exposure of excessive voltage input from the sensor is still remaining as the major issue [2.2-2]. Therefore we had built an advanced integrator system, which is composed of the three sets of the VFC-UDC unit with different amplitude gains and a digital signal processor (DSP) to prevent at least one of the operational amplifiers from saturating due to excessive voltage input. In addition, necessary numbers of FET-Zener diodes were added to the signal input line for protection of each operational amplifier.

The total performance test was conducted using one of the magnetic probes in JT-60. Figure I.2.2-1 (a) shows a good, accurate integration result even with a disruptive instability in tokamak discharge. In this case, no baseline change was observed before and after plasma discharge. Unexpectedly, soon after a few disruption plasma shots, clear baseline gap was again observed as shown in Fig. I.2.2-1 (b). The cause of this phenomenon has been identified to be semiconductor characteristics change of the FET-Zener diode elements equipped in the signal input circuit.

We have prepared three cases of modifications in the signal input circuit.

(1) Diode Withstanding ±1 kV (Case I)
A gap ("stepped change") of the integrated signal came to be observed after several plasma disruptions. We considered that the exposure of continual extreme high-voltage inputs could make the FET-Zener diode characteristics degrade drastically. To prevent this, a new diode withstands high voltage (±1 kV) has been superseded the FET-Zener diode.

The linearity errors for three ranges (range: 10 V, 100 V, and 1000 V) exceed the specification of the employed operational amplifier (±0.001%). The cause of this linearity error is presumed the large leakage current of the diode with a 250 V pull-up power supply.

(2) Attenuator Insertion (Case II)
Since we are concerned about a large amount of diode leakage current at the signal front-end in the above case, the diode elements were removed from the input circuit. This improvement makes the circuit withstand ±1.0 kV without a diode.

The linearity errors for this case are permissible (less than 0.001%) except for the 1000-V range.

(3) Power Mos FET-Zener (Case III)
Since the Case I trial board does not satisfy the accuracy requirements due to the large amount of leakage current, the Power Mos FET-FET, that does not need any high voltage power supply, has been chosen to make a short circuit in case of over-voltage input. The test for this case is under preparation.

Fig. I.2.2-1 A gap of integral results occurred after several exposures to high voltage at a disruption.
We have built three trial boards as a measure to avoid the semiconductor characteristics change of the FET-Zener diode caused by the continual extreme high-voltage inputs in JT-60, and tested two of them. We will conclude our development result soon after the tests on the Case III including an impulse surge test.

References

2.3. Power Supply System
2.3.1 Tentative Power Transmission from the T-MG to the NBI and RF Heating Systems
Since a serious trouble on the H-MG happened in February 2004, a large-scale repair came to be required for the complete recovery. In order to restart the plasma heating experiments as early as possible, we decided to disconnect the T-MG from the toroidal field coil power supply and to reconnect it to the heating devices in place of H-MG as shown in Fig. I.2.3-1. Followings are its details.

![Diagram of power supply system](image)

**Fig. I.2.3-1 Main circuit reconfiguration.**

(1) Circuit Configuration
From the viewpoint of energy available directly from the power grid, the maximum toroidal magnetic field coil current was considered about 70% of the rated current. This implies that large toroidal field could be produced sufficiently for the long pulse operation to study the high-beta plasmas. The actual difficulty was arisen in the coil current control capability in contrast with the output voltage control of the T-MG. Then, we modified the On/Off control scheme of the diode rectifier banks on the technical basis of discharge pulse prolongation (from 15 s to 65 s), conducted in the previous year.

The power line of H-MG was disconnected completely for safety at the point near the MG-pit by removing the bus-bars in the metal covered duct. The T-MG and the heating systems are connected by two lines of CV cables which has 22 kV rating and its cross section of 800 mm² by utilizing the unused switch gear boxes. According to the change of main circuit, the hard wired protection system was also modified to adjust the difference of ratings and the protection.

(2) T-MG Output Voltage to the Heating Devices
The major ratings of T-MG and H-MG are summarized in Table I.2.3. The voltage disturbance caused by the power fluctuation is basically anti-proportional to the capacity of generator. In this case, since the synchronous impedances for both MGs are similar in quantity, the voltage disturbance of T-MG is expected about two times of that of H-MG. To prevent the excessive over-voltage generation of T-MG that might be happened at the moment to separate the large electrical loads we reduced the T-MG output voltage to 16 kV from the rated value of 18 kV.

| Table I.2.3 Specifications of T-MG and H-MG |
|---|---|---|
| **T-MG** | **H-MG** |
| Capacity | 215 MVA | 400 MVA |
| Voltage | 18 kV | 18 kV |
| Current | 6,896 A | 12,830 A |
| Frequency | 80-56Hz | 77.6-54.2Hz |
| Drive Type | Thyristor | Scherbias |

(3) Re-Acceleration of T-MG
While the H-MG has an exclusive induction motor (IM) on the top of the generator for its acceleration, the T-MG is designed to be driven by thyristor drive device (AC-DC-AC drive) without IM. Therefore, the voltage distortion due to the commutation of thyristor drive was expected, because there is no inverter transformer as shown in Fig.1.2.3-2. To avoid the commutation failure in the thyristor drive device and the NBI system, we
stopped the re-acceleration of T-MG during the period of plasma discharge and the NBI conditioning.

2.3.2 A Temporary Method for TF Coil Current Control

(1) Toroidal Field Coil Power Supply Configuration

The toroidal field coil power supply (TFPS) consists of the four diode rectifier banks powered directly from the commercial line of 275 kV and the two diode rectifier banks powered by the T-MG as shown in Fig.1.2.3-2. Since the T-MG was exclusively connected to the heating systems due to the H-MG trouble, the toroidal field coil current must be controlled by the four diode rectifier banks solely.

(2) On/Off Control of the Diode Rectifier Banks

Since the four diode rectifier banks are completely identical, the DC output voltages of the banks are also identical. It means that only four levels of the TF coil voltage could be selected. Then, we utilized the transformer tap changer in order to improve the flexibility in determining TF coil voltage. Here, the transformer tap can control the secondary voltage from 0.95 p.u. to 1.17 p.u. compared to the present tap position. In the actual experiments, the tap position was chosen to different position from each other to maximize the flexibility of TF strength.

Fig.1.2.3-3 is an example of the On/Off control in the diode rectifier banks. The current waveform is usually provided by the physics operator as a pre-programmed form. The turn-on/off timings, which are the close timings of C1-C4 and the open timings of O1-O4, of the diode rectifier banks that could realize the expected toroidal field are automatically set up by the control system prior to the start of discharge sequence.

2.4. Neutral Beam Injection System

The pulse duration of the NBI system was extended from 10 s to 30 s to study quasi-steady state plasmas on JT-60U. As for four positive-ion based (P-NBI) units with tangential beams, the electric power supplies and the beam limiters were mainly modified and the pulse duration was successfully extended up to 30 s with 2 MW at 80 keV. Other seven P-NBI units with perpendicular beams, whose pulse durations were 10 s, were operated in series for 30 s in total instead of extending the pulse duration of each unit. The ion source of the negative-ion based (N-NBI) unit, whose
target beam energy is 500 keV for 10 s, was also modified to reduce the heat load of the acceleration grids for 30 s operation at ~350 kV. The pulse duration was extended up to 25 s at ~1MW and 20 sec at 1.6MW. The total injected energy reached up to 340 MJ with 330MJ for P-NBI and 10MJ for N-NBI.

2.4.1 Modification of Control System for NBI
The NBI control system gives the commands of the outputs and timing for each power supply and the gas introduction systems, where the operations of all components are optimized to generate a high power neutral beam. For the long pulse operation, two timing sequences have been developed to achieve a high power injection for 30 s. One is to extend the acceptable sequence period of the tangential P-NBI and N-NBI units from 10 sec to 30 sec, where the NBI control system accept the commands from the JT-60 control system (ZENKE). The other is to make it possible to start the sequence of perpendicular P-NBI units (10 sec) at any time during the 30 sec period. Thus, a long pulse injection of 30 sec can be obtained by adjusting the timing of each perpendicular unit in series.

2.4.2 Modification of Beam Limiter
The beam limiters are made of molybdenum and protect the drift duct from the divergent beam. The reduction in heat load onto the beam limiters is the critical issue to extend the pulse duration, because the beam limiter is not actively cooled. The temperature measurement of the original beam limiter indicated that the maximum temperature would rise up to above 750 °C for 2MW injection for 30 s and molybdenum would be transformed only after several long pulse operations. Therefore, the shape of the beam limiter was modified to reduce the heat load density, and the volume was enlarged to decrease the temperature rise. Figure 1.2.4-1 shows the time evolution of the maximum temperature of both the original beam limiter and improved one. The solid and dotted lines are measured and calculation results, respectively. It is found that the maximum temperature remains below 520 °C for 2MW injection for 30 s [2.4-1].

2.4.3 Reduction of Grid Heat Load of Negative Ion Source
One of main issues for extending the pulse duration of the negative NBI was to reduce the heat load to the ion source grids [2.4-2]. The large heat load to the grounded grid (GRG), in fact, limited the extension of the pulse duration. There are two causes; bombardments of accelerated negative ions and accelerated electrons stripped from negative ions. To reduce the electron stripping, it is necessary to decrease the neutral pressure in the accelerator. While the pressure in the ion source chamber is required to be more than ~0.3 Pa for negative ion production. The gas flows from the ion source chamber to the cryogenic pump through the grids.

![Fig. 1.2.4-1 Time evolution of temperatures for original and modified beam limiters at 2MW injection power. The solid and dotted lines are measured and simulation results, respectively.](image1)

![Fig. 1.2.4-2 Picture of second acceleration grid of the negative ion source. Both edge segments are large vent grids and only inside three segment s accelerate negative ions.](image2)
Fig. I.2.4-3 Simulated stripping loss in the ion source

The negative ion source is composed of five grids; plasma grid (PLG), extraction grid (EXG), first acceleration grid (A1G), second acceleration grids (A2G) and GRG. Each grid is divided to five segments. Both sides of the acceleration grids were removed to increase the gas flow conductance in the accelerator column as shown in fig.1.2.4-2. Correspondingly, both sides of the PLG were masked entirely so as not to extract negative ion beams, which decreased the extraction surface to 73% of the original area.

Stripping losses in the ion source were estimated from the calculated pressure profiles in the ion source, where the pressure in the extractor and accelerator were evaluated from the grids conductance and gas flow rate. Figure I.2.4-3 shows the integrated stripping loss along beam axis calculated from the cross section of neutralization on this pressure profile. The stripping loss is mainly generated in the region between the PLG and A1G where the pressure is high and the cross section of neutralization is high due to low beam energy. In the figure, the stripping loss in the ion source is reduced from 0.22 to 0.16 (a 27% improvement) by increasing the vacuum conductance. Figure I.2.4-4 shows the measured GRG heat load normalized by beam power as a function of the source pressure ($P_{in}$). The heat load ratio is decreased from 9.2% to 7.0% at 0.3 Pa after the modification. The heat load increases linearly with $P_{in}$ but the slope of the pressure dependence of the heat load became weaker after the modification. The pressure dependence of the heat load is presumed to arise from the interception of accelerated electrons which are created by stripping being intercepted on the GRG. So the improvement of vacuum conductance was effective in reducing the stripping loss. The offset of the heat load at zero pressure is assumed to be due to direct interception of divergent negative ions, which has little pressure dependence. Therefore the direct interception of the negative ion beam on the GRG is about 4.5% and the stripping loss is reduced from 4.6% to 2.6% (a 43% improvement) at 0.3 Pa after the modification. By this modification, the pulse length of negative NBI could be expanded up to ~20 s at 345 keV, 1.6 MW power.

Reference

2.5. Radio-Frequency Heating System
Performance of the JT-60U radio-frequency (RF) heating system has been constantly improved to extend the parameter region of experiments such as sustainment of high performance plasmas for a few tens seconds. In FY 2004, major improvements of the JT-60U RF heating system were extending the pulse duration of the electron cyclotron heating (ECH) system
and raising the performance of the lower hybrid (LH) system.

2.5.1 Long-Pulse Operation of the ECH System

Extension of the pulse duration of the ECH system was required since the maximum duration of the JT-60U discharge was extended from 15 s to 65 s in 2003 to investigate long sustained high performance plasmas. A tentative objective was set at the pulse duration of 30 s with an injected power of 0.6 MW. However, the ECH system was confronted with a difficulty in extending the pulse duration. It was a slight decay of the electron beam current of an oscillator gyrotron because of cathode cooling by electron emission. This decay caused abrupt termination of the gyrotron oscillation. Against the problem, some countermeasures were examined to compensate the beam current decay so as to keep the oscillation condition. Those were to control actively the magnetic field at the cavity, cathode temperature and electron pitch angle at the electron gun. Among those, in particular, we found that changing the electron pitch angle through controlling anode voltage of the gyrotron was most effective to keep the gyrotron oscillation, of which time response is quite faster than the others [2.5.1]. The electron pitch angle \( \alpha \) is defined by \( \alpha = v_\perp / v_\parallel \), where \( v_\perp \) and \( v_\parallel \) are the velocity of the electron perpendicular and parallel to the magnetic field line, respectively. In the operation without anode voltage control, as shown in Fig. I.2.5-1, the oscillation terminated at around 10.5 s, though the immediate changes of the beam currents, pressure or temperature in the gyrotron was not observed. It seems to be out of the oscillation condition through changing the electron density or spatial distribution at the cavity due to decaying the electron beam current. In the operation with anode voltage control so as to keep the oscillation condition, the gyrotron oscillation was sustained for 16 s, which was the setting time of the pulse duration, with increasing anode voltage by 400 V at 6 s after the operation start, as shown in Fig. I.2.5-1. A pulse duration of 16 s, in this moment, is the maximum operation time due to the limitation of the temperature rise of the DC break in the gyrotron. As a result of the operation way, the injected power of 1 MW for 15 s was obtained in combined operation of three gyrotrons, and a pulse duration of 45 s with 0.35 MW was achieved in series operation of four gyrotrons, as shown in Fig. I.2.5-2.

The temperature rise of the DC break is another issue for further extension of the pulse duration, as mentioned above. The DC break, originally made of \( \text{Al}_2\text{O}_3 \), between the body and the collector of the gyrotron is heated by scattered RF waves through diffraction loss at the outlet of the mode converter. To improve the excessive temperature rise of the DC break, \( \text{Si}_3\text{N}_4 \) which was developed for the ITER ECH (170 GHz) system in place of \( \text{Al}_2\text{O}_3 \) was tried to one of four gyrotrons. The new DC break is being tested.

![Fig. I.2.5-1 Oscillation duration was extended up to 16 s by anode voltage control indicated as "Controlled" while it was terminated at 10.5 s in normal operation.](image1)

![Fig. I.2.5-2 Progress in RF injection of the ECH system.](image2)
2.5.2 Performance of the LH System Having the Modified Launcher with Developed Carbon Grills

The LH system has contributed to studies of high performance plasmas such as reversed shear plasmas, adopting a multijunction-type launcher. However, the launcher [2.5-2] was damaged due to excessive heat loads around its mouth during 10-year operation. The injected power gradually decreased year by year. As a result, thin carbon grills were developed to recover the power injection capability [2.5-3], because carbon materials have high resisting capability against heat load, and further less harmful influence to plasma performances due to low ionic charge even if the sublimation of the materials occurs.

Eight carbon grills were connected with each of stainless steel (SUS) grills of the original LH launcher. Each carbon grill consists of a SUS base frame (10 mm thick), an RF conductor (~0.2 mm thick) and a carbon grill mouth (15 mm thick). The base frame was welded to the original grill. The RF conductor of a thin copper plate was used to improve electrical contact between the base frame and the carbon grill mouth. Each carbon grill mouth was held on the base frame by 22 bolts. The carbon mouth will be able to change when it is strongly damaged, even though the arc monitor system protects the LH launcher mouth from RF breakdown, which detected light emitted from the breakdown near the grill and immediately shut-off RF power.

After the modification of the LH launcher the launcher conditioning was substantially progressed both with and without plasma in 2004 operation. Injected energy, so far, has reached up to ~16 MJ into plasma, as shown in Fig. I.2.5-3, with a low reflection coefficient of 5 % by adjusting plasma position. A pulse modulation method was used to suppress RF breakdown owing to outgassing through temperature rise of the grills. Even in the conditioning phase, it was found that ~60 % of the plasma current of 1 MA was driven by LH injection by extracting the drop in one-turn loop voltage. The current drive efficiency is roughly estimated to be ~1.6 x 10^19 A/W/m², which is 50 - 70 % of that with the original launcher, and seems to be improved with progressing the conditioning. Thus the performance of the modified launcher shows sufficient abilities as a high power LH launcher. The technical key issue was to keep sufficient electrical contact for the LH antenna with the carbon grills, therefore a thin RF contactor made of copper was developed and inserted between the base frame and the carbon grill mouth. After the conditioning operation, severe damage by RF breakdown was observed around the base frames due to insufficient works of the arc monitor system. However, the contactor seems to trigger or to continue the RF breakdown. Therefore the contactor should be improved not to cause RF breakdown.

References
2.5-3 2.5-3 Seki, M., et al., “Performance of the LH Antenna with Carbon Grill in JT-60U,” to be published in Fusion Engineering and Design.

2.6 Diagnostic System
2.6.1 Infrared Imaging Video Bolometer
An infrared (IR) imaging video bolometer (IRVB) can provide a wide-angle view equivalent to hundreds of resistive bolometers. Radiation from the plasma is received with a metal foil absorber and the foil temperature is measured with an IR camera outside the vessel. Radiated power can be obtained by solving the two-dimensional heat diffusion equation numerically at each point in the foil. Recent progress in IR technology enables the sensitivity of the imaging bolometer to approach to that of the conventional resistive bolometer, as has been successfully demonstrated in the Large
Helical Device [2.6-1]. A feasibility study of the imaging bolometer under a tokamak environment was initiated as a research collaboration with NIFS in 2003 [2.6-2].

The IRVB is illustrated schematically in Fig. I.2.6-1. A 2.5 microns gold absorber foil of 9 cm x 7 cm was shown to be durable during two years of operation with 1800 tokamak discharges including disruptions. Taking advantage of the wide-angle view of the IRVB, an adjustment for semi-tangential view of the tokamak plasma was done by shifting the pinhole 15 mm horizontally. A radiating toroidal ring has been mapped and recorded onto the foil as a clear high temperature zone at a disruption, consistent with huge core radiation measured with the resistive bolometers. Radiation from the divertor could be identified also in the foil image as a thick line having toroidal curvature. This work was partly supported by Grants-in-Aid for Scientific Research of the JSPS, Nos.16560729/16082207.

2.6.2 Real-Time Evaluation Technique of Safety Factor Profile

In order to realize a real-time control system of the safety factor profile \( q(r) \), we developed, for the first time, a method to evaluate the safety factor profile in real-time [2.6-3] using a motional Stark effect (MSE) diagnostic [2.6-4]. The MSE diagnostic measures local magnetic pitch angles inside the plasmas. The newly developed method evaluates \( q(r) \) at 16 locations (maximum), within 10ms. A conventional method that solves the Grad-Shafranov equilibrium equation takes several tens of seconds to calculate \( q(r) \) in, although it is accurate. Instead of solving the Grad-Shafranov equation, the method employed here assumes that the last closed magnetic surface represents internal magnetic surfaces well. The \( q \) profile is calculated using pitch angle mapped on the internal magnetic surfaces.

Figure I.2.6-2 shows a safety factor profile evaluated with the real-time evaluation technique, in comparison to that by equilibrium reconstruction. The safety factor profile evaluated using the real-time technique agrees with the accurate profile obtained using the equilibrium reconstruction. This evaluation technique has now been built into the real-time control system of the safety factor profile.

2.6.3 Real-Time Processing of FIR Laser Interferometry for Long Pulse Discharges

A real-time processing (RTP) system has been newly developed for the FIR laser interferometer utilizing the compact-PCI modules and the software on the Real Time LINUX OS [2.6-5]. Using this system, real-time

![Fig. I.2.6-3. Typical waveforms of the density feedback control using the new RTP system. Here, \( I_p \) is the plasma current, \( \dot{n}_e \text{RTT} \) is the real-time line-integrated electron density measured with the new RTP system, \( Q_{\text{D2}} \) is the deuterium gas puff rate, and PNBI is the neutral beam injection power.](image)
correction of the fringe jump error, which is required for density feedback control in long pulse discharges, has become available. As a result, reliable density feedback control in the long-pulse discharges has been realized using this new RTP system (Fig. 1.2.6-3).

References
2.6-3 Suzuki, T., and JT-60 team, “Recent RF Experiments and Application of RF Waves to Real-Time Control of Safety Factor Profile in JT-60U”, Proc. Topical Conf. on RF power in Plasmas (Park City, 2005) in pr i n.

3. Design Progress of the National Centralized Tokamak Facility
Two designs proposed for the National Centralized Tokamak facility (NCT) is assessed with respect to the physics requirements such as break-even class plasma, heating and current drive capability, MHD stability, divertor performance, and plasma controllability. After the fruitful discussions with the scientists from universities and industries in Japan, the machine parameter with wider operational space in the plasma shape flexibility, which is regarded to be of importance for the achievement of high-$\beta$ plasma, is made to be a key factor for NCT. Engineering design of the main components of superconducting TF and PF coils, vacuum vessel, in-vessel facilities, and cryostat has been performed to optimize their structure from the view points of manufacturing processes, operation and maintenance feasibility.

3.1 Physics Design
The machine parameters are assessed from the view points of the capability to break-even class plasma, high-$\beta$ plasma, heat and particle controllability in divertor, flexibility of aspect ratio and plasma shaping, full current drive controllability [3.1-1].

Break-even class plasma of equivalent $Q_{\text{DE}}=1$ will be achievable with $I_p=5.5$ MA, $HH_{42}$=1.4, and the

![Graph](image)

Fig. I.3.1-1 Dependence of MHD stability on a parameter of normalized wall location evaluated by ERATO-J analysis. Four cases (aspect ratio $\Delta=2.5$ and 3 with $n=1$ and 2) are compared.
neutral beam power $P_{NB}=13$ MW. The consistency with a break-even class plasma and a high-$\beta$ plasma is estimated, i.e., $Q_{DT}=1$ and $\beta_N=3.5$ will be simultaneously achieved at $I_p/B_T=4.5$MA/2.3T, $q_{95}=3.5$, $HH_{q_9}=1.5$, $f_{GW}=0.9$, and NB power of 25MW. In such a condition, collisionless plasma with low normalized Larmor radius is satisfied in the range of $\rho^* = 0.005-0.008$, $\nu^* = 0.01-0.1$.

The accessibility of advanced operation of high-$\beta$ with full current drive accessibility at $I_p=3$ MA and $\beta_N=4$ was estimated by the ACCOME analysis on the assumption of $HH_{q_9}=2$, $q_{95}=6.1$, $q_{min}=2.0$, $f_{GW}=0.5$, with total NB power of 25MW in the case of negative NB at off-axis. Current profile control by the combination of on-axis and off-axis beams enables such an advanced operation scenario.

The advantage of low aspect ratio to the ideal MHD stability was evaluated by the ERATO-J code analysis [3.1.2]. Figure I.3.1-1 shows the dependence of the critical $\beta_N$ on a parameter of the normalized wall location, $r_w/a$ (r_w: wall location, a: plasma minor radius), by $n=1$ and $n=2$ toroidal modes in the double null reversed shear plasma with $\delta_{95}=1.8$, $\delta_{95}=0.4$, $q_{min}=2.4$, and parabolic pressure profile. In general, critical $\beta_N$ is higher in smaller $r_w/a$ due to the wall stabilization effect. As clearly seen in the figure, the critical $\beta_N$ tends to be lower in the low aspect ratio.

Preliminary analysis by ‘VALEN code’, under the collaboration with Columbia University and PPPL, was conducted to estimate the achievable $\beta_N$ with the active control of RWM stabilisation in the 3-dimensional geometry of vacuum vessel and the stabilising plates. The analysis indicates that the maximum achievable $\beta_N$ is about 3.8, and the limitation is brought by the weak coupling of the magnetic flux of the in-vessel coils with the plasma because of the shielding effect by the stabilising plates. The code analysis in the ITER geometry predicts that the coupling could be effectively enhanced if the in-vessel coils are located around the port duct.

Controllability of the EC resonance for the NTM suppression was estimated by modified Rutherford equation. The minimum EC power for the stabilization of $m/n=3/2$ and 2/1 mode in the resonance of fundamental EC wave injected with 90 GHz, O-mode into the normal shear plasma with $q_s=1$ is 0.51 MW, and 1.1 MW, respectively. Those requirements meet the present EC design.

Simulation analysis for divertor particle and heat flux controllability was performed using with SOLDOR/NEUT2D code. In the ITER-like divertor configuration with long leg length, a partial detachment is well maintained. Parameter surveys in the incline angle of the divertor plate and the distance between the pumping duct and the hit point on the plate were also made in order to optimize the divertor geometry. On the other hand, in the optimized shape configuration with a low aspect ratio and a high triangularity, the shortening of the leg length of the inner divertor and the insufficient cryopanel surface area bring the degradation of the pumping capability and of the particle controllability. Further optimization in both the aspects of the divertor pumping and of the plasma shaping is required.

3.2 Engineering Design
Based on the design with wider operational space, structures of main components and manufacturing processes were reviewed to optimize the space utility and maintenance, especially around the midplane area. Whole assemble of the NCT tokamak is illustrated in Fig. I.3.2-1.

3.2.1 TF and PF coils
In order to ensure the space margin for the extension of the flexibility in the aspect ratio and plasma shape, TF coil was been enlarged in the vertical direction. Each TF coil has 114 turns to correspond the maximum $B_T R$ of 8.11 Tm. The numbers of turns of PF coils are increased to realize the maximum plasma current of 5.5 MA for 100 s. Support structure of the CS and the divertor coil is unified with that of the TF coil in order to cancel out the mechanical stress by the electromagnetic force as an internal force.

3.2.2 Vacuum vessel
The total width of the double-wall is designed as 148 mm from the view points of the reinforcement. By the stress analysis with FEM code the interval of the ribs is determined to 300 mm with the welding depth of 24 mm.

3.2.3 Stabilizer plates
In order to compensate the thermal stress during the
baking of vacuum vessel, crank support structure is adopted. Support leg was made of SUH660 with the electrical insulation coating at the crank-pin and the joint part of crank support. Strength of such a structure was confirmed by the stress analysis with the temperature difference of 300°C between the vacuum vessel. Mechanical strength was also confirmed against the electromagnetic force during disruption event including a halo current.

3.2.4 Divertor
Movable louver or sliding shutter in order to adjust the divertor pumping speed during the plasma discharge was designed. It is located in front of the cryopanel under private dome or outer baffle plates. The effective pumping speed for deuterium gas was estimated as 100% to 10% due to the change of the conductance of the adjustable louver or shutter in 100 m³/s to 1 m³/s within the duration of about 1 s. Strength of the structure was confirmed by the stress analysis against the thermal stress and the electromagnetic force.

3.2.5 Cryostat
New design of the spherical cryostat is developed in order to ensure the enough space for maintenance in the joint area with NB injection port. It consists of upper pan, middle vessel, lower pan and support base. Each block is connected by the flanges with the lip seal to maintain the vacuum condition. Stress analysis performed by a 3D-model indicates that each part of the cryostat satisfies the structural strength against the complex load from electromagnetic force and seismic force.

3.2.6 Bending strain of Nb₃Al CICC
In order to estimate the effect of bending strain on the critical current (Iₚ) of Nb₃Al cable in conduit coil (CICC) [3.1-3], a test facility for loading the tensile and compressive stresses was designed and manufactured. The loading test was performed on Nb₃Al strand (strand sample) wound around the spring-shape holder, and two Nb₃Al strands and one Cu wire inserted into a stainless steel conduit (triple CIC sample). Iₚ of the strand was measured with the strain range from -0.86% to +0.18% at 4.2 K in the external magnetic field of 6-11 T. The dependence agreed well with the theoretical prediction by Durham's equation [3.1-4]. Based on those results some relaxation mechanism of bending strain in the conduit will be investigated.

3.2.7 Shielding material
The typical performance of the heat proof boron-doped neutron shield resin, developed last year, was examined [3.1-5]. The same level of the neutron shielding characteristic as that of polyethylene was confirmed by the penetration tests of 2.45 MeV DD-neutrons and of the continuous energy neutrons from ²⁵²Cf source. The heatproof temperature determined by the deflection load was about 300°C. The tensile, bending, and compressive tests based on the JIS standard show the enough mechanical strength both at room temperature and at 250°C. The resin is suitable for NCT to set up around the port section to suppress the streaming neutron and at the neutron shielding material for diagnostics systems around the vacuum vessel.

References
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II JFT-2M PROGRAM

A series of the experimental programs on the JFT-2M was completed in the last fiscal year. In this fiscal year, experimental data on the Advanced Material Tokamak Experiment (AMTEX) using the reduced activation ferritic steel (F82H), high performance experiment, characteristics of SOL and divertor plasma and compact toroid injection for fueling have been analyzed and evaluated. These results were presented in the 20th IAEA Fusion Energy Conference held at Vilamoura, Portugal in 2004, and submitted to journals for publication.

Concerning the AMTEX, analyses of high-beta experiments with the Ferritic Inside Wall (FIW) facing close to the plasma have apparently shown a wall stabilization effect. Moreover, by using an MHD equilibrium code in which the calculation accuracy in plasma pressure has been improved by raising the level of approximation for ferritic segments, it has been confirmed that a plasma with the normalized beta value of $\beta_n \sim 3.5$ is compatible to FIW.

As part of research program for understanding mechanism to improve plasma confinement, studies on fluctuations of electric potential in the transport-barrier region of H-mode have been made by collaboration with universities etc. Results have shown that there are two types of turbulent fluctuations, one is suppressed when the confinement is improved and the other helps to sustain the transport barrier. Moreover, the field structure has been analyzed by identifying the low frequency geodesic acoustic wave, and it has been shown for the first time that this mode influences the background turbulence and the turbulent particle flow. A comparison of experimental conditions of the new H-mode regime found in JFT-2M (HRS H-mode), which has an attractive performance for steady-state operation, with those of the EDA H-mode in Alcator C-Mod has suggested that there is a common physical process between them. On the edge plasma research, it has been found that the heat flux to the divertor plates in the ELM free HRS H-mode is reduced to about 15% of the maximum heat load due to ELMs in the ELMy H-mode.

On the research for fueling by compact toroid injection, magnetic fluctuations observed just after the CT injection have been analyzed. Results have shown that the fluctuation frequency approximately agrees with that of Alfvén wave of theoretical prediction and that the life time of CT is well reproduced by a simulation using a slow magnetic field reconnection model. The process of fueling by compact toroid injection has been well understood by these results.

1. Advanced Material Tokamak Experiment (AMTEX) Program

The reduced activation ferritic steel is a leading candidate of structural material for the blanket of a fusion demonstration reactor (DEMO). However, it is ferromagnetic material and it easily rusts in the air. Thus the investigation of the compatibility of the ferritic steel with plasma is important and has been investigated on the JFT-2M tokamak step by step [1-1, 1-2]. Since 2002, the inside vacuum vessel wall has been fully covered with the Ferritic Inside Wall (FIW) [1-2, 1-3]. The compatibility of the ferritic wall with high normalized beta ($\beta_n$) plasma is an important issue because high normalized beta plasma of $\beta_n = 3.5 \sim 5.5$ must be realized for a commercially attractive fusion reactor by using wall stabilization effect. Thus, high beta experiments with the close wall position were carried out by changing the plasma position in the final experimental campaign of JFT-2M. The experimental data has been analyzed more quantitatively in this year.

The equilibrium calculation is a key evaluation tool for this analysis because both the normalized beta and wall position are evaluated with this code. The calculation without including the ferromagnetic effect might contain systematic error because the magnetic sensors are affected by the magnetic field from the ferritic steel. The code including the ferromagnetic effect was developed a few years ago, but it was
unstable. Thus, the equilibrium code has been checked more precisely and improved [1-4].

At first, the model of the profile of plasma current and pressure was optimized. The realistic results were obtained reproducibly after this modification. The model of FIW was also improved. The ferromagnetic effect was modeled by placing the filament current on the surface of the wall. This method is suitable to represent the magnetic field structure at a relatively far region from the wall. On the other hand, the magnetic field near the wall is sensitive to the model e.g. density and position of the filaments. Since the probe position (24 B{sub b} probes and 8 flux loops on vacuum vessel and/or FIW) is very near to the wall, dependence of the magnetic field strength at the probe position on the model was investigated. After the optimization of the model, the obtained magnetic field profile at the probe position became more smooth and reliable. This modification caused ~5 % increase in obtained beta value. To investigate the separatrix position more precisely, new method was developed last year to measure separatrix position using two sets of the step probe as follows [1-5, 1-6]. Two sets of double probe with different pin length (5 mm difference) are inserted into the plasma. Direction of the voltage is opposite for the probes. When the probe passes across the separatrix, the difference of the probe current becomes maximum (see 2.3.2). The obtained peak position agreed well with that obtained from the equilibrium calculation within ~2 mm. If the separatrix position is estimated with the code without considering FIW, the obtained position is 12.7 mm inside the peak position. These results have shown that the effect of the FIW is properly included in the equilibrium code. The beta value estimated with the code was compared with the diamagnetic signal, and a linear relation was shown between them. It should be noted that the comparison is not self-consistent because the calculation of the beta value from the diamagnetic signal also needs the shaping factor calculated in the equilibrium code. However, the reliability of the code has been demonstrated for certain degree (at least for relative value) and the improved code has been employed in following analyses and discussions.

Figure II.1-1 shows the normalized beta just before the collapse against the normalized wall position (r_{wall}/a) for all effective shots. The data obtained 2 years ago are also shown in the figure. Due to the improvement of the operation scenario and the hardware, the operational region was extended to \( r_{wall}/a \sim 1.25 \). Thus, the compatibility of the FIW with DEMO-relevant high normalized beta plasma was demonstrated [1-4]. Scattering of the data is considered to be attributed by the difference in pressure and temperature profiles, caused by the difference in the electron density, the wall condition and so on.

To investigate the wall effect, discharges with similar condition were carefully chosen. They were taken on a same day with keeping radiation level at \( \sim 300 \text{ kW} \) during full power NB injection of 1.6 MW. The electron density is \( n_e/n_{GW} \sim 0.5 \) for co-injection phase and collapse occurs at \( n_e/n_{GW} \sim 0.6 \), where \( n_{GW} \) is the Greenwald density. Other parameters related to plasma stability are summarized in Table II.1-1. The parameters are almost reproducible. Another important feature is the behavior of soft X-ray profiles before the collapse. In this series of experiment, a sharp outward shift of the profile was observed, which was clearly different from the behavior of tearing mode disruption. These data suggested that the target plasmas and mechanism of the

<table>
<thead>
<tr>
<th>( r_{wall}/a )</th>
<th>1.48</th>
<th>1.36</th>
<th>1.26</th>
</tr>
</thead>
<tbody>
<tr>
<td>( R_{min} (m) )</td>
<td>1.35</td>
<td>1.38</td>
<td>1.39</td>
</tr>
<tr>
<td>( n_e/n_{GW} )</td>
<td>0.56</td>
<td>0.66</td>
<td>0.62</td>
</tr>
<tr>
<td>( \ell_i )</td>
<td>0.67</td>
<td>0.74</td>
<td>0.74</td>
</tr>
<tr>
<td>q_{95}</td>
<td>3</td>
<td>3</td>
<td>3</td>
</tr>
<tr>
<td>( P_{rad} (kW) )</td>
<td>300</td>
<td>260</td>
<td>330</td>
</tr>
<tr>
<td>( H_{e0} (a.u.) )</td>
<td>0.9</td>
<td>0.75</td>
<td>0.88</td>
</tr>
</tbody>
</table>

Fig. II.1-2. Time evolutions of normalized beta for similar plasma condition but different in wall position.
collapse were almost reproducible and only the wall position was scanned. Figure II.1-2 shows time evolution of the normalized beta for discharges shown in Table II.1-1. The waveforms are almost identical before 480 ms. A plasma in the configuration closer to FIW survives longer, and thus, reaches the higher normalized beta. It might correspond to the wall stabilization effect. A Clear difference was observed in magnetic probe signal [1-3, 1-4]. In both cases, n=1 mode (n is toroidal mode number) was observed with eight B₀ probes toroidally distributed at outer mid-plane as shown in Fig. II.1-3. The position of the mode locking is reproducible, which means that the mode locking is related to the external error field. In order to compare growth rate, the difference between signal intensities of two magnetic probes located toroidally on the opposite side is plotted in Fig. II.1-3(a) for the 3 discharges. The growth rate was estimated from the gradient and summarized in Fig. II.1-3(b) as a function of r_{wall}/a. The growth rate is smaller for the closer wall position. Similar to the resistive wall without ferromagnetism, reduction of the growth rate from the Alfvén time scale to wall time constant (a few milliseconds) was observed with ferritic wall. Thus, it has been concluded that the ferromagnetic wall shows similar behavior as normal resistive wall and the adverse effect related to ferromagnetism is not observed at least in this experimental condition. It should be noted that the thickness of the ferritic steel in this experiment (~10 mm) is much smaller than that of DEMO, but the normalized effect is expected to be comparable because of the larger minor radius and the strong toroidal field in DEMO. Thus, the compatibility of the ferritic wall with reactor-relevant high normalized-beta plasmas has been demonstrated.

References

2. High Performance Experiments
2.1 Study of H-Mode Physics
The analysis of the experimental data of JFT-2M made a progress on the confinement physics of the tokamak plasma under the collaboration with the Univ. of Tokyo and the National Institute for Fusion Science.

On the study of the stability of the edge transport barrier (ETB) during the H-mode, some characteristic electrostatic MHD oscillations were found. The typical H-mode in JFT-2M evolves in time as the three stages as 1) ELM-free H-mode, then 2) slightly degraded H'-mode, and at last 3) EDA-like H-mode. In the third stage, the ETB seems steady for a few hundred milliseconds in a high recycle edge state. We studied the characteristic MHD instabilities at the ETB during each stage [2.1.1, 2.1.2]. In the ELM-free phase, as the ETB pressure increases, reflectometer measurement shows an
The interaction of the GAM and the background turbulent density fluctuations in the edge region of the core plasma measured by HIBP. (a) GAM potential, (b) Spectra of the density fluctuation, (c) GAM potential (time expanded) (d) density fluctuation.

appearance of characteristic electrostatic multi-modes which have frequencies around 150-200 kHz. Only the modes with high toroidal mode number around 8-11 are unstable, the feature of which is similar to that of a ballooning mode [2.1-3]. The mode frequencies decrease continuously as the mode grows and in the successive H'-mode phase, the modes seems to unite to a single quasi-coherent mode of frequency ~70 kHz. In the high recycle EDA-like H-mode phase during which the edge $D_e$ level is high, the glowing electrostatic quasi-coherent mode co-exists with the higher frequency electromagnetic mode of ~300 kHz. Remarkably different part in the density fluctuation spectra between the ELM-free H-mode and the H'-mode is attributed to the quasi-coherent mode. By the heavy ion beam probe (HIBP) measurement the plasma potential at the ETB fluctuates with the quasi-coherent mode frequency during the H'-phase and the radial electric field is found to be reduced. Therefore, the quasi-coherent mode seems to decrease the radial electric field and degradation of the confinement occurs, which may contribute to the steadiness of the ETB during the H'-mode. Thus a control of these MHD instabilities at the ETB is found to be very important to improve the quality or steadiness of the H-mode. Also, it has been found that suppression of the turbulent density fluctuations below ~50 kHz in the ETB is a feature of these H-modes when compared to the L-mode [2.1-1].

We found a low frequency electrostatic coherent mode of 10-15 kHz in the Ohmic heating phase [2.1-4] and in the L-mode phase [2.1-5]. The characteristic mode is identified as the geodesic acoustic mode (GAM) which has a large wave length along the poloidal direction, and is a kind of the zonal flow. The GAM and the zonal flow have grown to important topics to study the mechanism of the transport improvement (at the internal transport barrier (ITB)) or of the reduction of the anomalous transport induced by the plasma turbulence. We clarified the structure of the electric field of the GAM [2.1-5]. The GAM seemed to interact with the background turbulence in parametric/modulational nonlinear fashion [2.1-4] and the power of the density fluctuation (background turbulence) was found to fluctuate with the potential fluctuation of the GAM (Fig. II.2.1-1) [2.1-5].

References

2.2 High Recycling Steady H-Mode
Finding alternative scenarios to Type-I ELMmy H-mode operation is a key area of research for current tokamaks. A new attractive operational regime without any large ELMs, “High Recycling Steady (HRS)" H-mode regime, was discovered in the JFT-2M tokamak that had many similarities to EDA H-mode regime on Alcator C-Mod [2.2.1-6]. To compare the properties, fluctuation behavior and access conditions of these regimes, a series of parameter scans were carried out in $q_{95}$-V_e.
is electron density, \( \ln \Lambda \) is Coulomb logarithm, \( T_e \) is electron temperature, and \( \varepsilon \) is inverse aspect ratio. As a result, a striking similarity of access conditions was seen as shown in Fig. II.2.2-2. On both devices, most ELMy H-mode regime having large ELMs was clearly classified as the collisionless regime of \( \nu_e^* < 1 \). On the contrary, HRS/EDA regimes are found to be a high collisionality phenomena of \( \nu_e^* \geq 1 \). A “Mixture” regime exists near the operational boundary at around \( \nu_e^* \sim 1 \). At higher \( q_95 \) (~5) on JFT-2M, it seems to be more easily to access the pure “HRS” regime at around \( \nu_e^* \sim 1 \), while the “Mixture” regime extends its operational regime toward a high collisionality regime of \( \nu_e^* \geq 1 \) at lower \( q_95 \) (~3). These results imply the importance of both \( q_95 \) and \( \nu_e^* \) to understand the access conditions for the HRS/EDA regimes. On the other hand, several differences in the edge fluctuations were found. Detail comparison will be reported in the near future.

References

2.3 Study on Divertor and Scrape-Off Plasma
2.3.1 Comparison between Divertor Heat Loads in ELMy H-mode and HRS H-mode
Recently, HRS H-mode [2.3-1] was shown to strongly reduce the ELM activity, and thus to have a potential for eliminating the severe ELM heat load on the divertor target. We compared the divertor heat load between the ELMy H-mode and the HRS H-mode [2.3-2].

Figures II.2.3-1(a) and (b) show time evolutions of \( D_o \) emission in the divertor \( (D_o, \delta D_o) \), electron temperature \( (T_e) \) and ion saturation current \( (I_s) \) around the strike points on the divertor targets at ELMy and HRS H-mode discharges. During the ELM event, \( T_e \) and \( I_s \) increases by 2-4 and 5-6 times, respectively.
compared with that just before its generation. This result shows that large heat and particle fluxes arrive abruptly on the diverter targets. On the other hand, a transition to the HRS H-mode occurs at \( t \sim 0.645 \text{ sec} \) after a brief ELM-free phase with enhanced \( D_\alpha \) emission. The electron temperature remains at a low level \((-10 \text{ eV})\), whereas \( I_e \) increases in a manner similar to the \( D_\alpha \) enhancement. These findings show that the heat load in the HRS H-mode is induced dominantly by an enhancement of particle transport.

Fig. II.2.3-1 Time evolutions of the \( D_\alpha \) emission in the divertor (\( D_{\alpha,\text{av}} \)), electron temperature (\( T_e \)) and ion saturation current (\( I_e \)) around the strike points on the divertor targets at ELMy and HRS H-mode discharges. Experimental conditions are \( I_p=0.24-0.27 \text{ MA} \), \( B_t=1.6 \text{ T} \), \( q_{95}=2.6-2.7 \), \( n_e=4-5 \times 10^{19} \text{ m}^{-3} \) and \( P_{\text{NBI}} = 1.1-1.4 \text{ MW} \) at lower single plasmas. (From Ref. [2.3-2]).

For the ELMy H-mode and HRS H-mode, the heat load (\( \Gamma_{\text{heat}} \)) on the divertor target is evaluated from \( \gamma J_e T_e \), where \( J_e \) is the ion saturation current density. The heat transmission coefficient \( \gamma \) is assumed to have a value of 7 [2.3-3]. It may change depending on the NBI power [2.3-4]. However, we consider that the change in the experiments is small because of the relatively low NBI power of 1.1-1.4 MW. As a result, the heat flux in the HRS H-mode phase becomes \( \Gamma_{\text{heat}} \sim 0.3 \text{ MW/m}^2 \) around the strike points. It corresponds to \( \sim 15\% \) of the peak value during an ELM event. From this result, we have confirmed an attractive feature of lower heat load in the HRS H-mode than that in the ELMy H-mode.

2.3.2 Direct identification of magnetic surface by the step probe in JFT-2M

In tokamaks magnetic surfaces are estimated from the magnetic measurement by using magnetic probes equipped on the vacuum vessel with aid of the tokamak equilibrium code. However, the calculated results would contain a systematic error when ferritic steel, which affect the magnetic measurement, is installed near the magnetic probe. The deviation of the separatrix position is estimated to be about 15 mm outer to the low field side than the calculation without considering the ferritic steel. In order to identify the deviation experimentally, we measured the absolute location of the separatrix directly by using two sets of the step probe, which is a double probe having a long and a short electrodes. The probe current of the step probe is expressed as

\[
\iota = \frac{a - \frac{1}{2} + \frac{\alpha + 1}{2} \Delta \eta - \Delta \eta - \Delta \eta_{\text{b}}}{2}
\]

(II.2.3-1)

where \( \alpha = n_c / n_\infty \), \( \eta_{\text{b}} = eV_{\text{b}} / kT_e \), \( \Delta \eta = \Delta V_e / kT_e \), and \( n_c, n_\infty \) is the density of core and the SOL plasma, respectively, \( V_{\text{b}} \) is the difference of the two step probe voltages, \( \Delta V_e \) is the difference of the space potential between the core and SOL plasma. Negative bias voltage to the long electrode so as to \( \eta_{\text{b}} \gg \Delta \eta \) and positive bias voltage to the short one so as to \( \eta_{\text{b}} \gg \Delta \eta \) are applied alternately. The time evolution of two probe currents is detected with the current probes, and the difference of them is measured. The difference of the probe currents has a peak when the probe head passes across the separatrix. The separatrix position was measured by jogging the plasma so as the step probe head to cross the separatrix. The measurement has shown that the observed separatrix point is shifted 12.7 mm outward from that calculated with the equilibrium code EQFIT. On the other hand, it had been found that the correlation of the floating potential fluctuations disappears at the separatrix position when they are measured with one set of the step probe. We measured the correlation by using the two sets of the step probe mentioned here. The result has indicated that the separatrix position is shifted 12.5 mm outward from the calculation one, which supports the result using the probe current measurement [2.3-5].

References
2.4 Compact Toroid Injection
Compact toroid (CT) injection is an advanced method of the particle fueling into the plasma, and has been investigated on JFT-2M with collaboration between JAERI and University of Hyogo (former Himeji Institute of Technology), and Hokkaido University. In 2004, we have studied the magnetic fluctuation induced just after the CT injection because the magnetic fluctuation is possibly concerned with the fueling process of CT. This fluctuation is observed after the CT injection and lasts for 30 - 40 μs corresponding to the time scale $\Delta t_{\text{ej}}$ of CT injection. Time-frequency analysis has shown that this magnetic fluctuation has the maximum spectral amplitude at 250 - 350 kHz. It had been found that Alfvén wave was excited by the CT injection [2.4-1]. The Alfvén frequency $f_A$ is given by $f_A = \nu_A/(2\pi qR)$, where $q$ is a safety factor and $\nu_A$ is the Alfvén speed $B(I_{\text{emu}})^{1/2}$. In a typical case of $\nu_A \sim 6 \times 10^6$ m/s, $q = 2 - 3$ and $R \sim 1.5$ m, $f_A$ is 210 - 320 kHz, which is well agreed with the observed frequency. Figure II.2.4-2 shows contour plots of the profile of toroidal and poloidal magnetic field measured with magnetic probes. The fluctuation excited by the injected CT propagates first in the toroidal direction (total toroidal length: $2\pi qR = 8-10$ m) within a few μs, and then its response can be also seen in the poloidal direction after ~10 μs. As shown by the time-expanded figure in Fig. II.2.4-2, the fluctuation propagates toroidally in both clockwise (CW) and counter-clockwise (CCW) directions at a velocity $v_\phi \sim 3-4 \times 10^6$ m/s. Such a fast propagation velocity in the toroidal direction agrees with $v_\phi$, but is two order of magnitude larger than the ion thermal speed $v_\text{th}$ of CT $\sim 3 \times 10^6$ m/s, where we assume that the ion temperature is equal to the conductivity electron temperature $T_{\text{cond}} \sim 10$ eV as calculated from the resistive decay time $\tau_\text{R}$ of ~20 μs [2.4-2]. These results suggest that the CT could deposit its fuel particles along the magnetic field line at first through the resistive decay process of the magnetic field and/or the β-limit collapse possibly caused by the compression due to the tokamak field pressure [2.4-3].

References
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III. THEORY AND ANALYSIS

The principal objective of theoretical and analytical studies is to understand the physics of tokamak plasmas. Much progress was made in transport simulation of current hole plasma, role of low order rational q-values in the ITB events, the theory of Alfvén eigenmodes in tokamaks and current spike behaviour of disruptive plasma, external MHD modes, which is the basis of resistive wall mode (RWM) and MHD ballooning mode stabilization by plasma rotation.

Progress has been made in the NEXT (Numerical EXperiment of Tokamak) project to investigate complex physical processes in transport and MHD phenomena. Formation of zonal flow and streamer in the toroidal electron temperature gradient (ETG) turbulence was examined in detail. Confinement improvement was shown by controlling the zonal flow driven by the ion temperature gradient (ITG) turbulence. The formation and sustaining process of current hole are shown by the MHD simulation.

1. Confinement and Transport

1.1 Transport Simulation of Current Hole Plasma In JT-60U

Profile formation and sustainment of the current hole (CH) plasma have been investigated by using 1.5D transport simulations. A model of the current limit inside the CH on the basis of the Axisymmetric Tri-Magnetic-Islands equilibrium is introduced into the transport simulation. We found that a transport model with the sharp reduction of anomalous transport in the reversed-shear (RS) region can reproduce the time evolution of profiles observed in JT-60U. Figure III.1.1-1(a) and (b) show profiles of $T_i$, $T_e$, $n_e$, j and q at 2 s after the simulation start. Each profile agrees well with the experimental one. The internal transport barrier (ITB) is formed in the RS region. Inside the CH region, profiles of $T_i$, $T_e$ and $n_e$ are nearly flat. A radius at the ITB foot position, $\rho_{\text{ct}}$ is almost the same as that at the minimum q surface, $\rho_{\text{qmin}}$. The transport becomes neoclassical-level in the RS region, which results in the autonomous formation of profiles with ITB and CH. The ITB width and the energy confinement inside the ITB agree well with JT-60U scalings. The plasma with small bootstrap current fraction shrinks due to the penetration of inductive current. This shrink is prevented and the CH size can be controlled by the appropriate external current drive (CD). The CH plasma is found to respond autonomously to the external CD [1.1-1].

Fig. III.1.1-1 Profiles of (a) $T_i$, $T_e$, $n_e$, and (b) j, q at 2 s after simulation start. Symbols denote experimental data points.

Reference


1.2 Role of Rational q-values in the ITB Events in Reverse Shear Plasmas

Non-local confinement bifurcations inside and around internal transport barriers (ITBs) with a ms timescale (ITB events) have previously been found in JT-60U reverse shear (RS) and high-$\beta_p$ plasmas. ITB events are observed as the simultaneous rise and decay of the electron temperature $T_e$ in two zones. They are created by an abrupt non-local reduction (or increase) of heat flux inside 30-40% of the minor radius. Under sufficient neutral beam power $P_{\text{NB}}$ (above ~8MW for the $I_p/B_t$ = 1.2-1.5/MA/3.8T pulses described below), ITB events were previously detected at various $q_{\text{min}}$ values ($q_{\text{min}}$ is the minimum safety factor in the RS configuration). However, the role of $q_{\text{min}}$ equal to 3.5, 3, 2.5, 2 is not obvious for ITB formation. We here focus on new features of ITB evolution near low-order-rational values of $q_{\text{min}}$. The formation of a stronger ITB and its further splitting into two radially separated ITBs is described. These ITBs are located in both positive and negative shear zones of a plasma with L-mode edge. The
similarity of space-time evolution of the electron temperature $T_e$ and the ion temperature $T_i$ at sufficient power is highlighted (even when the variation is significant and complicated in space and time). Within errorbars, ITB splitting occurs as $q_{\text{min}}$ passes through 2.5. The similarity of space-time evolution of $T_e$ and $T_i$ suggests a similarity in the qualitative behavior of electron and ion heat diffusivities in time and space. The temporal formation of an ITB in the zone with small positive shear, while $q_{\text{min}}$ passes through 3 (after periodical improvements and degradations via ITB events with 8 ms period) in H-mode, with $P_{\text{th}} = 8$ MW, is described. At lower powers, ITB events are observed only near rational values of $q_{\text{min}}$. In weak RS shots with $P_{\text{th}} = 4$ MW, transport is reduced via ITB events during 0.08 s at $q_{\text{min}} = 3.5$, and repetitive short-term phases of reduced transport are observed as $q_{\text{min}}$ passes through 3. The behavior of $T_e$ looks different. The difference in $T_e$ and $T_i$ evolution, which was detected regularly under low power, probably indicates a decoupling of $T_e$ and $T_i$ transport. [1.2-1].

Reference

2. MHD Stability
2.1 Development of the theory of Alfvén eigenmodes in tokamaks

New magnetohydrodynamic (MHD) phenomena with upward frequency-sweeping named Alfvén Cascades (ACs) were revealed recently on JT-6U and JET in discharges with non-monotonic safety factor (reversed magnetic shear, (RS)) and significant population of the hot ions. The first theoretical description of ACs was given in the paper [Berk et al., Phys. Rev. Lett. 87, 185002 (2001)], where radial localization of Alfvén Eigenmode (AE) was provided by the non-resonant response of hot ion population. Another mode localizing toroidal MHD effect was considered by Breizman et al. [Phys. Plasmas 10, 3649 (2003)].

We extended the theory of AE in RS tokamak plasmas (RSAE) by incorporating the effect of thermal plasma density gradient taken from theory of cylindrical Global Alfvén Eigenmodes (GAE) and kinetic (finite ion Larmor radius) effects from theory of Kinetic Toroidicity induced Alfvén Eigenmodes (KTAE).

It was shown that the localization effect of thermal plasma density gradient on AC mode can be stronger than the toroidal MHD effect as squared aspect ratio. Thus, the Alfvén Cascade modes can be theoretically demonstrated in cylindrical geometry approximation. Then the role of thermal plasma density gradient can be dominant if the localization effect of density gradient of large orbit hot ions is sufficiently weak. This effect was found to be localizing for the mode numbers satisfying the condition $q_{\text{min}} > m/n$ and delocalizing otherwise. The shift of the localization region of the eigenmodes and the eigenfrequency shift caused by the thermal plasma density gradient were found to be sufficiently small. [2.1-1]

Taking into account the finite ion Larmor radius (kinetic) effects in Alfvén mode equation allows us to predict a new branches of these modes called the Kinetic Reversed-Shear Alfvén Eigenmodes (KRSAEs). These modes are shown to possess the features of Alfvén Cascades even for homogeneous thermal plasma density in cylindrical geometry approximation. It was found that for existence of the radially localized mode structure the requirements on the hot ion density and its gradient are less strict than those for traditional RSAE. Thus it is possible that KRSAE can be more unstable than original RSAE. [2.1-2]

We studied continuum damping of RSAE and calculated rigorously the damping rate. The damping rate depends in a complex way on the hot ion density and its gradient. Therefore the damping condition should modify RSAE onset threshold as well. [2.1-3]

In the approximation of the large orbits of hot ions the dominant hot ion contribution to the RSAE equation comes from the electrons compensating the charge of the fast ions (indirect fast ion effect on the mode). Incorporating this effect in the theoretical description of the TAE modes in discharges with monotonic (positive) shear revealed new varieties of TAEs with different energy, eigenfrequency and parity of the modes. [2.1-4, 2.1-5]

The cross field drift of the compensating electrons was shown to be of crucial importance in the theory of Energetic Particle Mode (EPM) to provide quasineutrality of perturbations. In the limiting case when compensating electron effect is dominant, the unstable EPM modes were found to be substituted by
the new kind of eigenmodes, called Compensating Electron Alfvén Eigenmodes which are heavily damped due to continuum dissipation and resonance interaction with fast ions. Therefore in general, the picture of Alfvén instabilities excited by energetic ions in discharges with monotonic safety factor is more favorable than that predicted by the EPM theory. [2.1-6]

References

2.2 TSC Simulation on Current Spike Behavior of JT-60U Disruptive Plasmas
Characteristics and underlying mechanisms for plasma current spikes, which have been frequently observed during the thermal quench of JT-60U disruptions, were investigated by simulations of Tokamak Simulation Code (TSC) including the passive shell effects of the vacuum vessel. Positive shear and reversed shear (PS and RS) plasmas were shown to have various features of the current spike in the experiments, e.g. an impulsive increase in the plasma current (positive spike) in the majority of thermal quenches, while sudden decrease (negative spike) that has been excluded from past consideration as an exception. It was clarified for the first time that the shell effects, which become significant especially at a strong pressure drop due to the thermal quench of high $\beta_p$ plasmas, play an important role in the current spike in accordance with the relation of the radial location between the initial plasma equilibria and the vacuum vessel. As a consequence, a negative current spike may appear when the plasma is positioned much further out than the geometric centre of the vacuum vessel, whereas a positive current spike may appear when the plasma is positioned much more inward than the vacuum vessel centre (a broken line in Fig.III.2.2-1). It was also pointed out that a further lowering of the internal inductance, in contradiction to previous interpretation made in the past, is a plausible candidate for the mechanism for positive current spikes observed even in the RS plasmas (arrows in Fig.III.2.2-1). The new interpretation of the shell effects and the abrupt change of the current profile enables us to reason out the whole character and the underlying mechanism for a variety of current spikes of major disruptions as well as the minor disruptions in JT-60U [2.2-1].

Fig. III.2.2-1 Normalized current spikes $\delta i_p (\equiv \delta i_p / i_{p0})$ observed in JT-60U PS (●) and RS (▲) disruptive discharges versus plasma radial position $R_j$. The broken line indicates a TSC simulation. Notice that the abrupt change in the current profile due to the 'magnetic braiding' (denoted by arrows), together with the newly found shell effects due to the $\beta_p$ drop (denoted by the broken line), explains the whole character of the current spikes observed in JT-60U disruption experiments.

Reference

2.3 Critical $\beta$ Analyses with Ferromagnetic and Wall Geometry Effects
The critical beta, that is limited by Alfvén wave time scale external kink mode, is shown to be decreased by ferromagnetic effect by about 8 % for m/m₀=2, m and m₀ denote the permeability of ferromagnetic wall and vacuum, respectively, for tokamak of aspect ratio 3. The existence of the stability window for resistive wall mode opened by both effects of the toroidal plasma rotation and the plasma dissipation, which was not observed for high aspect ratio tokamak, is found for tokamak of aspect ratio 3. The effect of ferromagnetism on them is also investigated. The critical beta analyses of NCT (National Centralized Tokamak) plasma using VALEN code are started with stabilizing plate and vacuum vessel geometry with finite resistivity, and the
results for passive effect of stabilizing plate are obtained. The calculations including stabilizing effect of the vacuum-vessel and also active feedback control are also performed for present design of NCT plasma.

![Graph showing growth rate and mode frequency as a function of normalized wall radius for three relative permeability where the toroidal rotation velocity is 0.5 times poloidal Alfvén velocity.]

Fig. III.2.3-1 Growth rate and mode frequency as a function of the normalized wall radius for three relative permeability where the toroidal rotation velocity is 0.5 times poloidal Alfvén velocity.

Reference

2.4 Stability Analysis of External MHD Modes by the 2-D Newcomb Equation

The theory of the Newcomb equation has been applied to low-n external modes in a tokamak and a method has been developed for computing the stability matrix that gives the change of plasma potential energy due to external modes in terms of the surface values of the perturbations[2.4-1]. By using this method, the spectral properties of the ideal external modes have been elucidated, such as the coupling between external modes and internal modes, and the difference in the stability properties between a normal shear tokamak and a reversed shear tokamak. These results will also be useful in the stability analysis of resistive wall modes.

References

2.5 Rotational Stabilization of High-n Ballooning Modes

A ballooning perturbation in a toroidally rotating tokamak is expanded by square-integrable eigenfunctions of an eigenvalue problem associated with ballooning modes in a static plasma[2.5-1]. A special weight function is chosen such that the eigenvalue problem has only the discrete spectrum. The eigenvalues evolve in time owing to toroidal rotation shear, resulting in a countably infinite number of crossings among them. The crossings cause energy transfer from an unstable mode to the infinite number of stable modes; such a transfer works as the stabilization mechanism of the ballooning modes. A simple analytic formula is derived for estimating the toroidal rotation shear required to stabilize the ballooning mode.

References

3. Numerical Experiment of Tokamak (NEXT)

3.1 Structure Formations in Toroidal ETG Turbulence

Using a gyrokinetic toroidal particle code with global profile effects, the toroidal electron temperature gradient driven (ETG) turbulence in positive and reversed shear tokamaks is studied. In the simulation, initial saturation levels of the ETG mode are consistent with the mixing length theory, which shows a Bohm (gyro-Bohm) like \( \rho^* \)-scaling for a ballooning type (slab like) ETG mode in a positive (reversed) shear configuration, where \( \rho^* \) is the electron Larmor radius \( \rho_e \) divided by the minor radius \( a \). In a realistic small \( \rho^* \) positive shear configuration, the ETG mode has a higher saturation level than the large \( \rho^* \) positive shear configuration and the reversed shear configuration. In the nonlinear turbulent state, the ETG turbulence in the

![Contour plots of the electrostatic potential observed in ETG turbulence simulations in (a) positive and (b) reversed shear configurations.]

Fig. III.3.1-1 Contour plots of the electrostatic potential observed in ETG turbulence simulations in (a) positive and (b) reversed shear configurations.
positive and reversed shear configurations shows quite
different structure formations. In the positive shear
configuration, the ETG turbulence is dominated by
streamers which have a ballooning type structure, and
the electron temperature $T_e$ profile is quickly relaxed by
an enhanced heat transport in a turbulent time scale. In
the reversed shear configuration, zonal flows are
produced in the negative shear region, while the
positive shear region is characterised by streamers.
Accordingly, the electron thermal diffusivity $\chi_e$ has a
gap structure across the $q_{\text{min}}$ surface, and the $T_e$
gradient is sustained above the critical value reversed shear
tokamaks.

References
3.1-1 Idomura, Y., Tokuda, S., and Kishimoto, Y., Proc. 20th

3.2 Confinement Improvement by Control of
Zonal Flow Behavior
Based on global Landau-fluid ion temperature gradient
(ITG) driven turbulence simulations, it is found that
plasma confinement can be improved by control of
zonal flow behavior[3.2-1,3.2-2]. Zonal flows are
almost stationary in a low safety factor (q) region and
suppress turbulent transport effectively. On the other
hand, the zonal flows are oscillatory in a high q region.
The oscillatory zonal flows cannot suppress the ITG
turbulence effectively. Thus the stationary zonal flows
in the low q region are favorable for plasma
confinement.

![Fig. III.3.2-1 Radial profile of time averaged heat flux. The heat flux in 0.4<\rho<0.7 for the q profile shown in Fig. 2(a) (solid line) is reduced compared to that for the q profile shown in Fig. 2(b) (dashed line) due to expansion of the stationary zonal flow region.](image)

Therefore it is expected that when q profile has a broad
low q region, the turbulent transport is suppressed in a
broad region by the stationary zonal flows. Figure 1
shows radial profile of time averaged turbulent heat flux
for the q profiles shown in Fig. 2. In the case with the q
profile having a wider low q region shown in Fig.
III.3.2-2(a), the heat flux is reduced in a broad region
compared to that for the q profile shown in Fig. III.3.2-
2(b). This result indicates that the turbulent transport
can be controlled through the control of the zonal flow
behavior by the q profile.

![Fig. III.3.2-2 The q profiles used in the calculations: (a) \( q = 1.05 + 2(r/a)^{1.5} \) and (b) \( q = 1.05 + 2(r/a)^{3} \)](image)

References
3.2-1 Miyato, N., Kishimoto, Y. and Li, J., Physics of
Plasmas 11, 5557 (2004).
3.2-2 Miyato, N., Li, J. and Kishimoto, Y., Proc. 20th IAEA
file TH/8-5Rb.

3.3 Formation of Current Hole by Pair Vortex
Motion
Some negative currents should be driven in the central
region of the tokamak by bootstrap current and off-axis
current drive when the amplitude of driven current is
large enough. Once a surface with a zero poloidal
magnetic field appears, however, a toroidal equilibrium
is lost and any static state cannot exist. Plasma motion
along the horizontal direction occurs by the force
unbalance between the inside and outside of the torus. A
pair of vortices with counter rotation grows in this case.
Once the vortex rotation grows enough, plasma flow
across a poloidal magnetic field produces an effective
electric field, which almost cancel out a negative one-
turn voltage. the plasma current profile is kept flat by
this convective motion. We investigate the growth of
this convective motion and find the appearance of the
flat current profile, the formation of a current hole, by
resistive MHD simulations[3.3-1]. After the current hole
is formed, additional current drive to the central becomes difficult by the plasma flow. This process is considered as an inverse process of a dynamo effect observed in RFP plasma and geophysics.

References
IV. TECHNOLOGY DEVELOPMENT

1. Superconducting Magnet

Since superconducting magnet is indispensable in a tokamak fusion reactor to confine plasma without large energy consumption, JAERI has been developing superconducting magnet technology for fusion reactors. In the ITER Engineering Design Activity (EDA) started in 1992, a Central Solenoid (CS) model coil, a Toroidal Field (TF) model coil and three inserts (two used Nb$_3$Sn, one used Nb$_3$Al) were developed and tested, and all of their development goals have been achieved. The test results of the inserts using the Nb$_3$Sn conductors indicated that there was an unexpected degradation in the critical current of the conductors when a large current, such as 46 kA, was put in the conductors under high magnetic fields, such as 13 T. JAERI has then started an experimental and analytic investigation on this phenomenon.

In 2004, JAERI developed a new simulation model to predict degradation in critical current performance in a large current superconductor for the design optimization of the ITER conductor. In parallel, JAERI has performed the demonstration of mass production technique of Nb$_3$Sn strand whose critical current was enhanced to meet the optimized conductor design. In addition, a numerical code simulating the critical current performance of the CS model coil was developed to confirm that similar degradation also occurred in the CS model coil. Furthermore, JAERI is developing Bi-2212 strand to realize a high field (about 20 T) magnet using High Temperature Superconductor (HTS) strand for a fusion DEMO plant following the ITER. The following sections describe major achievements in these research activities.

1.1 Degradation of Critical Current Performance

JAERI has experimentally revealed that the degradation of the critical current of the Nb$_3$Sn conductors was caused by periodic bending deformation of the strands due to large electromagnetic force, as shown in Fig. IV.1.1-1 [1.1-1].

A new simulation model was then developed to quantitatively clarify the mechanism of degradation in the critical current. In our new model, 1) transverse load due to the electromagnetic force is calculated on each strand; 2) the periodical bending strain of individual strand is estimated; 3) the critical current of strand is calculated based on the experimental results of the critical current dependence on the periodical bending strain [1.1-2]; and 4) conductor critical current is evaluated by integrating electrical field on the conductor cross section. The degradation in current sharing temperature ($T_{cs}$; temperature at which normal transition takes place) of the inserts using the Nb$_3$Sn conductor can be explained well by this model, as can be seen in Fig. IV.1.1-2. The same calculation was performed for the Nb$_3$Al insert, in whose experiment no degradation in the critical current performance was observed [1.1-3]. The test results of the Nb$_3$Al insert can also be simulated well. These results indicate that the quantitative evaluation in the critical current

![Fig. IV.1.1-1 Periodic and local deformation of a strand in the cable due to transverse electromagnetic force.](image1.png)

![Fig. IV.1.1-2 Measured and calculated current sharing temperature degradation of the inserts using Nb$_3$Sn conductor (CSI, CS Conductor Insert, TFI, TF Conductor Insert) and Nb$_3$Al Conductor Insert (ALI). Exp. and Cal. in the figure indicate the experiment and calculation results.](image2.png)
performance becomes possible by using this method and the design of the ITER conductor, which was optimized based on the model coil test results (see 1.2), is appropriate.

References

1.2 Trial Production of ITER-TF Nb$_3$Sn Strands
The enhancement of critical current density ($J_c$) of Nb$_3$Sn strand for ITER TF coils was decided taking account of degradation in the critical current of the Nb$_3$Sn conductors, as described above. The required $J_c$ at 12 T and 4.2 K is more than 700 A/mm$^2$ for bronze processed Nb$_3$Sn strand and more than 800 A/mm$^2$ for internal tin processed one.

In 2003, small trial fabrication by bronze process was performed using bronze with high tin content of 15 to 16%, and strands having high $J_c$ of more than 700 A/mm$^2$ were made available. Recent analysis performed by the ITER International Team moderated the upper limit of hysteresis loss from 400 mJ/cm$^3$ to 1000 mJ/cm$^3$. This allows us to use Nb barrier, which increases hysteresis loss but improve productivity, instead of Ta barrier. In 2004, trial production was performed to demonstrate mass production technique of bronze processed Nb$_3$Sn strands in which the Nb barrier is used as shown in Fig. IV.1.2-1.

In addition, trial mass production of internal tin processed Nb$_3$Sn strands was also performed. For the internal tin process strand, size of modules, each of which consists of a tin rod and a lot of Nb filaments embedded in a copper cylinder, was reduced, while total amount of tin increases in order to satisfy both required high $J_c$ and low hysteresis loss, as shown in Fig. IV.1.2-2.

As a result, Nb$_3$Sn strands of 60-km and 22-km total length were fabricated by the bronze process and the internal tin process, respectively. Typical $J_c$ at 12 T and 4.2 K are 750 A/mm$^2$ and 1000 A/mm$^2$ for the bronze process and internal tin process strands, respectively. These performances satisfy the ITER requirements.

1.3 Numerical Simulation on the CS Model Coil Performance
The most important goal of the CS model coil was the demonstration of the rated operation up to 46 kA at 13 T. Therefore, no instrumentation was installed inside the winding to avoid the possible break of electrical insulation. This made it difficult to evaluate the conductor performance only from the measurements. Thus, a three-dimensional numerical simulation code was developed and supercomputers were used for the simulation.

Calculations were performed using the newly-developed code to simulate 8 experimental runs in which the current sharing temperatures of the CS model coil were measured. In the experiments, coolant was heated by the heater attached on pipes at the inlet of the coil until a certain level of resistive voltage was observed. In the simulation, thermo-hydraulic equation for coolant and equation of heat conduction for the coil were solved with the boundary conditions of the measured inlet temperature and pressure and the outlet.
Fig. IV.1.3-1 Comparison of the calculated and measured voltages of the current sharing temperature measurement of the CS model coil at 46 kA and 13 T.

Fig. IV.1.3-1 shows comparison between the measured and calculated resistive voltages of the innermost layer, whose length is 83.8 m, at the rated condition of 46 kA and 13 T. Since the agreement between the experiment and simulation is fairly good, we can conclude that the simulation using the developed code enables us to evaluate the critical current performance of the CS model coil.

The analysis results for all of the current sharing measurements support the conclusion from the insert tests; namely, the stronger the electromagnetic force on the superconductor, the more the critical current performance declines.

1.4 Development of a Low Silver Ratio HTS Strand

Bi$_2$Sr$_2$CaCu$_2$O$_8$ (Bi-2212) strand is fabricated in a powder-in-tube process, in which oxygen can be supplied to the Bi-2212 powder embedded in silver alloy tube during heat treatment without oxidation of the tube. Consequently, silver is indispensable in the Bi-2212 strand. However, a ratio of area of the silver to the superconductor, defined as silver ratio hereafter, should be reduced as much as possible in fusion application since silver is expensive and easily activated to produce a long half-life element.

Technique to decrease a silver ratio has therefore been studied. In a general fabrication process of Bi-2212 strand, silver tubes filled with the Bi-2212 powder are made and drawn into a certain diameter, and then they are put in a silver alloy sheath and drawn into a final diameter, resulting in a large silver ratio. In the trial fabrication performed in 2004, one-pass drawing was attempted instead of the general two-pass drawing and a strand having low silver ratio of 1.3 has been successfully manufactured. Fig. IV.1.4-1 shows the cross-sectional views of the conventional and developed strands. Detailed Jc characteristics of the strand will be measured and conceptual design of a conductor using Bi-2212 strand will be proposed.

2 Neutral Beam Injection Heating

2.1 1 MeV Class and High Current Density Beam Acceleration [2.1-1, 2.1-2, 2.1-3]

One of the key components of the ITER neutral beam (NB) system is a high energy and high current beam source, which is designed to produce 1 MeV, 40 A D$^+$ ion beam for injection of 16.5 MW neutral beams per module. However, any charged particles with ampere class current have never been accelerated up to the beam energy of MeV range. Moreover, the accelerator is insulated with vacuum instead of a conventional insulation gas such as SF$_6$ due to radiation-induced conductivity (RIC). At present, the R&D target is to demonstrate 1 A class H$^+$ ion beam acceleration at the current density of 200 A/m$^2$ up to 1 MeV in a "Proof-of-Principle" (PoP) beam source of the ITER NB system.

A cross sectional view of the PoP beam source is shown in Fig. IV.2.1-1. An insulator column, which is a stack of five FRP (fiber reinforced plastic) insulator rings, forms a vacuum boundary against SF$_6$ insulation gas. The accelerator main structure, i.e., five acceleration
grids at intermediate potentials (every 200 kV) are supported in vacuum by post insulators made of ceramics. Thus the accelerator main structure is immersed in vacuum to simulate the accelerator of the ITER NB system. Having the ion source directly mounted on the accelerator, gas fed in the source is pumped down through the accelerator, and the pressure in the accelerator distributes in the range of 0.05 ~ 0.2 Pa during the operation. Vacuum insulation technologies for the 1 MV high voltage have been established as follows: 1) Against vacuum arc discharge, extrapolation of conventional “clump” theory was found valid even up to the high voltage of 1 MV. 2) However, the pressure range during operation is high enough to strike glow discharge. Local pressure distribution and the accelerator geometry, i.e., vacuum insulation distance, were examined against Paschen law to prevent the glow discharge. 3) Reduction of electric field concentration at triple junction (interface of the FRP, metal flange and vacuum) of negative side was found effective to prevent flashover along FRP insulator surface. By the vacuum insulation technologies established above, stable voltage holding of 1 MV was achieved for 8,500 s without breakdown.

In the present PoP beam source, the KAMABOKO negative ion source, which has already achieved H⁻ ion production of 300 A/m² (45 keV), was installed to aim at generation of 1 MeV level and high current density beam. The number of apertures for the negative ion extraction was limited to 9 apertures (each 14 mm in diameter) drilled in a lattice pattern of 3 x 3 because of the power supply limitation (1 MV, 1 A). To enhance the negative ion density in KAMABOKO source, gas pressure, strength of magnetic filter, plasma grid temperature and filament number and length, and Cesium (Cs) seeding conditions were optimized under a high input power operation (40 kW). As a result, H⁻ ion beams of 800 keV, 140 mA were generated with the current density of 100 A/m². This is the first demonstration of MeV class negative ion beams acceleration at the current density of practical level. In such a high energy beam acceleration at high current density, it has been anticipated that breakdowns could be triggered, 1) by Cs leakage from KAMABOKO source to the accelerator, and 2) by insulator charged-up by bremsstrahlung followed by photoelectron effect. Fig. IV.2.1-2 shows trend records of the beam acceleration voltage and power supply drain current (H⁻ plus electron current) taken from the beginning of the experiment campaign. As the Cs is seeded at low voltage (450 kV), the current increased by enhancement of the negative ion surface production. Then increasing the voltage, breakdown occurred several times in the accelerator. However, the voltage holding capability recovered within tens of shots. Consequently, repetitive shots of H⁻ ion beams at 900 keV, 100 mA level were demonstrated for 175 shots (each pulse length: 1 s, every 60 s). Thus, 1 MeV accelerator R&D is in progress toward the ITER design current density of 200 A/m².

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Fig. IV.2.1-1 PoP beam source and ITER NB beam source.
source, the $H$ ion beam current increased typically by a factor of four. By seeding Cs, the plasma parameters were not varied at all. The beam profile was not uniform, and more negative ions were observed from upper part of the extraction area. Note that gradient of the non-uniform beam profile was reversed from that without Cs [2.2-3]. Namely the ions were extracted more from lower part in pure volume operation. From the correlation between the beam intensity and the plasma parameters, it was found that the beam intensity was higher from where the plasma density was higher in Cs seeded operation. From an analysis of primary electron trajectory, it was estimated that the energetic electrons, and hence, the plasma is localized by $B \times V_B$ drift in the source. Here, $B$ is the magnetic field around the filaments whose strength is sum of the magnetic filter field and the cusp field surrounding the ion source for plasma confinement.

The $B \times V_B$ drift of the fast electrons from filaments was suppressed by modifying the filament configuration. Figure IV.2.2-1 shows the beam profile before and after the filament modification. By the filament modification, the beam uniformity was drastically improved. The root-mean-square deviation of the beam intensity from the averaged value decreased to a half of that before the modification while the beam intensity integrated along the longitudinal direction was kept to be constant. This

![Fig. IV.2.2-1 Longitudinal distributions of the beam intensity after(circle) and before(square) filament modification.](image)

2.2 Negative Ion Uniformity under Cs Seeded Condition [2.2-1]

In the existing negative ion based NBIs for JT-60U and LHD, high current negative ion beams of $>20\,\mu$A have been already produced through a large extraction area (45 cm x 110 cm in JT-60U N-NBI). However, the beam pulse lengths at the high current operations are limited due to high power loading on acceleration grids and on beamline components [2.2-2]. One of possible reasons for the high power loading is non-uniformity of the negative ion density in the wide extraction area. Improvement of the negative ion uniformity is a critical issue for expansion of the pulse lengths in large negative ion sources. The objective of the present experiment is to identify the physics mechanism that degrades the negative ion uniformity.

The JAERI 10 ampere negative ion source has been served to the experiment, which has rectangular discharge chamber with similar magnetic configuration to that of the JT-60 large negative ion source (semi-cylindrical discharge chamber). Local plasma parameters, such as electron temperature or plasma density, were measured in the longitudinal direction in the extraction region, and correlated to the beam profile. The measurements were obtained both in pure volume and Cs seeded conditions, in which surface negative ion production is enhanced.

After seeding sufficient amount of Cs (~0.3 g) in the

![Fig. IV.2.1-2 900 keV, 100 mA class beam acceleration.](image)
result indicates that the suppression of the $B \times \mathbf{v}$ drift of the fast electrons is effective to improve the beam uniformity in the Cs-seeded negative ion source.

References

2.2-1 Hanada, M. et al., submitted to 4th IAEA Technical Meeting on Negative Ion (2005).

2.3 Conceptual Design of a NB System for Fusion DEMO Plant [2.3-1]
A design study of the NB system for fusion DEMO plant has been started at JAERI on a basis of the ITER technologies and their reasonable extrapolations. The NB system for the DEMO should be high efficiency (> 50 %), high energy (~ 2 MeV), high reliability in yearlong steady operation, and low maintenance frequency even under higher radioactive environment. Technical R&D issues for key components are extracted and identified for the NB system of the DEMO as follows.

2.3.1 Accelerator
From the viewpoint of high acceleration efficiency, use of conventional electrostatic accelerator is realistic. Due to operation under radiation environment, reduction of beam cross sectional area by beam merging technique would be necessary to install additional radiation shields in the beamline. Moreover, high voltage insulation by vacuum is essential in the accelerator. Database on vacuum insulation shows feasibility of conventional electrostatic accelerator up to the energy of ~ 2 MeV. It is essential to validate vacuum insulation technology up to the high voltage level, and furthermore, manufacturing technology of insulator rings of the diameter of 2 ~ 3 m should be developed.

2.3.2 Ion Source
For a yearlong operation without maintenance, production of negative ions in filament-less plasma is required, for example, by rf or ECR. Another key issue is high current density negative ion production by surface process, but without Cs, utilizing low work function materials such as LaB6 or alkaline metal implanted tungsten.

2.3.3 Neutralizer
Since the NB system efficiency is limited by neutralization efficiency, conventional gas neutralization (efficiency ~60%) cannot achieve the system efficiency higher than 40%. Moreover, for the long pulse cw operation of a yearlong, less or no gas consumption is desirable in fusion DEMO plant. The plasma neutralizer has been studied at JAERI to achieve a high neutralization efficiency of ~80%. Recently, a new concept of laser neutralizer, of which neutralization efficiency is higher than 90 %, was proposed utilizing array of semiconductor lasers as shown in Fig. IV.2.3-1, that run cw at the laser emission efficiency of 40 % with the power of 2.7 MW/array. The key to realize the laser neutralizer is in the optics development to minimize photon leakage from beam entrance/exit openings.

Fig. IV.2.3-1 A concept of laser neutralizer utilizing semiconductor laser array.

References

2.3-1 Inoue, T. et al., “Design study of a neutral beam injector for fusion DEMO plant at JAERI”, to be published in Fusion Eng. Design.
3. Radio Frequency Heating

3.1 Modification of 170GHz Gyrotron for Long Pulse Operation

ITER requires a 170 GHz high-power gyrotron system with a total power of 24 MW, for electron cyclotron heating (ECH), current drive (ECCD) and suppression of plasma instabilities. Intensive development of a 170 GHz gyrotron (1 MW, CW (continuous wave) operation and 50% efficiency) is under way for ITER. A gyrotron is a microwave tube that utilizes electron-cyclotron resonance maser effect and weakly relativistic electron beam (<100kV) with gyro-motion. Rotational electron beam is generated at a magnetron injection gun (MIG) and 170GHz millimeter wave with $\text{TE}_{31,8}$ mode oscillates in a cylindrical cavity with the magnetic field of ~6.7T. The power is delivered as the Gaussian beam through the artificial diamond window. Up to 2003, the gyrotron has demonstrated quasi-steady state operation of 100 s with the output power of 0.5 MW by reduction of heat generation due to stray RF in the gyrotron and optimization of velocity spread of electron beam at MIG [3.1-1].

In 2004, it was clarified that the cause of the limitation of the output power and the pulse length in the gyrotron are (1) diffraction loss of the internal mode converter and (2) the decrease of the beam current during the pulse duration due to emission cooling. For a further pulse extension and power increase, the gyrotron and its control system have been modified; i.e. a radiator in the built-in mode converter has been optimized for improvement of an efficiency of gyrotron output power and reduction of stray radiation, and pre-program controls of a cathode heater power has been employed for stabilization of the beam current and the output power [3.1-2].

3.1.1 Improvement of the Radiator in the Built-in Mode Converter

A built-in mode converter is widely used in high power gyrotrons, because the high order oscillation mode in the cavity must be changed to Gaussian beam, which is appropriate for long transmission through a waveguide. The inner wall of the radiator of a mode converter has a complex perturbation to convert the oscillation mode to focused beam. Previous mode converter had diffraction loss about 7% and parasitic oscillations. To reduce the diffraction loss, inner surface of the radiator was optimized for the next gyrotron. The deformation of the radiator, field intensity on the inner wall and the field intensity on a screen placed 60mm from the radiator axis are shown in Fig. IV.3.1-1 (a), (b) and (c), respectively. The designed radiator has a taper of 0.2 degrees and the output radius is 21.5mm. It is expected that 99.5% of the radiated power from radiator enter on the first mirror in the calculation. As a low power measurement using a mock-up of the designed radiator, measured RF power profile was in good agreement with designed profile and no side-lobe power was observed. Consequently, we have a prospect to remarkable reduction of the diffraction loss in the gyrotron.

3.1.2 Pre-programming Control of Beam Current

The beam current decreased from 35A to 25A as the pulse duration expanded during 0.5MW/100s operation. Consequently, the power decreased from ~0.6 MW (at short pulse) to ~0.46 MW (average power during 100 s) with the current decrease. It also causes the oscillation instability due to change of the oscillation mode from $\text{TE}_{31,8}$ to $\text{TE}_{30,8}$, so called mode jump. The current decrease is mainly caused by cathode cooling due to electron emission. A heater boost of the electron gun

Fig. IV.3.1-1  Design of the improved radiator for the next gyrotron, (a) deformation of the inner wall, (b) field intensity on the radiator wall, (c) field intensity on the screen placed 60 mm from the radiator axis, 150 mm x 150 mm.
with pulse duration was proposed to compensate the current decrease.

At a first step, electron beam emission test without oscillation was carried out. Fig. IV. 3.1-2 shows beam current behavior (initial current ~27 A) with/without pre-programming control of cathode power. As a result, beam current decrease was not observed with the pre-programming control and it was succeeded in 1000 s pulse, which is required in ITER.

Therefore, a good prospect was obtained to stabilize the oscillation and to increase the output power in 170GHz ITER gyrotron with CW operation by application of the above modifications.

![Graph](image1)

Fig. IV.3.1-2 Controlled beam current during 1000 s operation without oscillation.

References


3.2 Development of RF Launcher for mm Wave Steering

3.2.1 Development of Reliable RF Window

Since an RF torus window takes a role of vacuum and tritium barrier between an EC launcher (a reactor vessel) and a transmission line in a fusion reactor like International Thermonuclear Experimental Reactor (ITER) or DEMO, the window structure must have high reliability. In 1998, a synthetic diamond disk was applied for the RF window [3.2-1]. The disk edge was directly cooled by water and transmission capability of 1MW mm-wave beam was confirmed. However, it had been concerned about the ingress of cooling water into vacuum if a crack was produced at the window disk and grown toward the disk edge. In order to prevent the possible event, a new diamond window structure with Cu-coated edge has been developed [3.2-2]. The thickness of the coating is 0.5mm. The aluminum blaze between the diamond disk and the inconel cuff is also covered by the Cu-coating layer. Therefore, the coating layer is effective to avoid the aluminum corrosion as well. The photograph of the diamond window with Cu-coating layer is shown in Fig. IV.3.2-1. The high power RF experiment of the window was made to investigate the capability to remove the RF power deposition. Temperature increase of the window center at indicative power of 1.2MW and pulse length of 3.5sec is shown in Fig. IV.3.2-2. The circles are the experiment data, which agree with the calculation indicated by the solid line. It was recognized that the temperature increase was almost comparable to the one for the conventional structure of the diamond window. Thermo-mechanical

![Diagram](image2)

Fig. IV.3.2-1 Photograph of Cu-coated diamond window.

![Diagram](image3)

Fig. IV.3.2-2 Temperature increase of the window center. The circles and the solid line are the experimental and the calculation results, respectively.
analysis using ABAQUS code was also carried out. It was verified that the maximum induced stress in the diamond disk was 70MPa at the disk edge, which was less than the median fracture stress of synthetic diamonds (300MPa). It was concluded that the Cu-coating improved the reliability of the torus diamond window.

3.2.2 Development of a Flexible Cooling Tube for a Steering Mirror
In the development of a steering mirror, which controls incident angle of rf beams to plasma, the cyclic fatigue property of a flexible cooling tube, which provides cooling water into the steering mirror, was studied. A spiral tube has been introduced and the mock-up of the tube was fabricated for the cyclic loading test, where the mirror rotation of ± 6.5° was considered. The photograph of the spiral tube mock-up is shown in Fig. IV.3.2-3 [3.2-3]. The rotation cycle of 1.3x10⁶ was successful without failure. This cycle number exceeds the expected value from ASME data. It was concluded that the spiral structure was effective to the flexible tube for the steering mirror.

![Rotational shaft and Spiral tube](image)

Fig. IV.3.2-3 The mock-up of spiral tube.

References

3.3 Application of High Power RF
Studies of a microwave beaming propulsion were proceeded as application of high power RF, which was generated by a gyrotron, under the collaboration between JAERI and University of Tokyo. The concept of the beaming propulsion is to produce breakdown at a parabolic thruster corresponding to a nozzle of a vehicle. When electromagnetic waves are injected into the thruster, an instantaneous breakdown occurs near the focal point and plasma is formed. The plasma absorbs the following beam energy and expands outward generating shock waves. The shock waves reflect on the inner surface of the thruster, generating impulsive force. Since the input energy to the vehicle is supposed to provide from the ground and atmospheric air is utilized as a propellant, no fuel needs to be loaded onto the vehicle, in principle. Consequently, a remarkably low launching cost is expected.

Experiment of multi pulses injection into the thruster model were carried out in 2004. As shown in Fig. IV.3.3-1, it was verified that the flight altitude of the thruster was improved when the multi pulse injection with repetition rate of 100Hz was applied [3.3-1].

![Graphs showing microwave power and altitude](image)

Fig. IV.3.3-1 Multi pulse injection into the thruster model, (a) microwave power and (b) altitude by microwave propulsion.

Reference
4. Blanket

4.1 Fabrication of Cooling Channels of First Wall and Cooling Panels

As a blanket fabrication process, hot isostatic pressing (HIP) bonding has the great merit for near-net-shaping processing. The degassing conditions and surface roughness were investigated as parameters of HIP bonding conditions. The material used in this work was from the 5000kg of F82H produced in 1997; the details of this steel are presented elsewhere [4.1-1]. The processing operation for the HIP joint is presented in Fig. IV.4.1-1. The HIP conditions applied in this work are summarized in Table IV.4.1-1.

The bonding surfaces were grounded by mechanical milling followed by degreasing by acetone. The surface roughness measurement was performed by laser microscopy. The mean surface roughness, Ra was measured to be 4-40μm; this was equivalent to that of as-rolled plate and pipe which is significantly larger than that normally recommended for HIPping [4.1-2, 4.1-3]. Conventional HIP bonding was performed at 1373 K / 150 MPa for 2 h. In this condition, the prior austenite grain size was coarsened during the bonding [4.1-4] and this grain coarsening has a significant influence on impact properties. However, to complete a satisfactory bond, it is necessary to HIP at this high temperature. Therefore, the HIP was done at 1373 K followed by normalizing at 1233 K. These heat treatment conditions were determined by post heat treatment metallurgical investigation [4.1-4].

Fig. IV.4.1-2 shows optical micrographs of the HIP interface from E/F (Ra: 40μm). The location of the interface and the presence of pores were difficult to detect even in the joint E/F. The tensile properties of the HIP joints are summarized in Table IV.4.1-2. Better properties were obtained with smooth surfaces, however, the strength of the joint were larger than 80 % of that of Base Metal (BM) and these properties were better than the baseline properties of F82H. Examination by Scanning Electron Microscopy (SEM) revealed that all specimens were broken adjacent to the HIP interfaces, and the no brittle fracture was observed in the fracture surface. These results imply that the effects of HIP joining had only a slight impact on the tensile properties. On the other hand, the impact properties were significantly affected by HIPping. The results of Charpy impact testing of the HIP joints are presented in Fig. IV.4.1-3. The Ductile Brittle Transition Temperature (DBTT) of the HIP joint was above 273 K. Moreover, the Upper Shelf Energy (USE) of joint was lower than 40 % of BM level. The high DBTT and loss of USE was
Table IV.4.1-2 The Tensile properties of F82H HIP joint

<table>
<thead>
<tr>
<th></th>
<th>Base metal after HIP heat treated</th>
<th>HIP joint A/B</th>
<th>HIP joint E/F</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.2% proof stress (MPa)</td>
<td>571</td>
<td>528</td>
<td>497</td>
</tr>
<tr>
<td>Ultimate tensile stress (MPa)</td>
<td>644</td>
<td>640</td>
<td>617</td>
</tr>
<tr>
<td>Uniform elongation (%)</td>
<td>5.2</td>
<td>4.7</td>
<td>4.3</td>
</tr>
<tr>
<td>Total elongation (%)</td>
<td>25.0</td>
<td>21.9</td>
<td>21.8</td>
</tr>
<tr>
<td>Reduction of area (%)</td>
<td>78.0</td>
<td>76.1</td>
<td>75.4</td>
</tr>
</tbody>
</table>

The micrographs of the fracture surfaces are presented in Fig. IV.4.1-4. Traces of surface milling were observed in the HIP specimen, and the lateral expansion of the HIP joint was quite smaller than that of the BM. Higher magnification observation revealed that the fracture surface of the HIP specimen was covered with fine dimples. Elemental analysis by EDX revealed that the base of the dimples contained iron-oxide and chromium-oxide particles. It is considered that the oxide film on the bonding surface was divided into micro inclusions. These micro inclusions assist the crack propagation and the absorbed energy was significantly decreased. It is concluded that efficient degassing and the elimination of oxide formation on the HIP joint surfaces are necessary to obtain a satisfactory joint. To establish the quantitative criteria for the HIP pretreatment, it is important to evaluate the gas desorption spectrometry up to the HIP temperature.

Although the effects of surface finishing on impact properties seemed to be not significant, it could depend on the degassing conditions. Therefore the rough surface with sufficient degassing should be investigated in the near future.

References

4.2 Performance of Cooling Channels in Supercritical Water

In the blanket design of a fusion DEMO plant, it is planned that pressurized water of high temperature is used as a coolant. In the first ITER test blanket module, the coolant conditions are expected to be equivalent with pressurized water reactor conditions, however, in the advanced module super critical pressurized water (SCPW) would be used for higher energy efficiency. In order to establish a reliable blanket system, the compatibility of F82H with SCPW should be examined.

An autoclave with mechanical test machine system has been newly developed and successfully installed. The overview of the system is presented in Fig. IV.4.2-1. In this system, 23MPa with 873K Super Critical
Pressurized water is achieved as the test environment. The dissolved oxygen concentration and the electricity conductivity are monitored and controlled.

A structural material of blanket, F82H was examined by Slow Strain Rate Tensile (SSRT) test to evaluate the susceptibility for stress corrosion cracking (SCC) in SCPW. Simultaneously the weight measurement was performed to evaluate the corrosion rate in the SCPW. The test temperature and the dissolved oxygen contents were investigated as parameter of SCPW conditions. The specimen used for SSRT was round bar specimen with 4mm diameter and 20mm gauge length. The coupon specimen was also tested in the SCPW for weight measurement.

A preliminary test showed that all specimens demonstrated the ductile fracture and no SCC were observed. The result of weight measurement is presented in Fig. IV.4.2-2. In all cases, obtained curves could be fitted by a parabolic formula. These preliminary results showed the corrosion in SCPW was not different from that in boiled water and pressurized water, the difference was just the magnitude.

References

4.3 Thermo-mechanical Performance of Breeder Pebble Bed
In order to establish a reliable design of the blanket system, it is necessary to analyze heat transfer in the pebble bed. During operations, the pebble bed is deformed because of temperature distribution and different thermal expansions between the beds and structural materials. The deformation will result in the deviation of effective thermal conductivity of the beds. Therefore, it is important to study combined phenomena of thermal and mechanical properties of the bed. However, these properties were investigated independently in the previous works. With regard to simultaneous measurement of thermal conductivity and mechanical conditions of the pebble bed at different temperatures, there are limited numbers of reports. In this year, effective thermal conductivity and stress-strain properties of Li$_2$TiO$_3$ pebble beds were measured simultaneously using the apparatus that was developed in the last year.

The configuration of the apparatus is shown in Fig. IV.4.3-1. Pebbles of Li$_2$TiO$_3$ with about 2mm diameter were packed into a container made of alumina. The initial packing fractions of the beds were 65 to 67%. The container was inserted into a quartz tube, which was located on a tensile test-apparatus, INSTRON. Temperature of the pebble beds was regulated from 400 to 700°C by an infrared furnace. Under compression up to 10MPa, the effective thermal conductivity of the bed was measured by the hot wire method.

As shown in Fig. IV.4.3-2, increases of effective thermal conductivity due to the compressive deformation were confirmed at temperatures ranging from 400 to 700°C. During several cycles of compression and measurement, repeatability of obtained data was good at lower temperatures than 500°C. Above 600°C, it was found that compressive strain at the same stresses increased when the bed was compressed in.
In order to elucidate the effect of the annealing of the pebbles, the pebble bed was heated at 700°C for 1 day without loads after installation into the quartz tube. When temperature was lowered and regulated at 400°C, effective thermal conductivity was measured. A sequence of the measurements was as follows and the sequence was successively conducted in three times. At each temperature up to 700°C, the bed was compressed and thermal conductivity was measured. After the measurement at 700°C, the bed was cooled down to 400°C and the same procedure was executed again. The results of three series of the experiments are shown in Fig. IV.4.3-3. The comparison of the results in Figs. IV.4.3-2 and IV. 4.3-3 shows that the measured effective thermal conductivities were higher in the annealed bed than the bed without pre-annealing. Differences in thermal conductivity were large at lower temperatures. It was found that annealing of the pebbles leaded to increase in effective thermal conductivity of the bed. Fig. IV.4.3-3 shows that both thermal conductivity and compressive strain increase when the thermal and/or mechanical loads successively worked on the bed. The present study shows that history of the thermal and mechanical loads on the bed affect the thermo-mechanical properties of the bed. It is important to consider the history of the loads on the pebble bed when the pebbles are packed into a blanket container and one predicts behavior of the pebbles after the long operation period.

Fig. IV. 4.3-2 Effective thermal conductivity of a compressed Li₃TiO₃ pebble bed.

Fig. IV. 4.3-3 Effective thermal conductivity of annealed Li₃TiO₃ pebble bed during three series of experiments.
In the previous calculations for the thermo-mechanical design of the ceramic pebble beds, the effect of compressive deformation on the effective thermal conductivity has been neglected. From the data obtained in a series of the works, it will be possible to analyze combined phenomena of thermal and mechanical properties with the assumption that the pebble bed was treated as a continuum model. In the next stage, it is important to confirm and refine the current design by the new calculation based on the present study.

References
4.3-1 Tanigawa, H., et al., “Effective thermal conductivity of a compressed Li$_2$Ti$_3$O$_7$ pebble bed,” to be published in Fusion Eng. Des.

4.4 Design of Test Blanket Module
Test Blanket Module (TBM) test in ITER is one of the most important milestones in the development of the breeding blanket in Japan [4.4-1]. This year, designs of Water Cooled Solid Breeder (WCSB) TBM and Helium Cooled Solid Breeder (HCSB) TBM were performed, for the purpose of the coordinated test plan and test program development among the ITER participant parties and International Team (IT) under the frame of Test Blanket Working Group (TBWG) [4.4-2]. Table IV.4.4-1 summarizes the design conditions and major specifications of TBM adopted in the design study. The most important point of developed structure is slit structure to reduce electro-magnetic force in disruption event and to enhance the durability against internal over-pressure in the case of coolant ingress in the module. The available space for TBM is limited by design constraint of horizontal test port of ITER. Fig. IV.4.4-1 shows the structure of typical cross sections of Water Cooled Ceramic Breeder TBM. The major dimension of WCSB TBM is about 680 width x 1940 heigh x 600 thick. In the case of WCSB TBM, two sub-modules have same box structures and internal structures. The first wall made of F82H has built-in rectangular cooling paths. As for internal structure, it has multi-layer pebble beds structure same as the DEMO blanket. Breeder and neutron multiplier formed by small pebbles are packed separately in inner box structure made of F82H thin plates, which is separated into four layers by cooling panels. The

<table>
<thead>
<tr>
<th>Structural Material</th>
<th>Water Cooled</th>
<th>He Cooled</th>
</tr>
</thead>
<tbody>
<tr>
<td>Neutron Multiplier</td>
<td>Be, or Be$_{12}$Ti</td>
<td></td>
</tr>
<tr>
<td>Tritium Breeder</td>
<td>Li$_2$Ti$_3$O$_7$, or other Li ceramics</td>
<td></td>
</tr>
<tr>
<td>SHF (aver., max.)</td>
<td>0.3, 0.5MW/m$^2$</td>
<td></td>
</tr>
<tr>
<td>NWL (aver.)</td>
<td>0.78MW/m$^2$</td>
<td></td>
</tr>
<tr>
<td>FW Area</td>
<td>0.68 x 1.94 m$^2$</td>
<td></td>
</tr>
<tr>
<td></td>
<td>1.49 x 0.91 m$^2$</td>
<td></td>
</tr>
<tr>
<td>Coolant Pressure</td>
<td>15 (25) MPa</td>
<td></td>
</tr>
<tr>
<td></td>
<td>8 MPa</td>
<td></td>
</tr>
<tr>
<td>Inlet/ Bypass/Outlet</td>
<td>280/325 (360/380*)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>300/401/500 (472)</td>
<td></td>
</tr>
<tr>
<td>Heat Deposition [MW]</td>
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<tr>
<td></td>
<td>1.61</td>
<td></td>
</tr>
<tr>
<td>Tritium Production Rate</td>
<td>0.156g/FP</td>
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<td>0.18g/FP</td>
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</tr>
<tr>
<td>Coolant Flow Rate</td>
<td>5.18kg/s</td>
<td></td>
</tr>
<tr>
<td></td>
<td>1.8kg/s</td>
<td></td>
</tr>
</tbody>
</table>

cooling panel consists of F82H tubes, which are the inner diameter of 9mm and the thickness of 1.5mm, and thin plates connecting adjacent tubes. The inner box structure is welded to the first wall and the back plate. The thickness of each layer and pitches between tubes at each cooling panel were optimized to experience similar level of temperatures and possibly stresses as those in the DEMO blanket according to the transient performance analyses of temperature evolution and tritium generation / release performance. Based on the TBM design, the design of ancillary systems (cooling systems, tritium recovery system and tritium measurement system) were also performed.

Performance analyses, such as, thermo-mechanical analysis of FW, TBM box with internal pressure, and analysis of temperature distribution and tritium release performance, were performed to ensure the integrity of structure and functional performance of TBM. In the structure design, the thickness of major parts, such as structural wall, breeder layers and multiplier layers, were decided to maintain the same peak temperatures in
the DEMO blanket, 550°C for structural material, 900°C for breeder material and 600°C for multiplier material. Total tritium breeding ratio (TBR) of 1.42 and 1.47 are obtained for WCSB TBM and HCSB TBM by one dimensional calculation.

Thermo-mechanical integrity was evaluated by FEM analysis. Thermo-mechanical endurance is one of the most important test issues. Fig. IV.4.4-2 shows the temperature distribution in the first wall of WCSB TBM evaluated by two dimensional thermo-mechanical analysis. To obtain the similarity of temperature on the first wall structure, the front part thickness was determined to be 10 mm as seen in Fig. IV.4.4-2. Consequently, the highest temperature of the structural material, 539 °C, which satisfies the F82H design window, appeared at the most distant part of plasma side surface from cooling channel. Fig. IV.4.4-3 shows the stress distribution in the first wall of WCSB TBM evaluated by two-dimensional thermo-mechanical analysis. By stress analysis, it was shown that the stress range was within elastic range. The highest TRESCA stress 359 MPa appeared at the same place as the highest temperature appeared. This stress value was evaluated to satisfy 35m value for F82H. In the case of He Cooling TBM, heat transfer coefficient of coolant He flow in the first wall channel is small. Consequently, the front part thickness of the first wall structure need to be 4 mm, therefore, peak TRESCA stress is smaller value 270 MPa than Water Cooled TBM case.

Detailed two dimensional nuclear analyses were performed on Water Cooled Solid Breeder TBM and He Cooled Solid Breeder TBM for the purpose of clarification of TBR distribution, induced activity and
decay heat after TBM irradiation in ITER [4.4-3]. Also, safety analysis was performed to show the relevancy of WCSB TBM to ITER safety policy [4.4-4]. By the performance analyses, the feasibility and relevancy of designed TBMs were shown.

References
4.4-4 Department of Fusion Engineering Research and Department of Material Science, JAERI-Review 2005-012.

4.5 Development of Tritium Breeder and Neutron Multiplier Materials

4.5.1 Development of Tritium Breeder

Application of Li$_2$TiO$_3$ pebbles has been proposed in Japanese and European blanket designs of fusion reactors. Development of Li$_2$TiO$_3$ pebble fabrication by the direct wet process was performed, focusing on the dissolving process of Li$_2$TiO$_3$, the dropping of the solution into coagulant, and the drying/calcinating and sintering processes of the droplets for achieving acceptable sphericity and for decreasing cracks at the pebble surfaces [4.5-1].

Under the IEA cooperation agreement, the reaction of Li$_2$TiO$_3$ ceramics with H$_2$ was studied in a thermo-chemical environment simulating (excepting irradiation) the environment at the hottest pebble-bed zone of breeding-blanket actually designed for fusion power plants [4.5-2]. The results revealed the followings: i) the H$_2$ reaction with stoichiometric Li$_2$TiO$_3$ at 900°C is negligible; ii) this reaction is enhanced as the “initial” Li-depletion (or TiO$_2$-doping) increases in the specimens; iii) Li loss by evaporation also increases on the mean time inducing a complex synergy with the O-vacancy generation.

In order to control the grain growth at the time of high temperature use, development and characterization of Li$_2$TiO$_3$ which is added with a small amount of oxide (CaO, ZrO$_2$ or Sc$_2$O$_3$) has been performed. In the weight change measurement of each Li$_2$TiO$_3$ specimen in the range from 200 to 1000°C (see Fig. IV.4.5-1), the weight of Li$_2$TiO$_3$ slightly decreased, generating O-vacancy by reduction from about 300°C, and the weight decreased quickly with a strong reduction reaction above about 800°C [4.5-3]. Further, Ca-Li$_2$TiO$_3$ had fewer oxygen defects than the other kinds of Li$_2$TiO$_3$, so the order of the ratio of weight change above 800°C is the following:

Ca-Li$_2$TiO$_3$ < Li$_2$TiO$_3$ < Zr-Li$_2$TiO$_3$ < Sc-Li$_2$TiO$_3$.

On the other hand, in the results of vaporization measurements of Li$_2$TiO$_3$ and Li$_2$TiO$_3$ with some different oxide additives under the conditions of vacuum and D$_2$ atmospheres, the vaporization properties of each Li$_2$TiO$_3$ specimen were not influenced by the atmospheres, and the value of partial pressure for Li$_2$TiO$_3$ with and without the additive was mostly in agreement.

![Fig. IV.4.5-1 Temperature dependence of the weight change of Li$_2$TiO$_3$ with oxide additives in reduction atmosphere.](image)

In a related work, irradiation behavior of Li$_2$TiO$_3$ under a fusion reactor environment was simulated by simultaneous irradiation of Li$_2$TiO$_3$ by triple ion beams and respective single ion beams of O$^{2+}$, He$^+$ and H$^+$ [4.5-4]. Microstructural changes observed by Raman spectroscopy suggest that formation of the TiO$_2$ layer in the sample irradiated with the triple ion beams is mainly caused by the effect of the O$^{2+}$ ion irradiation. Results of Fourier transform-infrared photoacoustic spectroscopy suggest that the amount of TiO$_2$ formed is proportional to the displacements per atom (dpa), as shown in Fig. IV.4.5-2. It is also suggested that defects generated by the irradiation would become trapping sites of hydrogen near the surface. It must be noted that such the hydrogen-trapping sites would also trap tritium generated in the fusion environment and would disturb the rapid recovery of tritium from the breeding material.
4.5.2 Development of Neutron Multiplier

Beryllium alloys such as Be-Ti alloys have been expected as promising candidates for advanced neutron multipliers. Recently, Be-Ti alloys (Be-5at%Ti and Be-7at%Ti) with α Be phase have been studied for the pebble fabrication by the rotating electrode method [4.5-5]. As a part of the research, the compatibility between Be-Ti alloys (Be-5at%Ti and Be-7at%Ti) and SS316LN was investigated [4.5-6].

rate of the reaction layer for the Be-Ti alloys was smaller than that of Be, and it decreased with increasing the Ti content in the Be-Ti alloy.

Thus, it is suggested that the Be-Ti alloys would be used at temperatures up to about 650°C, from the viewpoint of compatibility between the alloys and SS316LN.

References

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4.6 Irradiation Technology Development for In-pile Functional Tests

Envisaging the irradiation tests of the Test Blanket Module (TBM) using ITER, development of irradiation technology has been progressed with the aim of performing reactor irradiation tests simulating the structure and environment of the TBM. For these purposes, research activities were made in 1) post-irradiation examinations of a radiation-resistant small motor, 2) tests of ceramic coating for reduction of tritium permeation, and 3) analyses of irradiation data of a Li$_2$TiO$_3$ pebble bed.

4.6.1 Radiation-Resistant Small Motor

Post-irradiation examinations (PIEs) of some parts of a radiation-resistant small servomotor were carried out. Bearing, magnet and fixation agent for the coil and ceramic adhesive had been irradiated in Japan Materials Testing Reactor (JMTR) at 100°C for 18 hours at maximum; after the 18 hour irradiation, the fast neutron
fluence (E > 1 MeV) was 5 x 10^{21} m^{-2}.

The PIE works revealed the followings: 1) the rotating torque of the non-organic lubricant bearing did not increase even after the irradiation for 18 hours, while the organic lubricant bearing could not be rotated after the irradiation for 2 hours due to degradation of the lubricant, leading to increase in the torque; 2) the cohesive force of the Nd-Fe system magnet lowered with the irradiation time, while that of the Sm-Co system magnet did not change (see Fig. IV.4.6.1); 3) no marked degradation with the irradiation time was observed in the fixation agent for the coil and in the ceramic adhesive.

It is clear from the results above that the improvement in the radiation resistance will be achieved by selecting the non-organic lubricant bearing and the Sm-Co system magnet.

4.6.3 Analyses of Irradiation Data of Li_2TiO_3 Pebble Bed

An irradiation test of a Li_2TiO_3 pebble bed was continued using JMTR [4.6-2]. The irradiation period was 12,500 hr, and the fast (E > 1 MeV) and thermal (<0.683 eV) neutron fluxes were 2 x 10^{20} and 2 x 10^{24} n/m^2, respectively. It was clarified that the tritium release properties were hardly changed by the irradiation so far.

Furthermore, the effective thermal diffusivity of the pebble bed was evaluated under neutron irradiation by the constant temperature raising method. It was found that the diffusivity was also hardly changed by the neutron irradiation up to the fluence.

For evaluation of tritium production by in-pile tests in JMTR, actual tritium amount produced in irradiated tritium monitors (Al-Li alloy and LiAlO_2) was measured and compared with tritium amount calculated by a Monte-Carlo code (MCNP). From the result of this comparison, it was confirmed that the calculated tritium production was in good agreement with the measured one within -1 to +25% [4.6-3]. Furthermore, it was suggested that an MCNP calculation can predict the amount of produced tritium within an error of 10%, when the reactor condition due to the change in the control rod position is considered in the calculation model.

References


5. Plasma Facing Components
5.1 Critical Heat Flux of a Screw Tube at High Water Temperature

In the R&Ds on Plasma Facing Components (PFCs), JAERI has developed a screw tube that has as high heat removal performance as a swirl tube applied as a cooling tube in the ITER divertor. In 2004, its Critical Heat Flux (CHF) testing using a hydrogen ion beam facility was carried out under higher temperature and pressure conditions of cooling water compared with those for the previous experimental campaign at 1.5 MPa and room temperature to examine its heat removal performance under the ITER-relevant conditions [5.1-1]. The test sample was the screw tube formed of pure copper with the thread of M10 and its pitch of 1.5 mm. Fig. IV.5.1-1 shows experimental values of CHF of the screw tube at the cooling tube wall compared with those of smooth tube predicted with the existing CHF correlation at the cooling conditions of 4MPa and 100 °C. It was found that the augmentation of heat removal capability by the screw fin remains effective under high pressure and temperature coolant conditions. The wall CHF of the screw tube under the coolant flow velocity of 10 m/s reached 70 MW/m², which is almost twice as high as that of a smooth tube.

5.2 Thermal Cycle Experiment of a Tungsten Armored Divertor

Advance bonding technology for armor and structural materials has been developed for the ITER divertor application. A divertor mockup with a pin-shaped tungsten armor was developed by hot-pressing method, as shown in Fig. IV.5.2-1. Thermal cycling experiment of the divertor mockup with the pin-shaped tungsten armor was carried out to find the durability of the bonding structure. The mockup withstood a heat flux of 10 MW/m², which simulates the steady state heat flux of the ITER divertor, for more than 1000 cycles.

5.3 Synergistic Effects of Heat and Particle Loads on Tungsten

Investigation of synergistic effects of heat and particle loads on tungsten, which is a candidate material of plasma facing material of DEMO, was carried out under collaboration with Kyushu Univ. In this study, electron beam irradiations were performed onto stress-relieved tungsten with different degree of processing, i.e., reduction ratios of 80% and 95%. Irradiations onto the sample were repeated up to 5 cycles and the highest temperature of the sample was 1400 °C.

Figure IV.5.3-1 shows results of grain size analyses of the samples before and after the experiments. It was observed that in the tungsten sample with higher reduction ratio, its grain size grows through recrystallisation caused by heating. That means that the maximum grain size in the sample with the reduction ratio of 95 % was enlarged from 5 μm before the experiment to 50 μm after that. To the contrary, the maximum grain size in the sample with the reduction ratio of 80 % remained less than 10 μm.

Fig. IV.5.1-1 CHF defined at cooling wall of screw tube compared with those of smooth tube.

Fig. IV.5.2-1 Divertor mockup with a tungsten pin armor.

References
This coarsening of grain after recrystallisation is ascribed to larger strain energy stored in tungsten with higher reduction ratio that is relieved at heating. This phenomenon should be taken account of in applying tungsten as plasma-facing material.

6. Structural Material

6.1 Development of Structural Materials for Blanket

The neutron irradiation effects of F82H, Reduced Activation Ferritic/Martensitic (RAFM) steel developed in JAERI, have been investigated as one of the most promising structural materials for the ITER breeding blanket modules and DEMO blankets. In 2004, neutron and ion irradiation experiments using HFIR, JMTR, and so on have been continued.

6.1.1 Irradiation Effects on Fracture Toughness Properties

Under the Japan-US collaboration of fusion materials research, neutron irradiation experiments using High Flux Isotope Reactor (HFIR) in ORNL are in progress. In 2004, the irradiation schedules of a target capsule of JP26 and four rabbit capsules were completed. The JP26 mainly includes samples for the research of fracture toughness of F82H. Neutron irradiation was started for the target capsules of JP27, JP28 and JP29, aiming at neutron irradiation damage level of 17, 50 and 50dpa, respectively. As an example of the status of the neutron irradiation database of F82H, data of ductile-to-brittle transition temperature (DBTT) shift vs. displacement damage at irradiation temperatures between 300 and 400 °C are summarized in Fig. IV.6.1-1 [6.1.1].

It was found that DBTT shift tends to saturate with the damage level and RAFMs like F82H exhibit smaller shifts compared with conventional martensitic steels. However, transmutation-produced helium atoms have been suggested to enhance the DBTT shift, through observations using B- or Ni-doped alloys. As there seems to be a positive correlation between DBTT shift and hardening by irradiation, hardening effect due to doping itself must be extract from the observed data.

6.1.2 Irradiation Effects on Fatigue Properties

Post irradiation fatigue tests at room temperature were performed for F82H irradiated in JRR-3 to 3.8dpa at 250°C [6.1.2]. Fig. IV.6.1-2 shows the result that irradiation causes no change in fatigue life, except for the case of the smallest plastic strain range of 0.04%. The number of cycles to failure decreased to ~15% of the unirradiated specimen at the strain range. Based on the observations of fracture surface, it was suggested that the reduction might be attributed to channel deformation under cyclic stress.
(2) an increase of Cr and decrease of W contained in precipitates, and (3) the disappearance of MX (TaC) in ORNL9Cr and JLF-1 (see Fig. IV.6.1-3). The results generally suggest that irradiation effects on precipitation in RAFs appeared to accelerate the approach to thermal equilibrium, but there are still several discrepancies in the results, and further detailed analyses on microstructure are required.

Fig. IV.6.1-3 XRD peaks of extracted residue from unirradiated and irradiated RAFs. Peaks marked with diamonds correspond to the peaks from M\textsubscript{23}C\textsubscript{6}, and those marked with triangles correspond to peaks from MX (TaC).

6.1.3 Irradiation Effects on Precipitation Behavior

The effects of irradiation on precipitation of RAFM were investigated to determine how these effects might affect the mechanical properties [6.1-3]. The precipitation behavior of the irradiated steels was examined by weight analysis, X-ray diffraction analysis, and chemical analysis on extraction residues. These analyses suggested that irradiation caused (1) an increase of the amount of precipitates (mainly M\textsubscript{23}C\textsubscript{6}),

References

6.2 Transition Phase Activities in IFMIF

International Fusion Materials Irradiation Facility (IFMIF) is an accelerator-based D-Li neutron source designed to produce an intense neutron field that will simulate the neutron environment of a D-T fusion reactor. IFMIF will provide a neutron flux equivalent to 2 MW/m² in a volume of 500 cm³ and will be used in the development and qualification of materials for fusion systems. In 2003, "Comprehensive Design Report (CDR)" has been published to dedicate judgment of decision maker for promotion of IFMIF project in each party [6.2-1]. After the "Key Element Technology Phase (KEP)" during 2000-2002, transition phase activities have been under way prior to Engineering Validation and Engineering Design Activity (EVEDA). Major achievements during 2004 are summarized as follows.

6.2.1 Accelerator

In the IFMIF, a 175MHz Radio-Frequency Quadrupole (RFQ) linac is used for a pre-accelerator unit, and a deuteron beam of 125 mA is designed to accelerate from 0.1 to 5 MeV. In order to accelerate the large current beam of 125 mA, the operation frequency of 175MHz is selected, and a 12m-long RFQ linac is needed for the output energy of 5 MeV. The cross section area of transmission line using rectangular waveguide at this frequency becomes 1x0.5 m², and accordingly an RF input coupler using a conventional iris type is also large. Therefore, an RF input coupler using the co-axial transmission line was investigated with a multiple loop antenna configuration. It was found that a good power balance could be obtained by using a configuration with four loop antennas in comparison with that with single loop antenna or two loop antennas though the examinations at low power level. Calculations of electric field profiles in beam bore when installing a slug tuner were also calculated by the MW-Studio. The results suggested that the design requirement of distortion limit of 1% could be attained by employing 1cm slug tuner with insertion depth of 1 cm or less. As one physical section of the RFQ will be 1m-long, 20 tuners per one section (5 locations with 20cm interval and 4 tuners for each quadrant) is necessary for satisfying the required RF power-balance of 10% [6.2-2].

![Slug Tuner and RFQ](image)

Fig. IV.6.2-1 RFQ and slug tuner for RF power-balance control.

6.2.2 Li Target Facility

A thermal-stress analysis of the back-wall of IFMIF target during neutron irradiation was performed by using the FEM code ABAQUS with a heat source simulating the neutron induced nuclear heating (25 W/cm³ maximum at the center). Calculated temperature in the back-wall ranged from 300 to 440 °C by assuming heat transfer coefficient between back-wall disc and base assembly, α = 13 W/m²K. In a rotation binding condition at outer edge of the back-wall disc, thermal stress resulted in ~50 MPa and 2 mm displacement of the back-wall center, where the wall thickness is 1.8mm. However, in case of α = 150 W/m²K by clamping with a pressure of 1 MPa, thermal stress was reduced to 260 MPa, which is almost equal to acceptable value for the stainless steel, and the displacement becomes 0.3 mm [6.2-3].

To design radiation shielding and to determine scenario of maintenance around Li loop, an accessibility to the loop was evaluated by using ACT-4 code of THIDA-2 system and QAD-CGPG2R code considering...
depositions of corrosion products from activated back-wall. Activities after 1 day, 1 week, 1 month and 1 year cooling were 3.1x10^{14}, 2.8x10^{14}, 2.3x10^{14} and 7.5x10^{13} Bq/kg, respectively. It was found that, in the case of 10% deposition of the corrosion materials, hands-on maintenance becomes possible assuming a permissible level of 10 μSv/h (Fig. IV.6.2-2). Therefore, a removal of radioactive material in Li more than 90% is needed for the maintenance work. Also, effect of other source such as ^7Be is under going [6.2-4].

To confirm safety condition of the IFMIF Li loop and appropriateness of design of tritium-processing system, the permeation and inventory of tritium in the IFMIF Li loop was evaluated considering temperature and area of each component. Evaluated amount of total tritium permeation rate was 9.6x10^5 Bq/h, of which about 95% was a permeation rate at the V-Ti hot trap due to the highest temperature among the component. The total permeation rate was sufficiently lower than a capacity of an exhaust detritiation system for the tritium contaminated Ar. The total tritium contained in the walls wet by liquid Li in the IFMIF is 5.3x10^7 Bq, and is much smaller than 4.9x10^{15} Bq of the tritium contained in the Li flow with a volume of 9 m^3 [6.2-5].

![Fig. IV.6.2-2 Dose rate around Li loop](image)

6.2.3 Test Cell Facility
In the high flux region of IFMIF, two different types of high flux test module (HFTM) to irradiate specimens have been proposed so far, i.e., liquid metal bonding vs. helium gas gap methods to fill the space between the specimens and their holder (capsule). The structural design of the latter case was carried out and Fig. IV.6.2-3 shows the result that contains 3 rigs, each consisting of 3 capsules. The temperature of specimen is kept at the testing temperature by nuclear heating (~5W/g) and the electric heaters. It is also allowed to assemble by use of remote handling because the specimens will be reloaded to as many capsules as required to accumulate dose up to 100–150 dpa. To know the compatibility between specimens and liquid metal, the mass transfers of metallic elements and non-metallic elements were evaluated [6.2-6].

HFTM will be equipped with many thermocouples and the change of the thermo-electric characteristics may occur due to neutron irradiation. The calculation shows that about 1 wt% Fe is formed by the transmutation in the Ni based thermocouple during one year irradiation.

![Fig. IV.6.2-3 IFMIF High Flux Test Module design by JAERI](image)

References
7. Tritium Technology

7.1 Tritium Processing Technology Development for Breeding Blanket

In the research and development of BTR (Blanket Tritium Recovery system), an electrochemical hydrogen pump using ceramic proton conductor is a promising candidate of the advanced BTR. Its hydrogen transportation properties have been investigated to apply to the ITER test blanket. An outline design of the electrochemical hydrogen pump has been carried out using a simple approximation analysis code, and the prospect of BTR using the hydrogen pump has been recognized.

In this year, the research was focused on hydrogen transportation process in ceramic proton conductor. The correlation between hydrogen concentration in gas phase and proton concentration in ceramic was investigated to make a more detailed simulation code. The proton concentration in the ceramic decreased with the hydrogen partial pressure in the gas phase, but the dependence of the partial pressure became weak when water vapor existed in the gas phase. From this result, it is considered that the amount of oxygen in the ceramic that depends on the humidity in the gas phase affects the proton concentration in the ceramic.

A pressure swing adsorption has been studied as a new water processing method for a fusion power plant: the first stage of the system treating a large amount of tritium water. A series of adsorption and desorption experiments were carried out by using several kinds of zeolites with tritium water. One of the most important issues for this technology is to develop an adsorbent having a large desorption rate for water. It has been observed that the ratio of Si/Al of the zeolite is a key parameter controlling the desorption rate of the water. The temperature of the purge gas in the desorption was also a significant parameter for this technology.

Tritium produced in the blanket is released in the form of not only hydrogen molecules but also water. For the processing of tritiated water in the BTR system, the electrolytic reduction method using Yttria Stabilized Zirconia (YSZ) have been proposed. For increasing the efficiency of the process, new types of electrodes contained Ceria (CeO₂) were developed. A double layer electrode, which consists of Ceria layer and Pt/YSZ compound layer, achieved three times higher current density, which represents the reduction efficiency, than that of a standard electrode (single layer electrode mixed with Pt and YSZ). From the viewpoint of simplicity for the manufacture process, a single layer electrode mixed with Ceria and Pt was also tested. Fig. IV. 7.1.1 shows a comparison of the current density of the three kinds of electrodes. The single layer electrode containing Ceria also showed an increase in the current density, although the current density of single layer containing Ceria was slightly lower than that of double layer electrode. This result indicates that the electrode containing Ceria is useful in spite of its simple structure and easy fabrication.

![Graph](image)

Fig. IV.7.1-1 Current density through the YSZ for each applied voltage.

7.2 Tritium Accounting Technology Development

For developing an alternative mean for on-line and real-time tritium monitoring, the method of detecting bremsstrahlung X-rays was examined by measuring the counting rate of bremsstrahlung X-rays as a function of pressure in two mixed tritium gases diluted with hydrogen or helium [7.3-1]. As shown in Fig. IV.7.2-1, at a constant tritium partial-pressure ratio of 0.010, the relationship between the counting rate of bremsstrahlung X-rays, X (cpm) and the total pressure, P (Pa) can be expressed as $X = 4.1 \times 10^3 \left(1 - e^{-4.6P/10000}\right)$. In addition, the relation between X and P can be simplified to a linear function in the low-pressure region of $P < 2.5 \times 10^4$ Pa. This figure also indicates that the counting rate depends only on the tritium partial-pressure and the total pressure if the mixed gases contain no other species but hydrogen isotopes and helium isotopes, suggesting that the present method of tritium monitoring by measuring the counting rate of bremsstrahlung X-rays is very promising for the fuel..
processing system of fusion reactors, especially for the tritium recovery system of breeding blankets.

References

7.3 Basic Study on Tritium Behavior
For plasma-surface interaction studies, the ion species ratios in low energy high flux deuterium plasma beams formed in a linear plasma generator were measured, and the species control in the plasma generator was evaluated by changing the operational parameters like neutral pressure, arc current, and axial magnetic confinement to the plasma column [7.3-1]. As a result, we can achieve plasma beams highly enriched with a single species of D⁺, D₂⁺, or D₃⁺, to a ratio over 80% with suitable adjustment.

In order to establish effective surface decontamination methods for the construction materials of fusion facilities, a so-called ‘soaking’ effect is important and more systematic data of tritium sorption and desorption of various materials are required under actual conditions. This effect is based on sorption of tritiated water on the materials and subsequent desorption from them.

A series of tritiated water vapor exposure experiments on various materials such as stainless steel (SS), acryl resin, and epoxy paint, has been carried out as a function of time in order to accumulate ad/absorption data of tritium on the materials. Sorption of various materials (epoxy paint, butyl rubber, SS304 and acrylic resin) saturated within 1 ~ 4 weeks.

The amounts of tritium sorbed on epoxy paint, butyl rubber and acrylic resin were several thousand ~ tens of thousands times higher than that on SS304, showing the order of epoxy paint > acrylic resin = butyl rubber >SS304. In the desorption experiment, the desorption rates of epoxy paint and acrylic resin during purging of N₂ gas were evaluated. Fig. IV.7.3-1 shows the desorption curves for blank and epoxy paint in the chamber by purging of N₂ gas (containing less than 10 ppm of water vapor). When the water vapor concentration in the purge gas is less than 10 ppm, the desorption rates were relatively small compared with the reported data values obtained in the purge gases of higher water vapor concentrations. The difference would be caused by the concentration of the water vapor in the purge gas. When several thousands ppm of water vapor was added in the chamber after the desorption by purging gas of N₂ for 370 minutes, 20%~50% of tritium sorbed on the materials were quickly removed by the isotope exchange reaction between water vapor and tritium sorbed on the materials. It becomes clear from this result that it is effective to add water vapor into purging gas for the tritium removal from the surface of the materials.

To have better understanding of radiochemical reactions among oxygen baking products in a fusion reactor, reactions in equimolar tritium molecule (T₂) and carbon dioxide (CO₂) were examined by laser Raman spectroscopy and mass spectrometry. After mixing them at room temperature, both T₂ and CO₂ decrease as the elapsed time increases in the first 30 minutes after
mixing, and then stay at almost stable values, suggesting that some reactions between $T_2$ and $CO_2$ occur intensively within the first 30 minutes and then achieve a quasi-stable state. As the predominant products of the reactions, carbon monoxide (CO) and tritiated water ($T_2O$) were found in gaseous phase and condensed phase, respectively. In addition, polymers such as polyformaldehyde are likely formed as the solid products of mixtures of $CO_2$ and $T_2$, and they are thermally decomposed into $CO$, $CO_2$, $T_2$ and $T_2O$ during baking up to 523 K [7.3-2].

In a fusion reactor, high-level tritiated water of more than PBq/m$^3$ will be generated and stored in the various areas. The high-level tritiated water decomposes by itself and generates hydrogen mainly, and becomes a tritiated hydrogen peroxide water. In order to summarize safety requirements for long-term storage of high-level tritiated water, the characteristics, such as effective G-values of hydrogen and hydrogen peroxide, pH and Oxidation Reduction Potential (ORP), have been investigated storing tritiated water from TBq/m$^3$ to EBq/m$^3$ for several years. Because the effective G-value of hydrogen increased with decreasing tritium concentration, the storage tank for wider range of tritium concentration requires not only depression but also adequate processing of the cover gases. High-level tritiated water of more than PBq/m$^3$ was acid and some case was corrosive condition, and these characteristics have been maintained for several years. Therefore, the ORP-pH situation should be monitored periodically to avoid changing to the corrosive condition.

References

7.4 Successful Operation Results of Tritium Safety Systems in TPL
The safety system of Tritium Process Laboratory (TPL) consists of Glove Box Gas Purification System (GPS), Air Cleanup System (ACS), Effluent Tritium Removal System (ERS) and Dryer Regeneration System (DRS). The GPS was operated for about 8,000 hours by controlling tritium concentration in the glove boxes.

The ACS was operated for cleaning 84,100 m$^3$ of air during the experiments of Caisson Assembly and maintenance of the glove boxes, experimental apparatus and other tritium operations. The ERS removed about 4.3 TBq of tritium mainly out of the exhaust gas from the experimental apparatus. The DRS removed 127 liters of tritiated water (220 GBq) from the GPS and ACS dryers.

The tritium safety system of TPL has been in service to support operations with use of tritium since 1988. Some maintenance work such as periodical inspection or replacement of a superannuated glove box has been carried out in this fiscal year. Additionally, software upgrade of a central control system of the safety system of TPL has begun. Fig. IV.7.4-1 shows monthly environmental tritium release from the stack of TPL during this fiscal year. Total amount of released tritium was 65 GB, which is sufficiently lower than the target value at TPL.

![Fig. IV.7.4-1 Monthly environmental tritium release from the stack of TPL during this fiscal year.](image-url)
8. Fusion Neutronics

8.1 Blanket Neutronics Experiments

8.1.1 International Benchmark of TPR Measurement for Blanket Neutronics Experiments

Integral experiments on the blanket mock-up with D-T neutrons are a key issue to verify the accuracy of the tritium production rate (TPR) evaluation in the blanket. From the viewpoint of the fusion blanket design, the accuracy of the prediction of the tritium breeding ratio should be less than 5%. Therefore, it is important to verify the accuracy of the TPR measurement technique through benchmark tests.

The international tritium measurement benchmark started as a part of the neutronics sub-task under the IEA collaboration. JAERI, ENEA and the Technical University of Dresden (TUD) are participating in the activity. Several steps were planned in order to meet the objective. To identify a common start point, the fist step was performed with HTO samples without any treatment procedures. Measured data were analysed in comparison with theoretical values that were evaluated based on the HTO provider certificate. The analysis showed that all measurements were in agreement with theoretical values within the estimated uncertainty. The second step included treatment procedures for lithium carbonate pellets (Li₂CO₃) irradiated in the uniform condition by 14-MeV neutrons at the Fusion Neutronics Source (FNS). Results of the tritium activity in the Li₂CO₃ pellets measured by three parties are presented in Fig. IV.8.1-1. There is an agreement between tritium activities obtained at JAERI and TUD. However, the value obtained at ENEA show an underestimation when compared with FNS and TUD values. Further analysis is in progress to explain the underestimation, which may result from either the tritium losses or the calibration in the chemical procedure.

8.1.2 Blanket Mockup Experiment

In the fusion DEMO reactors, the blanket is required to provide a tritium breeding ratio (TBR) of more than unity. The solid breeding blanket being developed by JAERI for tokamak-type DEMO reactors consists of lithium titanate (Li₂TiO₃) or other lithium ceramics as the tritium breeder material, beryllium as the neutron multiplier material, reduced activation ferritic steel F82H as the structural material and water as the coolant. Using DT neutrons, neutronics integral experiments have been performed with the blanket mockup at FNS facility in JAERI [8.1-1]. Figure IV.8.1-2 shows a schematic view of the experimental assembly. The mockup is a rectangular parallelepiped with the dimensions of 450 mm in height, 450 mm in width, and 360 mm in thickness. The mockup is constructed by a set of layers of the first wall panel, water panel, Li₂TiO₃ and beryllium with the dimensions of 16, 7.8, 12 and 101.6 mm in thickness, respectively. It is composed of two Li₂TiO₃ layers and three beryllium layers. The Li₂TiO₃ layers are sandwiched by the beryllium layers. Enrichment of ⁶Li is about 40% in lithium of Li₂TiO₃.

The first wall and water panels are mainly composed of F82H and water. The water is filled in the box structure made of F82H and SS316. The first wall panel is composed of the 3 mm thick front wall, 6 mm thick water and 7 mm thick back wall. The water panel is composed of the 1.8 mm thick front wall, 4.2 mm thick water and 1.8 mm thick back wall. Numerical calculations have been conducted using the Monte Carlo neutral particle transport code MCNP-4C with the Japanese Evaluated Nuclear Data Library JENDL-3.3. Figure IV.8.1-3 shows distributions of the ratio of the calculation value to experimental one (C/E) for the local TPRs. The C/E in the first and second Li₂TiO₃ layers are 1.01 and 1.08 for the integrated tritium productions. The C/E is 1.03 for sum of the integrated tritium...
productions in the first and second Li$_2$TiO$_3$ layers. It can be concluded that sum of the integrated tritium productions are predicted accurately.

Fig. IV.8.1-2 Schematic view of the experimental assembly.

8.2 Cross Section Measurements for Fusion Materials

8.2.1 Charged-particle Emission DDX Measurement

It is important to measure double-differential cross sections of emitted charged-particles (DDXc) induced with DT neutrons for evaluations of the kinetic energy released in materials and the primary knock-on atom spectrum in fusion reactor materials.

The measurements of DDXc for beryllium and carbon have been completed using the FNS D-T neutron beam and a silicon surface barriered detector counter telescope method. Figure IV.8.2-1 shows the obtained DDXc for $^9$Be(n,xα) nuclear reaction and the evaluated nuclear data (JENDL-3.3 and ENDF-B/VI). Some differences were pointed out from the comparison of the present experimental data and the evaluated one. Further analysis to reveal the reaction mechanism and the contribution from 3-body breakup is in progress.

Fig. IV.8.2-1 DDX measured for $^9$Be(n,xα) reactions at 15, 45 and 105 degrees of emission angle and comparison with JENDL-3.3 and ENDF-B/VI.

8.2.2 Activation Cross Sections of IFMIF Accelerator Structural Materials

In the design of IFMIF (International Fusion Materials Irradiation Facility), long-term operation with more than 70 % total facility availability is required. However, the activation of structural materials composing the IFMIF accelerator due to the bombardment by deuterons beam limits the maintenance time and makes the long-term
operation difficult. Therefore, the accurate estimation of deuteron-induced radioactivity and the selection of low activation structural materials are important. Thus, measurements of deuteron-induced activation cross sections for structural materials (vanadium, iron, nickel and tantalum) were performed on the basis of a stacked-foil technique at TIARA. As a result of the experiment, we have obtained the activation cross sections for the reactions $^{60}$V(d,x)$^{63}$Cr, $^{63}$Ni(d,x)$^{66,61}$Cu or $^{55}$Co, $^{56}$Fe(d,x)$^{58,59}$Co and $^{178,180}$Ta(d,x)$^{181}$Ta in 15-40 MeV range. These results were compared with other experimental cross sections and estimated data in the ACSELAM library. Figure IV.8.2-2 shows the cross sections for the $^{56}$Fe(d,x)$^{59}$Co. Present results agreed with the previous data obtained by Hermann, et al. However, ACSELAM data overestimate them by a factor of 4 at the maximum.

![Cross section graph](image)

Fig. IV.8.2-2 Cross sections for the $^{56}$Fe(d,x)$^{59}$Co reaction.

8.3 Beam Analyses of the Armor Tiles

Hydrogen isotopes show complicated behavior on the surface of plasma facing components (PFCs) in fusion devices. The study is important for the design of the fuel recycling, plasma control, etc. In FNS, microanalytical studies for PFC have been developed by applying the particle beam method [8.3-1].

Depth profiles of hydrogen isotopes in PFCs of the Tokamak Fusion Test Reactor (TFTR) of the Princeton Plasma Physics Laboratory were measured by the deuteron-induced nuclear reaction analysis. DT plasma experiments were made from 1993 to 1997, and clean-up treatments of PFC surfaces were made using helium discharge, baking, etc. after the experiments. The analyzed sample (15 mm x 20 mm x 10 mm) was a part of a tile made of a carbon fiber composite, which was placed at K bay, column C, row 16 of the inner bumper limiter.

Deuterium and tritium depth profiles are shown in Fig. IV.8.3-1. Tritium density is multiplied by 50 in the figure. Negative values of depth arose from the finite depth resolution of the detector system. The tritium concentration had a peak at 0.5 µm with an atomic density of $7.4 \times 10^{25}$ T/m$^3$, whereas the deuterium showed a broad distribution up to the depth of 1.5 µm with atomic densities of $3.4 \times 10^{27}$ D/m$^3$. The both of them were distributed over a deeper region in the profiles because of the co-deposition of hydrogen isotopes with eroded carbon particles that resulted from TFTR operations of plasmas in contact with the inner bumper limiter.

![Depth profile graph](image)

Fig. IV.8.3-1 Deuterium and Tritium depth profiles in plasma facing components of TFTR evaluated from NRA energy spectra.

References


8.4 Water Cherenkov Detector

Feasibility studies of the new fusion power monitor [8.4-1] require development of a water Cherenkov detector that will satisfy neutron monitoring requirements.

The new detector design is based on collection of direct and diffusive Cherenkov light by quartz fibers. The radiator is viewed by quartz fibers that trap the light within the fiber acceptance cone. The detector design is shown in Fig. IV.8.4-1 and is elaborately described in the reference [8.4-1]. The radiator is viewed by two quartz fiber bundles inserted inside the water at the opposite side of the radiator. Each fiber bundle contains twenty quartz fibers with a fiber diameter of 0.1 cm. Experimental tests of the detector were carried out at the
FNS facility. Temporal detector parameters were studied using the source pulse mode for different water flow rates. The typical detector response to the pulsed neutron flux is shown in Fig. IV.8.4-2. Based on obtained results, it is possible to conclude that the detector temporal parameters are quite adequate to the current experimental conditions. Design and the experimental setup are not yet optimized for best performance; obtained detector parameters demonstrate potential capabilities of the proposed design.

The developed water Cherenkov detector with a quartz fiber readout has demonstrated the ability to work properly in a radiation environment and appears suitable for a D-T neutron monitoring system using activation of flowing water. Temporal detector parameters can be improved according to neutron monitoring requirements by optimizing the detector design.

8.5 Operation of the FNS Facility

Operation of the FNS facility has been carried out to cope with various operation patterns in a variety of experiments requested by JAERI and the other collaborative parties. The total operation time was 940 hours in the fiscal year 2004. In those experiments, eight fixed small targets and two rotating large targets were consumed for the 80° and 0° beam line operations, respectively. The total amounts of tritium were 200 GBq for the small targets and 30 TBq for the large targets. Then, the tritium amount in the vacuum exhaust processed by the Tritium Adsorption Processor (TAP) system reached 15.3 TBq.

Maintenance activities performed are as follows. As a result of the periodic check-up, the oil-free vacuum pumps and CRYO pump mounted on 0° beam line were overhauled. The accelerator control circuit was inspected every six months.

As for renewal activities, the performance of the pulse beam system was improved to obtain a shorter pulse width and a long pulse with a rectangular shape. The beam deflector was replaced with a double-deflector. The length of the deflector electrode was shortened from 150 mm to 50 mm. The main unit for making trigger pulse was changed from the vacuum tube into the FET. The rise time of the trigger pulse voltage was shortened. As a result of the improvement, shortening the pulse width from 50 to 30 nsec became possible and the peak current of the pulse beam could be increased from 1 mA to 2 mA. In the state of long pulse with 0.1 ~ 1 sec width, the pulse shape was also improved like a rectangular shape.

Fig. IV.8.4-2 Detector response to the pulsed neutron flux.

References

9 Vacuum Technology

For R&Ds on fuel supplying/pumping system, separation/recovery tests on gas mixture of 4-fluoro carbon and 6-fluoro ethane (a kind of greenhouse gases) have been performed using a large-scale test device (Fig. IV.9-1) under corporation with ORGANO Corp., as the application of H₂/He selective pumping technique (C³: Continuous Circulation Chromatograph) method.

As the results, each gas with purity of 99.9 % was obtained with recovery rate of 99 % (flow rate of gas mixture: 10 little/min) after separation (Fig. IV.9-2). These results have been reported at the international symposium on semiconductor manufacturing (ISSM 2004). The paper has been selected as one of the ISSM 2004 Best Papers for publication in the prestigious IEEE Transactions on Semiconductor Manufacturing.

Fig. IV.9-1  Large-scale test device.

Fig. IV.9-2  Experimental results of separation of CF₂-C₂F₆ mixtures.
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V. International Thermonuclear Experimental Reactor (ITER)

1. Overview of the ITER Program and Activities

1.1 ITER Transitional Arrangements (ITA)
In January 2003 the ITER Transitional Arrangements (ITA) started under the auspices of the IAEA, following the successful completion of the Coordinated Technical Activities (CTA). The purpose of the ITA is to prepare for an efficient start of the Agreement, if and when so decided, and to maintain the integrity of the ITER Project. Along the work plan approved by the ITER Preparatory Committee (June 2004), the Design and R&D Tasks started among the Participant Teams (PTs). Based on the task agreement between the International Team (IT) and each PT, these shared Tasks make the implementation of preparing the procurement documents for facilities and equipments that are scheduled to be ordered at an early stage of ITER construction such as superconducting magnets and vacuum vessel sectors. In FY2004 JAERI was in charge of fifty-five Design Tasks that are described in 2.1.

1.2 Progress of Negotiations and Prospective Schedules
Current participants in ITER negotiations are six parties; Japan, the European Union, the United States, Russia, South Korea and China. Technical aspects of the two candidate sites, Rokkasho and Cadarache, were analyzed by working groups consisting of members from the parties. "A Broader Approach" concept to fusion power was also discussed in workshops and a number of initiatives have been identified which would be advantageous for the development of fusion. These are ITER Research Centre, Satellite Tokamak and International Coordination for Fusion Power Technology including power plant design, material development and IFMIF. Bilateral negotiations over the site's location were ongoing between Japan and the European Union based on a viewpoint of the Broader Approach concept. Delegations of the six parties also met in June and November to discuss on the site. The schedule for ITER construction is shown in Table V.1.2-1 with recent progress as of March 2005.

Table V.1.2-1 Schedule for ITER Construction

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(as of March 2005)
2. Domestic Activities

2.1 ITA Design Task

2.1.1 Superconducting Magnets

JAERI performed several manufacturing studies for the procurement of TF coil structures and CS jacket including full-scale trial fabrication according to the ITER Task Agreements.

(1) TF Structure

The following products of JJ1 (0.03C-10Mn-12Cr-12Ni-5Mo-0.24N) and nitrogen strengthened 316LN (0.03C-2Mn-18Cr-11Ni-2Mo-0.20N, referred to as ST316LN) for the TF coil cases and radial plates were produced to demonstrate mass productivity and mechanical properties at cryogenic temperatures.

- JJ1 rectangular forged block: 3.7 m (length) × 0.94 m (width) × 0.4 m (thickness)
- ST316LN rectangular forged block: 4.7 m (length) × 0.96 m (width) × 0.43 m (thickness)
- ST316LN (SUS316LN) hot rolled plate: 1.6 m (length) × 1.8 m (width) × 0.2 m (thickness)
- ST316LN (SUS316LN) hot rolled plate: 4.8 m (length) × 1.8 m (width) × 0.14 m (thickness)

Figure V.2.1-1 shows an overview of a ST316LN block at an intermediate stage of several forging processes. These products will be fully qualified by using ultrasonic testing, microscopic observation, and cryogenic testing of mechanical properties by the end of November 2005.

Manufacturing studies of coil cases, radial plates and intercoil structures are in progress under the collaboration with heavy industries, in order to develop a feasible manufacturing plan including segmentations, machining, welding and inspection of TF coil structures. Demonstration activities to support the manufacturing study have also been performed in parallel with the design work. A radial plate, which satisfies the ITER requirement on flatness of within 1 mm, could be fabricated in a trial fabrication as shown in Fig. V.2.1-2. Manufacturing of radial plates according to ITER procurement schedule was also demonstrated to be feasible by using a rational manufacturing process in which combination of parallel machining of grooved plates and their connection by YAG laser welding are used.

(2) CS Jacket Section

A stainless steel having a low coefficient of thermal expansion is required as a CS jacket material to obtain compressive force on winding packs by cooldown from room temperature to 4K. From this point of view, a new material, JK2LB was selected for the CS jacket. Trial fabrications of circle-in-square JK2LB tubes with the processes of hot extrusion followed by cold drawing were performed to confirm achievable accuracy of dimensions and available unit length of the jacket section. The produced jacket section is shown in Fig. V.2.1-3. It was confirmed that tolerance of ± 0.25 mm for both outer and inner dimensions and unit length of more than 7 m could be guaranteed as specifications, which satisfies the ITER requirements.
2.1.2 Vacuum Vessel
A large number of interface structures, such as keys and housings to support the blanket modules on the VV, have been introduced in the present design of the VV instead of the backplates in the 1998 ITER design, in order to reduce the procurement cost. As a result, the number of weld joints in the VV has increased and the distance between weld joints becomes much shorter than that in the previous design, for which fabrication feasibility has already been demonstrated in the EDA R&D of VV. The weld joints of housings for blanket support are very close to the reinforced ribs connected between inner and outer walls of VV especially in the inboard cylindrical region. Furthermore, accessibilities of welding and weld inspection are extremely restricted by the highly curved inner and outer walls in the top curved region. The critical issues in the present VV design are therefore to confirm feasibilities of fabrication and quality inspection even with the expected large welding distortion and the low accessibility for welding and weld inspection.

JA-PT is now fabricating a partial VV mock-up (shown in Fig. V2.1-4), which consists of an inboard cylindrical region and a top curved region, in order to confirm the fabrication methods and NDT methods and in order to examine whether distortions caused by welding are within tolerances. The basic data necessary for preparing the technical specifications of the VV procurement, such as weld distortions, minimum space required for welding and inspection, welding and inspection accessibilities to the weld joints in the present design and applicability of weld inspections, will be obtained through the fabrication of the partial mock-ups.

2.1.3 Blanket and Divertor
As for electro-magnetic analysis of blanket modules in updated disruption scenarios, detailed 3-D EM analyses have been performed for linear current decay with 40 ms quench time, based on renewed disruption scenarios. Almost all EM forces due to eddy currents are kept below the design allowable limits except for a few inboard modules. For the marginal modules, design improvements have been made such as deepening slots for #5 module and applying the stiffer key structure for #9 module in order to meet the design requirements. For stress analyses of blanket supports, limit analyses and plastic analyses have been carried out to calculate the collapse loads of the support keys. The allowable load of the keys obtained from the analyses is enough to support the design loads of the blankets, considering dynamic amplification factor of 1.5. From the results of plastic analyses, residual deformations of the keys are within the tolerances between the keys and key grooves.

For the design of module #10, a new inner-structure design has been investigated based on the new slits position evaluated from the EM analysis, taking inner pressure, neutron wall loads, flow distribution and suitable fabrication method into account. On the design of first wall (FW) support connection between FW panel and shield, structures and maintenance method with cutting and rewelding to replace FW has been examined based on the race track type connection developed by IT. As for the alternative design of FW support connection, a new two circular tubes type with using the disc cutter for cutting has been investigated as a mean of adopting the high practicable method in Hot Cell at the maintenance stage instead of the race track one with using the Laser beam cutting proposed by IT.

As for R&D on improved blanket design, the partial mockup of first wall panel with real scale width (350mm width x 400mm length x 70mm thickness) was fabricated using CuCrZr for the heat sink material and SUS316L backing plate for the purpose of evaluation of the selected conditions of Hot Isostatic Pressing (HIP) joining of CuCrZr and SUS316L in the real scale
fabrication process (Fig. V.2.1-5). The mockup features the most important structures, built-in SUS cooling channels and the coolant collector at the edge parts of the first wall panel. Dimensional accuracy of most important part, location of built-in cooling channels, was within 1% and met the requirement for temperature control of the structure.

![Fig. V.2.1-5 Fabricated First Wall Mockup with CuCrZr and SUS316L.](image)

An annular flow cooling arrangement was proposed as a means of eliminating the exposed coolant tubes at the lower end of the divertor vertical target with the aim of providing a more robust and compact design. Design of small-scale CFC mockups with annular flow cooling tubes for high heat flux testing has just been started in this year. In addition, to investigate the applicability of an all-tungsten armored target as a back-up option of the divertor, R&D on the divertor plate with tungsten pin armor has been started. In parallel, as for the acceptance inspection of divertor plate, a high temperature coolant feed system has been added to an existing high heat flux test facility. This system is capable not only to perform a high heat flux testing under hot coolant condition up to 100 °C as one of the acceptance inspection items of the divertor, but also to simulate the divertor baking condition at around 200 °C.

2.1.4 Tokamak Assembly

ITER tokamak assembly procedures have been investigated for the sub-assembly of the 40 degree sector composed of two TF coils, vacuum vessel sector and thermal shield in the sub-assembly hall, and for the final assembly of the 9 x 40 degree sectors into the full torus in the tokamak pit, respectively. The most critical issue for ITER tokamak assembly is that the wedges of the adjacent TF coils have to be assembled with a target gap less than 0.3 mm between them. In order to solve this issue, new procedures for the sub-assembly and final assembly have been developed after review of the current IT design. In particular, for the final assembly of the 9 sets of the sub-assembled sectors, the installation sequence of the sub-assembled sectors has been modified instead of the IT procedure of sequential installation of all sub-assembled sectors in a clockwise direction. That is, it has been proposed that three sets of the sub-assemblies are first installed at every 120 degree and adjacent sub-assemblies are then installed to each of the three sub-assemblies, respectively, in order to minimize the accumulated installation error at the final sectors in the clockwise procedure for all sectors. The metrology concepts, and scenarios necessary to control the dimensions of the sector and full torus assemblies have been also investigated including the position control method for the TF coils, TF coil wedge adjustment method, and position and dimension measurement schemes for the VV sector and the TF coil.

2.1.5 Remote Maintenance

According to the change of IT design such as blanket segmentation and structure, the remote handling scenario/procedure for blanket maintenance has been studied to improve the major interfaces between blanket module and remote handling equipment, such as key configuration and end-effector. To avid the interference between modules and remote handling equipment, kinematic (CAD) analyses are performed. As a result, some dimensions of the remote handling equipment, such as arm length, rotational mechanism around the rail and height of end-effector, are modified, including structural analysis.

In addition to the design study, the Blanket Test Platform (BTP) has been updated in order to improve the sensing technique for module installation and removal. The mock-up composed of typical inboard and outboard modules has been also updated considering the
detailed interfaces such as key structure and gap between modules based on the latest IT design.

2.1.6 EC and NB Heating Systems
For heating system, following five tasks have been agreed between IT and PT (Japan), and design and R&D works have started. Three tasks are for electron cyclotron (EC) heating and current drive system and two tasks for neutral beam (NB) heating and current drive system. Titles and the contents are as follows:

(1) Design of the Equatorial Launcher and Development of Launcher Components
The design of overall system layout for equatorial launcher will be performed, in which structural, nuclear and thermal analyses are included. The development and test of components for the launcher will be also carried out using 170GHz gyrotron.

(2) Development and Test of Diamond Window
A low loss synthetic diamond window has prepared. On the edge of the diamond disk, Cu layer is coated to protect a brazing part from a corrosion by water. The coating also has a role as a protector from the water leakage to the vacuum vessel when the cracking occurred in the disk. The disk will be assembled for the launcher window and the high power transmission experiment will be carried out.

(3) Long Pulse Operation of 170GHz Gyrotron
A long pulse experiment will be performed with the 170GHz gyrotron with the power level of 1MW. For this purpose, cooling system and the control system have been upgraded for CW operation at 1MW output.

(4) ITER Neutral Beam Test Facility Development
The design work of facility infrastructure for the test bed of NB system will be performed.

(5) High Current, High Current density, Acceleration of H/D (200A/m²) up to 1MeV
Negative hydrogen ion beam acceleration experiment is in progress aiming at the current density of 200A/m² up to the beam energy of 1MeV level with good beam divergence. Also, development of large bore ceramic insulator ring has been started as part of vacuum insulation technology development for the ITER NB accelerator.

2.1.7 Operation Scenario
(1) Plasma Operation
The ITA task related plasma operation scenario has been studied to contribute to the detailed design of ITER through analyses of plasma equilibria taking account of the recent design improvement of the poloidal field coil position and geometry.

Design Scenarios of fusion power 400MW with burn time 400s, 360MW/3000s and Assessed Scenario of 70MW/100s were updated with the recent design improvement of PF system. It was confirmed that the coil currents, the maximum magnetic fields, the operation voltages, the electromagnetic forces on the PF coils and the burn fluxes were within their design limits.

A modification of 400MW/400s operation scenario where inboard region of the first wall is used as a start-up limiter was produced. The coil currents, the maximum fields and the electromagnetic forces are within the design limits. To keep the design limits of the CSJ coil voltage, the duration during the ramp-up phase should be longer several seconds than the values of original scenario. However, this PF scenario would be available due to decrease of the heat load on the limiter. At the same time, a limiter space can be used for various plasma shapes. Increasing of the plasma shape to the outside will result in improving of the coupling with ICRF and the plasma performance.

(2) Tritium Plant
The conditions imposed on the tritium plant and the other systems handling tritium for these standard states have formed the basis of the design of the subsystems and for the operation as an integrated plant. The Plant Integration Document (PID) of ITER specifies a number of standard states of operation, including short-term and long-term maintenance, plasma operation, wall-conditioning, standby, etc. However, there are a certain number of “sub-standard” and “non-standard” states for the tritium plant including some “common Site-specific” ones besides the standard ones described in PID, e.g., recovery from Ingress of Coolant Event (ICE), tritium recovery from a transfer capsule, tritium recovery from heavily tritium contaminated plasma
facing material to be wasted, etc., for which tritium plant operation is required.

To recognize and describe all the subsequent operations of the tritium plant and also to check any additional devices to be newly required, investigation of all the potential operational states, including non-standard ones, and definition of the operational conditions of the tritium plant for these have been carried out.

As a result, there are some important "sub-standard" operational states for the tritium plant to be differentiated under the standard operations described in PID, one example is processing of deuterium (including process of deposited deuterium in the vacuum vessel) exhausted during the D-plasma or D-GDC operations expected in D phase or even in DT phase. This affect the deuterium storage capacity of SDS (which will depend on how much deuterium contaminated with tritium, which will be generated by D-D reactions or remained in the vacuum vessel after D-T plasma experiments, is accumulated in the vacuum vessel) and operation of ISS (which shall be done by change in the hydrogen isotope profiles in the cryogenic distillation columns). Re-confirmation of operational requirements from plasma physics/diagnostics shall be needed for assessment of the maximum amount of deuterium to be supplied/stored in the D and DT phases.

It is rather a site-specific matter, but it is necessary to consider the way to treat water highly contaminated tritium in some safe manner, when Ingress of Coolant Event (ICE, which is not regarded as an "accident," because the in-vessel components are "test components", which were not safety-credited.) takes place. A fairly large amount of water from the in-vessel component cooling system, which the pipe has leak/break, will be introduced into the vacuum vessel and the vacuum vessel pressure suppression system. In the worst case, this water introduced into the vacuum vessel contained tritium of hundreds grams. Any scenario to treat this water has not been fully established yet, but it shall be fixed according to the site-specific conditions as a part of basis for the safety assessment under normal operation. In addition, after the ICE, a scenario of tritium recovery inside the vacuum vessel in a condition that all the in-vessel components are wet shall be made for the maintenance of the vacuum vessel and that of in-vessel components to be carried out in Hot cell successively. Some possible scenario including up-grading the Water Detritiation System (WDS) is being examined.

(3) Review Ventilation Systems

According to the current design, the ITER buildings including the tokamak system, tritium plant and the hot cell have ventilation systems comprising Heating and Ventilation Systems (HVAC) and Atmosphere Detritiation Systems (ADS) and Vent Detritiation Systems (VDS), as well as extraction ducts and additional associated equipment to maintain building depression.

The design and layout of the ventilation systems has evolved over many years in accordance with the progress of design and layout of the buildings, and in particularly in conjunction with the development of the detailed confinement strategy and implementation in ITER. The development of the latter over many years has contributed to the fact that with time additional equipment has been added to the initial systems, which has resulted in a number of dedicated, independent systems, hence complexity of the operation scenario of the systems.

The task was carried out to review the present ITER design and layout of the ventilation systems from viewpoints of (i) functional adequacy, (ii) simplifications. As to check the first item, some reliability evaluation of ventilation and detritiation systems was carried out based on operation scenario under normal and accidental conditions. As a result, assuming an accident with tritium release the probability of failure in mitigation by that of the detritiation systems can satisfy the level required for the reliable mitigation system for ITER. However, it was revealed that the VDS for Hot cell (HC-VDS), which will have to run whenever the Hot cell encloses wastes contaminated with activated ducts and tritium, may fail at a rather high probability due to a high frequency (12 valves activated/cycle × 2 cycles/d × 365 d/y = 8760 times/y) of switching valves for absorption/regeneration operation of molecular sieve dryers, though that will be back-up by another redundant system, HC-ADS. Now some possible improvements to make the failure rate of HC-VDS lower is being examined.

Also, zoning and arrangement of HVAC in the Hot cell was examined. The simplification of zoning and
ducting has been considered mainly by re-arrangement of the amber zone, which contains transition areas between red and green zone.

As changes allowed for Japanese site, ventilation systems in the building, into which all the tokamak, tritium plant and hot cell are merged for simplification of the seismic isolation structures, were conceptually designed. Due to use of a single air handling unit for all the areas in the building and merging green zones in the tokamak and hot cell areas, some simplification of the system design was achieved.

2.1.8 Tritium-Material Interactions

To discuss the tritium behavior in the concrete used for the ITER buildings and the hot cell, a series of exposure experiments has been carried out for a standard concrete, a cement paste and a mortar sample. The exposure experiments have been carried out within a chamber by releasing tritium water vapor into it. In the exposure experiments for a maximum of two months, tritium water vapor was captured about 2 cm deep in the standard concrete as shown in Fig. V.2.1-6. The tritium water vapor was captured into the concrete by the adsorption as well as the isotope exchange reaction, since the concrete has an appreciable amount of water in it. It was expected that the tritium capture be caused by the complicated phenomena described above.

The gas of SF₆ is used as an insulating material in ITER. Under the assumption of an accidental release of SF₆, the effect of SF₆ on the performance of tritium removal facilities was studied. The tritium facilities consist of a catalysts bed and an adsorption bed. It was observed that the SF₆ gas was cracked by the catalysts bed, and the cracked SF₆ gas decomposes the water produced by the catalyst bed. As a serious effect of the SF₆ gas, the decrease of the detritiation factor of the tritium removal facilities was thus clearly observed. On the other hand, in the case where the supply of the SF₆ gas was stopped, it was observed that the cracked SF₆ gas was removed from the catalysts bed, and the performance of the tritium removal facilities was gradually recovered.

A series of irradiation tests of the ion exchange membrane of the electrolyzer of WDS (Water Detritiation System) of ITER was carried out from the viewpoints of its durability against tritium. The durability of the membrane for the tensile strength and ion exchange capacity roved to be 850 kGy (three years use under ITER condition). The membrane is made from a fluorine resin. The fluorine ions should be dissolved into water with the degradation of the membrane by the irradiation. Fig. V.2.1-7 shows the irradiation dose dependence of the fluorine ions in the water. It was observed that the amount of dissolved fluorine ions from the membrane could correlate closely with the tensile strength and the ion exchange capacity. It can be concluded that we can monitor the durability of the membrane by continually measuring the amount of fluorine ions in the water.

![Fig. V.2.1-6 Distribution of tritium in the standard samples (concrete, cement paste and mortar).](image)

![Fig. V.2.1-7 Irradiation dose dependence of dissolved fluorine ions into water.](image)

2.1.9 Shielding Analyses

Three-dimensional Monte Carlo shielding analyses are conducted on the ITER NBI duct. The surface heat fluxes are evaluated for the bremsstrahlung and line radiations distributions by the photon transport
calculation. The nuclear heating rates are evaluated for the nuclear fusion reaction distribution by the neutron and photon transport calculation. The detailed distributions of these heat loads are evaluated on the side wall facing the plasma, that hidden from the plasma, upper and bottom walls. The analytical representations of these nuclear responses are established as functions of the distance from the blanket surface and the duct wall surface.

Previously, the duct size larger than 1.2 m × 1.2 m in cross section, more than one bends and the first leg longer than 300 cm were estimated necessary in the design of a pressure relief line to reduce the streaming effect. The next problem is to decrease the permeating components from the first to the second or third leg through the duct shield. The neutron permeation through the shield was studied by Monte Carlo calculations with MCNP code. The result mentions that 15 cm thick iron shield must be enough to suppress the permeating component from the outside. In addition, the volume of the shield can be reduced by about 30% if the optimized iron shield structure having localized thickness along intense penetration path is employed to shield the pressure suppression line.

2.1.10 Plasma Diagnostics
Under the ITA Design Task on plasma diagnostics, “Support to the ITER diagnostic design,” design works on Impurity Influx Monitor (divertor), Microfission Chambers, Thomson Scattering (edge) System, and Integrator for magnetic measurement were conducted. Relating to the Diagnostic Port Engineering Taskforce, initial analyses of the neutron shielding and penetration in the upper port, and electromagnetic force on the upper port and the port plug were carried out.

2.2 Site Related Activities
2.2.1 Site Preparation
A site-dependent design study has been in progress to prepare for siting of ITER in Japan.

Water supply and sewage systems have been designed based on the number of workers during the construction phase. Rearrangement of the site layout from the 2001 design has been considered with the Rokkasho site conditions such as land, water and power supply etc.

Since seismic isolation is adopted for the tokamak complex building, the passage between isolated and non-isolated buildings is one of the concerns as to whether the criteria of radiation control area can be satisfied at the passage. At this point, the concept of the use of the tokamak complex building has been discussed to minimize cask transfer through the passage.

Some methods of excavation for tokamak complex building have been studied. Methods of providing a water shield wall to deal with the ground water, an approach ramp and its combination have been compared. By this study, the basic data has been obtained to decide the excavation method from a viewpoint of shortening the schedule after the site selection. Study of the final heat sink has also been continued.

2.2.2 Codes and Standards
For formulating the ITER structural code by American Society of Mechanical Engineers (ASME), the draft for construction of vacuum vessel and superconducting magnets has been prepared through the review by code experts under cooperation with the ITER International Team. Also, the Japan Society of Mechanical Engineers (JSME) started review of the relevant codes independently in the neworganized Subcommittee on fusion power. The completion of the first draft to be submitted for examination in the ASME is aimed at the end of 2005.

As the first part, the draft mentioned above has been made through the review by code experts from a viewpoint of code acceptability. For facilitating this activity, JSME organized some working groups under the new Subcommittee on fusion power.

2.3 Contributions to International Tokamak Physics Activity (ITPA)
Naka Fusion Research Establishment continued to contribute to all the aspects of International Tokamak Physics Activity (ITPA). Its emphasis is inter-machine experiments of key plasma characteristics, aiming at development of methodologies of projection and control of ITER and power reactor plasmas. The following shows main contributions to ITPA, details of which were shown in section I, II and III.

2.3.1 Transport Physics Topical Group
One group meeting was held and two members participated. As the candidate of steady-state plasma,
weak shear discharges with $q_{\text{min}}^{-1.5}$ and $q_{\phi}^{-1.5}$ have been developed; $\beta_N=2.4$ and $f_{\text{RB}}^{-45\%}$ were maintained for 5.8 s. In a strong reversed shear plasma, $f_{\text{RB}}^{-75\%}$ was maintained for 7.4 s. Regarding hybrid regime development, long sustainment of high $\beta_N$ ($\beta_N=2-3$) was achieved in high $\beta_p$ H-mode plasmas with $q(0)-1$ and $q_{\phi}^{-3.4}$. In addition, $\beta_N=2.5$ was maintained for 16.5 s. In QH/QDB-mode studies, toroidal rotation was investigated by varying co- and counter-NB power, and QH-mode with a nearly zero rotation at the edge was observed. Profile data of a long-sustained (15.5 s) high $\beta_N$ (2.5) plasma and scalar data of advanced scenario discharges has been submitted for transport database.

2.3.2 Confinement Data and Modeling Topical Group
One group meeting was held and two members participated. The dependence of $\beta_T$ on the energy confinement was examined keeping $\rho_{-\tau}$ and $\nu_\perp$. The degradation of energy confinement with increasing $\beta_T$ was observed, satisfying the relation of $B_T \tau_E \sim \beta_T^{-0.6-0.7}$. This dependence is a little weaker than that predicted by the IPB98(y,2) scaling. For inter-machine exp. on improving the condition of global H-mode and pedestal databases, small difference in the pedestal structure between D and H discharges with the same $\beta$ but different $\rho^*$ suggests weak $\rho^*$ dependence of $\Delta_{\text{ped}}$.

2.3.3 Pedestal and Edge Physics Topical Group
One group meeting was held and three members participated. The following subjects have contributed to ITPA. The type of ELM can be controlled by the toroidal rotation from type-I to grassy. No edge fluctuation was observed in contrast with quiescent H-mode (QH-mode). Regarding QH-mode, QH-mode has been produced without counter-NB in JT-60U, although one of the requirements for producing QH-mode in DIII-D, ASDEX-U and JET is counter-NB. Nearly zero toroidal rotation at the edge was observed in QH-mode with co-NB, suggesting that large counter-toroidal rotation and counter-NB are not necessary conditions for producing QH-mode. As for the simulation, the finite-$n$ ideal MHD stability code (MARG2D) has been developed based on the eigenvalue problem associated with the Newcomb equation for the analysis of ELMs.

2.3.4 Steady-State Operation Topical Group
One group meeting was held and three members participated. Topics is the long sustainment of $\beta_N=2.3$ and $G>0.4$ for 22.3 s by optimizing high $\beta_p$ ELMy H-mode plasma. This duration corresponds to $t_{\phi}$. The time evolution of $\beta_N$ was carefully optimized in order to reach $\beta_N$ as high as possible but avoiding the neo-classical tearing mode (NTM). Steady state candidate plasma described in 2.3.1 was also presented.

2.3.5 MHD Topical Group
One group meeting was held and three members participated. It was shown that early ECCD is more effective to suppress NTM. Island size ($-d_B/d_B/|l|$) is quickly suppressed and calculated necessary power for full suppression based on modified Rutherford equation agrees well with the experiments. The contributions for disruption were fast shutdown by mixed gas injection and mitigation of post-disruption runaway electrons. The former result show that lower growth rate of runaway generation can be achieved for the Kr injection case in comparison with Ar and Xe cases. Later one is effective on prompt exhaust of runaway electrons and reduction of runaway plasma current by impurity pellet injection.

2.3.6 Scrape-Off-Layer and Divertor Physics Topical Group
One group meeting was held and two members participated. Hydrocarbon was injected into divertor region and emission of CD band etc was measured to improve chemical sputtering data, and it was shown that photon efficiency was comparable to Behring/Nakano models and previous data. Experiment using W-coating C-tiles installed at outer divertor in JT-60U has shown that erosion of tile was generally small, and redeposition was found locally. In addition, fluctuation level at midplane was 5 times larger than X-point & divertor. Statistical analysis (Probability Distribution Function) showed that non-gaussian distribution (suggesting intermittent transport) appeared at midplane. ELM propagation towards first wall was evaluated at midplane SOL: radial velocity (1.3-2.5 km/s) was larger than other tokamaks (0.5-1 km/s).

2.3.7 Diagnostics Topical Group
Two group meetings were held and six members participated. Relating to high priority topics, alpha particle diagnostics (CO2 laser CTS, He beam, Multi-
Foil), neutron diagnostics (microfission chamber, liquid activation system), magnetic measurement (long-pulse integrator), and radiation effects on diagnostics components have been studied. Design studies on laser oscillator for edge Thomson scattering system, impurity influx Monitor (divertor), Microfission Chamber have been conducted towards ITER. Poloidal FIR laser polarimeter/interferometer, toroidal CO₂ laser polarimeter/interferometer, imaging diagnostics (bolometer, ECE, and reflectometer), diagnostic beam/laser/gyrotron development (High repetition rate LD-pumped Alexandrite laser for LIDAR) and advanced plasma control (jtr) and NTM) have been also studied as ITER-relevant diagnostics. A new project Advanced Diagnostics for Burning Plasma Experiment has been initiated.

2.4 Engineering Safety Demonstration

2.4.1 Dynamic Vibration Test for Gravity Support

The vibration experiments of the support structures with flexible plates for ITER major components were performed using small-sized flexible plates aiming to obtain its dynamic behaviors such as dependence of the stiffness and frequency on the loading angle. The experimental results were compared with the analytical ones in order to estimate an adequate analytical model for ITER support structure with flexible plates.

The results are summarized in the followings [2.4-1, 2.4-2]. Firstly, the experimental results obtained by the hammering and frequency sweep tests were agreed each other, so that the experimental method is found to be adequate. The basic mechanical characteristics such as dependence of the stiffness and frequency on the loading angle are obtained as a basis of the support structure with flexible plates (see Fig. V.2.4-1).

Secondary, the numerical analyses were performed for comparison with the experimental results. As a result, the bolt connection of the flexible plates on the base plate strongly affected on the stiffness of the flexible plates. Among three typical analytical models, the analytical results modeling the bolts with finite stiffness only in the axial direction and infinite stiffness in the other directions agree well with the experimental ones (see Fig. V.2.4-1). This analytical model is therefore found to be adequate to estimate the dynamic behaviors of the support structure with flexible plates fixed by bolts.

Finally, the support structure with flexible plates and bolt connection was modeled as a spring model composed of only two spring elements simulating the in-plane and out-of-plane stiffness of the support structure with flexible plates including the effect of connection bolts. Using the simplified spring model, the dynamic analysis of the VV and TF coil for the ITER were performed for the estimation under the design earthquake at Rokkasho area. As a result, it is found that the maximum relative displacement of 8.6 mm between VV and TF coil is much less than 100 mm, so that the integrity of the major components (VV and TF coil) of the ITER tokamak device are ensured for the earthquake event.

Fig. V.2.4-1 Basic performance tests on vibration of support structure with flexible plates.

References

2.4-1 Takeda, N., et al., J. Plasma and Fusion Research, 80, 988 (2004).

2.4-2 Takeda, N., et al., to be published in J. Plasma and Fusion Research.
VI. FUSION REACTOR DESIGN STUDY

1. Reactor Design Study

1.1 Conceptual Design of Fusion DEMO Plant

Recent reactor design activity at the Japan Atomic Energy Research Institute has been mainly focused on fusion DEMO plant which is placed beyond ITER. The DEMO plant aims to demonstrate 1) an electric power generation of 1 GW level, 2) self-sufficiency of T fuel, 3) year-long continuous operation. At the same time, DEMO should present an economical prospect for commercialization. To meet the requirements, three options with different capabilities of CS are considered: 1) the "Slim CS" option, being capable of providing sufficient Ampere-turns for plasma shaping but supplying a V-sec limited to \(I_p\) ramp-up to 3.8 MA; 2) the "Full CS" option, being capable of plasma shaping and V-sec supply enough to reach the flattop plasma current, like conventional tokamaks; 3) the "CS-less" option, having no CS functions but the most compact reactor size. The "CS-less" option corresponds to VECTOR concept [1.1-1]. The parameters comparison between those three options is shown in Table VI.1-1.

Table VI.1.1 Parameter for CS-less, slim CS and full CS options

<table>
<thead>
<tr>
<th></th>
<th>CS-less</th>
<th>Slim CS</th>
<th>Full CS</th>
</tr>
</thead>
<tbody>
<tr>
<td>(R_p) (m)</td>
<td>5.1</td>
<td>5.5</td>
<td>6.5</td>
</tr>
<tr>
<td>(A) (m)</td>
<td>2.1</td>
<td>2.1</td>
<td>2.1</td>
</tr>
<tr>
<td>(\kappa)</td>
<td>2.5</td>
<td>2.6</td>
<td>3.1</td>
</tr>
<tr>
<td>(\delta)</td>
<td>2.1</td>
<td>2.0</td>
<td>1.9</td>
</tr>
<tr>
<td>(B_T/B_{max}(T))</td>
<td>5.6/18.2</td>
<td>6.0/16.4</td>
<td>6.8/14.6</td>
</tr>
<tr>
<td>(I_p) (MA)</td>
<td>17.4</td>
<td>16.7</td>
<td>15.0</td>
</tr>
<tr>
<td>(q_{95})</td>
<td>4.5</td>
<td>5.6</td>
<td>5.1</td>
</tr>
<tr>
<td>(\beta_N)</td>
<td>4.8</td>
<td>4.3</td>
<td>3.9</td>
</tr>
<tr>
<td>(H_H)</td>
<td>1.3</td>
<td>1.3</td>
<td>1.3</td>
</tr>
<tr>
<td>(n/n_{GW})</td>
<td>0.94</td>
<td>0.98</td>
<td>1.0</td>
</tr>
<tr>
<td>(P_{em})</td>
<td>3.0</td>
<td>3.0</td>
<td>3.0</td>
</tr>
<tr>
<td>(P_{a}(MW/m^2))</td>
<td>3.7</td>
<td>3.5</td>
<td>3.0</td>
</tr>
<tr>
<td>(Q)</td>
<td>48</td>
<td>52</td>
<td>54</td>
</tr>
<tr>
<td>Weight (tons)</td>
<td>15,700</td>
<td>17,500</td>
<td>23,900</td>
</tr>
</tbody>
</table>

Among these three options, "Slim CS" is being considered to be the prime option in view of the tradeoff between required technologies especially in plasma physics and the reactor size. In this sense, "Slim CS" is a design compromise on the advanced commercial reactor VECTOR. This option shown in Fig. VI.1-1 produces the fusion output of 3 GW with the major radius of 5.5 m, the aspect ratio of 2.6, the normalized beta of 4.3 and the maximum field of 16.4 T. The estimated reactor weight is lighter than other conventional tokamak reactors, suggesting an economic advantage. The plant uses rather conservative technologies such as Nb3Al superconductor, water-cooled solid breeder blanket, low activation ferritic steel as the structural material and tungsten monoblock divertor plate.

Fig. VI.1-1 Conceptual view of DEMO "slim CS" option

References


1.2 Current Ramp Simulation for CS-less Tokamak Reactor

Simulation study on internal transport barrier (ITB) of tokamak plasma has been carried out in the aspect of reactor design study.

A non-inductive current ramp-up, which is necessary to start up a CS-less tokamak like VECTOR, was investigated with the TSC code [1.2-1,1.2-2]. It was shown that a cooperative linkage between the non-inductive current and ITB-generated bootstrap (BS) current exhibited a recurrence of positive and negative shear profiles when a CDBM (current diffusive balloning mode) model was adopted. In addition, another cooperative linkage between BS current and the resulting modulated magnetic shear showed an oscillatory current ramp for highly BS current-driven plasmas. Figure VI.1.2-1 shows the
consequent zero magnetic shear profile arising from the oscillation. It was shown that a profile misalignment between the BS current and the magnetic shear gave rise to these oscillatory behaviors.

![Fig. VI.1.2-1 Flattened shear profile over wide region of 0 < $\rho$ < 0.6 and inward drifting negative loop voltage, arising from cooperative link between BS current and BS current-modulated shear profile. Shear profile is widely supressed to nearly “ZERO”, avoiding Current Hole formation.](image)

2. Waste Management Study

2.1 Nitrogen Concentration of F82H for Shallow Land Burial

Reduced activation ferritic steel (F82H) is considered to be the prime structural material for fusion DEMO plant. The relation between nitrogen concentration in F82H and the waste classification of the used F82H was assessed. Previously, neutronics assessment indicated that nitrogen concentration should be as low as 20 ppm to dispose of most of the used F82H by shallow land burial. This is because nitrogen can produce carbon-14 which plays a critical role in waste classification. Such a low nitrogen concentration can adversely affect toughness and creep strength of F82H. Thus, nitrogen concentration of 100-200 ppm is required on the material side. In order to find a good compromise and provide a guide line for the material development, the allowable concentration ratio of nitrogen for the shallow land burial was estimated with neutronics codes. The result indicates that most of the used F82H can be qualified to be disposed by shallow land burial when 95%-enriched nitrogen-15 is used at 100 ppm in F82H fabrication instead of nitrogen with natural abundance.

2.2 Waste Assessment for D-3He Reactor [2.2-1]

Quantitative comparison of waste was carried out for D-T fuel and D-3He fuel fusion reactors. There is an opinion that D-3He reactors produce much smaller amount of waste than D-T reactors. However, our waste assessment indicates that the D-3He reactor will not necessarily have an advantage of reducing the radioactive waste although the reactor is widely believed to reduce the load on the environment compared with a D-T reactor because of its very low neutron yield. Our assessment indicates that the D-3He reactor with the electric output of 1 GWe will produce radioactive waste of more than 10,000 tons due to parasitic D-T reactions. The amount will be more than the radwaste of the D-T reactor with the same electric output. From the point of view of waste management, a disadvantage of the D-3He reactor is that it requires massive poloidal field coils to maintain the equilibrium of plasma with several tens of MA. If we assume an extremely advanced technology of disassembling the superconducting coils into individual homogeneous materials, the waste of the coils would qualify for clearance waste, reducing the radioactive waste to as low as 1,835 tons.

References

Appendix A.1 Publication List (April 2004 – March 2005)

A.1.1 List of JAERI Report


A.1.2 List of papers published in journals


A.1.3 List of papers published in conference proceedings


12) Hayashi, H., Takizuka, T., Ozeki, T., “Profile Formation and Sustainment of Autonomous Tokamak Plasma with Current Hole Configuration,” 20th IAEA FEC 2004 (Vilamoura, Portugal), accepted in Nucl. Fusion.


A.1.4 List of other papers


Appendix A.2 Personnel and Financial Data

A.2.1 Charge in number of personnel and annual budget (FY1993-2004)

A.2.2 Organization Chart (Mar.31, 2005)
A.2.3 Scientific Staff in the Naka Fusion Research Establishment
(April 2004- March 2005)

**Naka Fusion Research Establishment**
- SEKI Masahiro (Director General)
- FUJIWARA Masami (Scientific Consultant)
- INOUUE Nobuyuki (Scientific Consultant)
- SHIMOMURA Yasuo (Scientific Consultant)
- MATSUI Hideki (Invited Researcher)
- KOHYAMA Akira (Invited Researcher)
- YAMADA Hiroshi (Invited Researcher)
- KISHIMOTO Yasuaki (Invited Researcher)
- AZUMI Masafumi (Prime Scientist)
- USHIGUSA Kenkichi (Staff for Director General)
- ISEI Nobuaki (Staff for Director General)
- OOHARA Hiroshi (Staff for Director General)

**Department of Administrative Services**
- KIKUCHI Isao (Director)
- KOBAYASHI Haruo (Deputy Director)

**Department of Fusion Plasma Research**
- NINOMIYA Hiromasa (Director)
- KIKUCHI Mitsuru (Deputy Director)
- NAGAMI Masayuki (Prime Scientist)
- TERAKADO Yuichi (Administrative Manager)

**Tokamak Program Division**
- MIURA Yukitoshi (General Manager)
- FUJITA Takaaki
- KURITA Gen-ichi
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- TSUCHIYA Katsuhiko

**Plasma Analysis Division**
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- KUBO Hirota (Deputy General Manager)
- ASAKURA Nobuyuki
- HOSHINO Katsumichi
- ISAYAMA Akihiko

---

- HAYASHI Nobuhiko
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- TAKIZUKA Tomonori

- KAWASHIMA Hisato
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- TAMAI Hiroshi

- IBA Katsuyuki (*24)
- NAITO Osamu
- SAKATA Shinya
- SUZUKI Mitsuhiko (*31)

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- INOUE Akira (*20)
- KAMIYA Kensaku
KASHIWA Yoshitoishi  KAWANO Yasunori  KITAMURA Shigeru
KOIDE Yoshihiko  KOKUSEN Shigeharu (*19)  KONDOH Takashi
KONOSHIMA Shigeru  MATSUNAGA Go (*22)  MIYAMOTO Atsushi (*19)
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SAKUMA Takeshi (*20)  SHINOHARA Kouji  SUNAOSHI Hidenori
SUZUKI Takahiro  TAKECHI Manabu  TAKENAGA Hidenobu
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SUGAHARA Akiohito (*24)  TOKUDA Shinji  TUDA Takashi

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KASAI Satoshi  OGAWA Hiroaki  SHIINA Tomio

Reactor System Laboratory
TOBITA Kenji (Head)
KURIHARA Ryoichi  NAKAMURA Yukihiro  NISHIO Satoshi
SATO Masayasu

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HOSEOGANE Nobuyuki (Deputy Director)
YAMAMOTO Takumi

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SHIMADA Katsuhiko  SUEOKA Michiharu  TAKANO Shoji (*31)
TERAKADO Hirohiko (*7)  TERAKADO Tsunehisa  TOTSUKA Yoshiyuki
YAMASHITA Yoshihiko (*7)  YONEKAWA Izuru

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MIYO Yasuhiko  NISHIYAMA Tomokazu  SASAJIMA Tadayuki
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SATO Fumiaki (*19) SAWAHATA masayuki SEKI Masami
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NOTO Katsuya (*19) OKANO Fumihori OSHIMA Katsumi (*19)
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SHIBATA Takatoshi (Deputy General Manager)
SAITO Hideo (*20)

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TAKATSU Hideyuki (Prime Scientist and Deputy Director)
KATOGI Takeshi (Administrative Manager)
SHIHO Makoto (TSUJI Hiroshi passed away on December 18, 2004)

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INOUE Takashi KASHIWAS Yoshitada (*4) KASHIWAGI Mieko
KASUGAI Atsushi KOBAYASHI Noriyuki (*30) KOMORI Shinji (*20)
MINAMI Ryutaroh (*22) ODA Yasuhiro (*32) SEKI Takayoshi (*3)
TAKADO Naoyuki (*13)  TAKAHASHI Koji  TANIGUCHI Masaki
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UZAWA Masayuki (*18)  YAMADA Masayuki  YAMANISHI Toshikiko

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ANDO Masami  FURUYA Kazuyuki  IDA Mizuho (*5)
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NAKAMURA Kazuyuki  TANIGAWA Hiroyasu  YUTANI Toshiaki (*30)
UMETSU Tomotake (*16)

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NISHITANI Takeo  (Head)
ABE Yuichi  KONDO Keitaro (*21)  KUBOTA Naoyoshi (*22)
KUTSUKEKE Chuzo  NAKAO Makoto (*12)  OCHIAI Kentaro
OGINUMA Yoshikazu (*20)  SATO Satoshi  SEKI Masakazu
TANAKA Shigeru  YAMAUCHI Michinori (*30)  VERZIOV Yury (*6)

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HOSHINO Tsuyoshi (*22)  ISHIDA Takuya (*20)  TSUCHIYA Kunihiko
YAMADA Hirokazu (*12)

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MATSUMOTO Hiroyuki  (Administrative Manager)
KOIZUMI Koichi
ODAJIMA Kazuo

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ANDO Toshiro  (Head)
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KATAOKA Yoshiyuki (*3)  MARUYAMA So  MIZOGUCHI Tadanori (*3)
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OOHASHI Hironori (*11) SATO Kazuyoshi SEKIYA Shigeaki (*16)
TAKAHASHI Hideo (*27) TAMURA Kousaku (*15) YAGENJI Akira (*2)
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NAKAHIRA Masataka OBARA Kenjiro TAKEDA Nobukazu

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department of Material Science

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JITSUKAWA Shiro (Group Leader)
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OKUBO Nariaki SAWAI Tomotsugu SHIBA Kiyoyuki
TANIFUJI Takaaki YAMADA Reiji YAMAKI Daiju
WAKAI Eiichi

Department of Nuclear Energy System

Research Group for Reactor Structural Materials
MIWA Yukio

Neutron Science Research Center

Research Group for Neutron Scattering from Functional Materials
IGAWA Naoki

Research Group for Nanostructure
TAGUCHI Tomitsugu

*1 Graduate University for Advanced Studies
*2 Hazama Corporation.
*3 Hitachi, Ltd.
*4 Ibaraki University
*5 Ishikawajima-Harima Heavy Industries Co., Ltd.
*6 JAERI Fellowship
*7 JP HYTEC Co., Ltd.
*8 Japan Atomic Power Company Co., Ltd.
*9 Japan EXPert Clone Corp. (JEX)
*10 Kajima Corporation

*11 Kandenko Co., Ltd.
*12 Kawasaki Heavy Industries, Ltd.
*13 Keio University
*14 Kokan-Keisoku Corporation
*15 Kounoike Construnction Co., Ltd.

*16 Kumagai Gumi Co., Ltd.
*17 Mitsubishi Electric Corporation
*18 Mitsubishi Heavy Industries, Ltd.
*19 Nippon Advanced Technology Co., Ltd.
*20 Nuclear Engineering Co., Ltd.

*21 Osaka University
*22 Post-Doctoral Fellow
*23 Princeton Plasma Physics Laboratory (USA)
*24 Research Organization for Information Science & Technology
*25 Shimizu Corporation

*26 Sumitomo Heavy Industries, Ltd.
*27 Taisei Corporation
*28 Tokyo Institute of Technology
*29 Tomoe Shokai Co., Ltd.
*30 Toshiba Corporation

*31 Total Support Systems
*32 University of Tokyo
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国際単位系（SI）と換算表

### 表1 SI基本単位および補助単位

<table>
<thead>
<tr>
<th>名称</th>
<th>記号</th>
<th>記号</th>
</tr>
</thead>
<tbody>
<tr>
<td>長さ</td>
<td>メートル</td>
<td>m</td>
</tr>
<tr>
<td>質量</td>
<td>キログラム</td>
<td>kg</td>
</tr>
<tr>
<td>時間</td>
<td>秒</td>
<td>s</td>
</tr>
<tr>
<td>電流</td>
<td>アンペア</td>
<td>A</td>
</tr>
<tr>
<td>熱力学温度</td>
<td>ケルビン</td>
<td>K</td>
</tr>
<tr>
<td>物質の密度</td>
<td>モル</td>
<td>mol</td>
</tr>
<tr>
<td>光速</td>
<td>カンデラ</td>
<td>cd</td>
</tr>
<tr>
<td>均等角ラジアン</td>
<td>rad</td>
<td></td>
</tr>
<tr>
<td>立体角ラジアン</td>
<td>sr</td>
<td></td>
</tr>
</tbody>
</table>

### 表2 SIと併用される単位

<table>
<thead>
<tr>
<th>名称</th>
<th>記号</th>
<th>記号</th>
</tr>
</thead>
<tbody>
<tr>
<td>分、時、日</td>
<td>min, h, d</td>
<td></td>
</tr>
<tr>
<td>周、分、秒</td>
<td>°, ′, ″</td>
<td></td>
</tr>
<tr>
<td>リットル</td>
<td>L</td>
<td></td>
</tr>
<tr>
<td>トン</td>
<td>t</td>
<td></td>
</tr>
<tr>
<td>電子ボルト</td>
<td>eV</td>
<td></td>
</tr>
<tr>
<td>原子質量単位</td>
<td>u</td>
<td></td>
</tr>
</tbody>
</table>

1 eV=1.00218×10⁻⁵kg
1 u=1.66054×10⁻²⁷kg

### 表3 国軸の名称をもつSI補助単位

<table>
<thead>
<tr>
<th>名称</th>
<th>記号</th>
<th>記号</th>
</tr>
</thead>
<tbody>
<tr>
<td>強度</td>
<td>ヘルツ</td>
<td>Hz</td>
</tr>
<tr>
<td>压力</td>
<td>バスカール</td>
<td>Pa</td>
</tr>
<tr>
<td>計算エネルギー</td>
<td>ジョールト</td>
<td>J</td>
</tr>
<tr>
<td>工率、放射束</td>
<td>ウット</td>
<td>W</td>
</tr>
<tr>
<td>電気量、電荷</td>
<td>クーロン</td>
<td>C</td>
</tr>
<tr>
<td>電圧、電流密度</td>
<td>ボルト</td>
<td>V</td>
</tr>
<tr>
<td>電気容量</td>
<td>ファラド</td>
<td>F</td>
</tr>
<tr>
<td>電気強度</td>
<td>オーム</td>
<td>Ω</td>
</tr>
<tr>
<td>コンダクタンス</td>
<td>ジーメンス</td>
<td>S</td>
</tr>
<tr>
<td>組成密度</td>
<td>レンジ</td>
<td>Wb</td>
</tr>
<tr>
<td>電気伝導度</td>
<td>デシラム</td>
<td>T</td>
</tr>
<tr>
<td>星度</td>
<td>ステラ</td>
<td>T</td>
</tr>
<tr>
<td>深度</td>
<td>セルシウス度</td>
<td>°C</td>
</tr>
<tr>
<td>光度</td>
<td>ルー</td>
<td>lm</td>
</tr>
<tr>
<td>照度</td>
<td>アクス</td>
<td>lx</td>
</tr>
<tr>
<td>放射線量</td>
<td>ベクレル</td>
<td>Bq</td>
</tr>
<tr>
<td>吸収線量</td>
<td>グレイ</td>
<td>Gy</td>
</tr>
<tr>
<td>照度</td>
<td>シーベルト</td>
<td>Sv</td>
</tr>
</tbody>
</table>

### 表4 SIと共に定義時に導入される単位

<table>
<thead>
<tr>
<th>名称</th>
<th>記号</th>
<th>記号</th>
</tr>
</thead>
<tbody>
<tr>
<td>オンストークス</td>
<td>A</td>
<td></td>
</tr>
<tr>
<td>ペーセント</td>
<td>%</td>
<td></td>
</tr>
<tr>
<td>パーセント</td>
<td>%</td>
<td></td>
</tr>
<tr>
<td>ガル</td>
<td>G</td>
<td></td>
</tr>
<tr>
<td>キュリ</td>
<td>Ci</td>
<td></td>
</tr>
<tr>
<td>レントゲン</td>
<td>R</td>
<td></td>
</tr>
<tr>
<td>ラド</td>
<td>rad</td>
<td></td>
</tr>
<tr>
<td>レマ</td>
<td>rem</td>
<td></td>
</tr>
</tbody>
</table>

1 A=0.1mm=10¹⁵m
1 b=10⁰mm=10⁻⁸m
1 cm=1.01604³⁹m³
1 Ci=3.7×10⁻¹⁰Bq
1 R=2.58×10⁻⁷C/kg
1 rad=1cGy=10⁻²Gy
1 rem=1cSV=10⁻²SV

### 表5 SI換算

<table>
<thead>
<tr>
<th>名称</th>
<th>記号</th>
<th>記号</th>
</tr>
</thead>
<tbody>
<tr>
<td>力</td>
<td>MPa</td>
<td>bar</td>
</tr>
<tr>
<td>力</td>
<td>N</td>
<td>kgf</td>
</tr>
<tr>
<td>力</td>
<td>atm</td>
<td>mmHg (Tor)</td>
</tr>
<tr>
<td>1 MPa=10bar</td>
<td>1 N=0.101972 kgf</td>
<td></td>
</tr>
<tr>
<td>9.80665 N=1 kgf</td>
<td>2.20462 lbs</td>
<td></td>
</tr>
<tr>
<td>4.44822 kgf=1 lbs</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

### 换算表

<table>
<thead>
<tr>
<th>名称</th>
<th>記号</th>
<th>記号</th>
</tr>
</thead>
<tbody>
<tr>
<td>溫度</td>
<td>K</td>
<td>°C</td>
</tr>
<tr>
<td>1 Pa=1N/m²=10¹³m²/s²=10¹⁰Pa (ポアス)(g/cm²)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>性質</td>
<td>J=10¹⁰erg</td>
<td>kgf·m</td>
</tr>
<tr>
<td>1</td>
<td>2.77777×10⁻¹⁰</td>
<td></td>
</tr>
<tr>
<td>9.80685</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>3.6×10²</td>
<td>3.67098×10⁶</td>
<td></td>
</tr>
<tr>
<td>1.88505</td>
<td>0.42085</td>
<td></td>
</tr>
<tr>
<td>1055.06</td>
<td>107.58</td>
<td></td>
</tr>
<tr>
<td>1.53532</td>
<td>0.13825</td>
<td></td>
</tr>
<tr>
<td>1.60218×10¹⁰</td>
<td>1.63377×10⁻¹⁰</td>
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</tr>
</tbody>
</table>

### 表6 SI単位

<table>
<thead>
<tr>
<th>名称</th>
<th>記号</th>
<th>記号</th>
</tr>
</thead>
<tbody>
<tr>
<td>体積</td>
<td>リットル</td>
<td>L</td>
</tr>
<tr>
<td>体積</td>
<td>キログラム</td>
<td>kg</td>
</tr>
<tr>
<td>体積</td>
<td>原子質量単位</td>
<td>u</td>
</tr>
<tr>
<td>体積</td>
<td>電子ボルト</td>
<td>eV</td>
</tr>
<tr>
<td>体積</td>
<td>原子質量単位</td>
<td>u</td>
</tr>
</tbody>
</table>

1 eV=1.00218×10⁻⁵kg
1 u=1.66054×10⁻²⁷kg

(86年12月26日現在)