Annual Report of Naka Fusion Research Establishment
from April 1, 1997 to March 31, 1998

Naka Fusion Research Establishment

Japan Atomic Energy Research Institute
Naka-machi, Naka-gun, Ibaraki-ken

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This report provides an overview of research and development activities at Naka Fusion Research Establishment, JAERI, during the period from April 1, 1997 to march 31, 1998.
The activities in Naka Fusion Research Establishment are highlighted by high temperature plasma research in JT-60 and JFT-2M, and progress in ITER-EDA, including technology development.

The objectives of the JT-60 project are to contribute to the ITER physics R&D and to establish the physics basis for a steady state tokamak fusion reactor like SSTR.

Improvements and regulation of the facilities and developments of the instruments were performed. The construction for the divertor modification from the original open type to the W-shaped semi-closed type for improving the particle control was finished in May 1997. The modification intends to investigate effects of divertor geometry on divertor functions such as particle and impurity controls, and to realize radiative divertor compatible with good confinement.

With respect to the negative-ion based NBI, input power to the plasma was gradually increased along with improvement of operational optimization to attain 5.2 MW at 350 keV with deuterium negative ion beams and 4.2 MW at 360 keV with hydrogen negative ion beams.

Experiments simulating the helium exhaust in ITER were performed with the W-shaped pumped divertor. Helium atoms introduced in the ELMy-H plasmas for 6 sec by helium neutral beam injection were efficiently exhausted by helium pumping with Ar frosted cryopumps in the divertor. The experiments successfully demonstrate high helium exhaust capability of $\tau_{He}^{*}/\tau_E \approx 4$ in steady state, which satisfies the ITER requirement. These results strongly support the divertor design of ITER.

Because long heating time with a total heating energy of 203 MJ was achieved without harmful increase in impurity and particle recycling, a DT equivalent fusion gain of $Q_{DT}^{eq} \approx 0.1$ was sustained for 9 sec in a ELMy-H mode.

Toward the advanced feedback controls of multiple parameters, the JT-60U started new feedback controls of central line density and divertor neutral gas pressure in addition to the existing controls of off-axis line density, radiation power and neutron production rate. Characteristics of

halo current during disruptions were also studied. The NNB-driven current was identified directly from the internal magnetic measurement and driven current profile was confirmed to be consistent with the ACCOME calculation. The current profile control with LHCD successfully sustained the internal transport barrier in reversed shear plasmas. Continuous TAE modes were observed with NNB for the first time in the world as beam-driven TAE modes.

Objectives of the JFT-2M program are (1) advanced and basic researches for the development of high-performance plasmas for nuclear fusion and (2) contribution to the physics R&D for ITER, taking full advantage of flexibility of a medium-size device.

In the closed divertor experiments, it is found that the closer the divertor geometry becomes, the wider the high confinement regime coexistent with a dense and cold divertor plasma results. A compact toroid (CT) injection system has been installed in collaboration with the Himeji Institute of Technology for the development of the advanced fueling for fusion reactors, such as ITER. Encouraging results were obtained with initial CT injection experiments, such that reduction of radiation loss power was observed after the CT injection into OH plasmas. A heavy ion beam probe system, which was developed by the National Institute for Fusion Science has been installed for clarifying mechanism of improved confinement more definitely through fast measurements of the electric field.

The primary objective of theory and analysis is to improve the physical understanding of the magnetically confined tokamak plasma. Remarkable progress has been made on physical understanding of the reduced transport and the stability not only of ideal MHD modes but also of kinetic ballooning mode in reversed shear plasmas. Progress was also made on the neoclassical transport calculation by the Matrix Inversion method and on the scaling law of an offset nonlinear form for the ELMy-H-mode confinement. A five-point model for the scrape-off layer and divertor plasmas was developed and the inside/outside divertor asymmetry was investigated.

The main focus of the NEXT (Numerical Experiment of Tokamak) project is to simulate tokamak plasmas using particle and fluid models on the developing technology of massively parallel computers. A particle-fluid hybrid model was developed for simulation of the kinetic MHD instabilities. The self-generated radial electric field derived by the Reynolds stress and its effect on transport have been studied to contribute to understanding of improvement of the confinement.

R&D of fusion reactor technology has been focused on the ITER/EDA-related area. Major highlights in FY1997 are as follows.

Winding and heat treatment of Nb_3Sn conductors of all eight layers for the outer module of ITER CS model coil have been successfully completed and the assembling technology for the model coil has been developed. Production of 46 kA cable of Nb_3Al strands for the insert coil was also completed.

Fabrication of two full-scale 1/40-sector models of the ITER vacuum vessel was completed in the end of September 1997. The cross section is D-shape of 15 m high and 9 m wide. Both sectors, each of which uses different fabrication procedures and welding techniques, satisfy a dimensional accuracy of within ±3 mm. Hot isostatic pressing (HIP) technology was developed to fabricate a prototype mock-up of ITER shield blanket modules. Regarding the development of ITER divertor, full scale mock-ups of the vertical plates and the wings were successfully fabricated using newly developed bonding technology. The mock-ups were subjected to thermal cycle tests under an ITER steady-state heat load condition of 5 MW/m^2. The tests prove that all the mock-ups can endure the heat load for a repetition of 10^3 cycles without any damages. As to the development of the ITER blanket handling system, performance tests of the full-scale vehicle system was started for demonstration of remote replacement of 4 ton blanket modules.

A stable negative hydrogen ion beam of 25 mA has been successfully accelerated to 1 MeV with a five-staged accelerator. Development of a gyrotoron has progressed to deliver a maximum energy of 520 kW for 5 sec at 170 GHz with a diamond disk window. A large caisson of 12 m^3 was installed in a room of TPL to investigate tritium behavior released into confinement system and environment.

In the fusion reactor design, the DREAM design activity was focused on the prototype reactor. In the area of safety research, safety evaluation code development, LOVA and ICE experiments using small scale models, and the study of tokamak dust removal methods were also carried out.

The Final Design Report (FDR) of ITER was issued by the Director in December 1997. After the review by the Technical Advisory Committee (TAC) in January 1998, the FDR was presented to the ITER Council at its 13th Meeting held in February 1998. The FDR is composed of various technical documents on the detailed design of plasma parameters, tokamak components, plant system and the tokamak building. The major results of safety analyses described in the Non-site Specific Safety Report (NSSR) -2 was also included in the FDR. The technical review of the FDR is being conducted by the four Parties. The Japanese Home Team contributes to the design progress in the various fields through the conduction of design tasks in close collaboration with the Joint Central Team (JCT). The JCT member built up to 161 including 46 Japanese members as of December 1997.

Keywords: Fusion Research, JAERI, JT-60, JFT-2M, DIII-D, Plasma Physics, Fusion Engineering, ITER, EDA, Fusion Reactor Design, Annual Report

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I. JT-60 PROGRAM

Objectives of the JT-60 project are to contribute the design of Experimental Reactor (ITER) and to establish the physics basis for a steady state tokamak fusion reactor like SSTR. The previous open divertor was modified to a W-shaped divertor with pumps from February to May in 1997. The aim of this modification is to investigate effects of divertor geometry and control on divertor functions and to realize radiative divertor compatible with good confinement simultaneously. The W-shaped divertor is characterized by inclined targets and a dome in the private flux region and pumping from the inboard side in the private flux region, which has never been found in other tokamaks. Therefore, divertor performance obtained in this divertor will have strong influence on the determination of divertor structure of future tokamaks like ITER.

The JT-60U experiments in 1997 focused mainly on the steady-state tokamak research with new divertor and the negative ion based neutral beam (NNB) in addition to the existing profile and shape control techniques developed in JT-60U. The research on divertor physics was accelerated under the new divertor system with many of fine diagnostics: Detachment characteristics, pumping control, impurity control, recycling characteristics, etc. In the steady (5s) helium pumping experiment using the core fueling helium beams to model the helium ash, τ_{He}^*/τ_E~4 satisfying the ITER requirement was obtained.

The main purpose of confinement and stability studies in 1997 was to improve steadiness of high confinement plasmas with the new divertor. Since a long heating time with 203MJ of the total heating energy became possible without harmful increase in impurity and particle recycling, the DT equivalent fusion gain Q_{DT}^{eq}~0.1 was sustained for 9 sec in a ELMy H-mode discharge. The progress in the high confinement reversed shear operation was demonstrated by a quasi-steady sustainment of the internal transport barrier with an ELMy H-mode edge. Researches progressed also for the formation conditions of the internal and the surface transport barriers in the high-βp mode, the reversed shear mode and the H-mode.

Toward the advanced feedback controls of multiple parameters, the JT-60U started new feedback controls of central line density and divertor neutral gas pressure in addition to the existing controls of off-axis line density, radiation power and neutron production rate. Characteristics of halo current during disruptions was also studied. Optimization of NNB operation progressed steadily and injection power increased up to 5.2 MW. The NNB-driven current was identified directly from the internal magnetic measurement and driven current profile was confirmed to be consistent with the ACCOME calculation. The current profile control with LHCD successfully sustained the internal transport barrier in reversed shear plasmas. Continuous TAE modes were observed with NNB for the first time as beam-driven TAE modes.

1. Operation and Machine Improvements

In FY 1997, a total of 2,197 pulses were run during the period of 9-cycle operations and wall conditioning. The total number of shots carried out for the past thirteen years amounts to 25,142 as shown in Fig.I.1-1.

Modification from open divertor to W-shaped pumped divertor was completed in May. Various maintenance works including inspection of high pressure gas facilities such as the NBI cryopump system were also performed in May. Following this shutdown, coil excitation tests were performed for confirming the integrity of the tokamak machine after modification of the W-shaped divertor. Stable W-shaped divertor discharges were obtained in the middle of June. In mid-July, boronization were conducted for wall conditioning. After that, campaign of the divertor studies and negative-shear experiments were started aiming at realization of steady-state operation with high performance.

Annual maintenance of the JT-60 facilities were performed from November through December in 1997. The operation restarted late in January 1998. After boronization conducted in the middle of March, discharge optimization was started for the succeeding high Q_{DT} experiments. In spite of the operation after modification of the divertor, JT-60 was satisfactorily operated throughout the year. The database obtained in the operation and maintenance was arranged and made useful for maintenance plan and measures for the aged deterioration of the facilities. In particular, an overall revision of the operation manuals for the JT-60 facilities were made for ensuring safety in the operation.

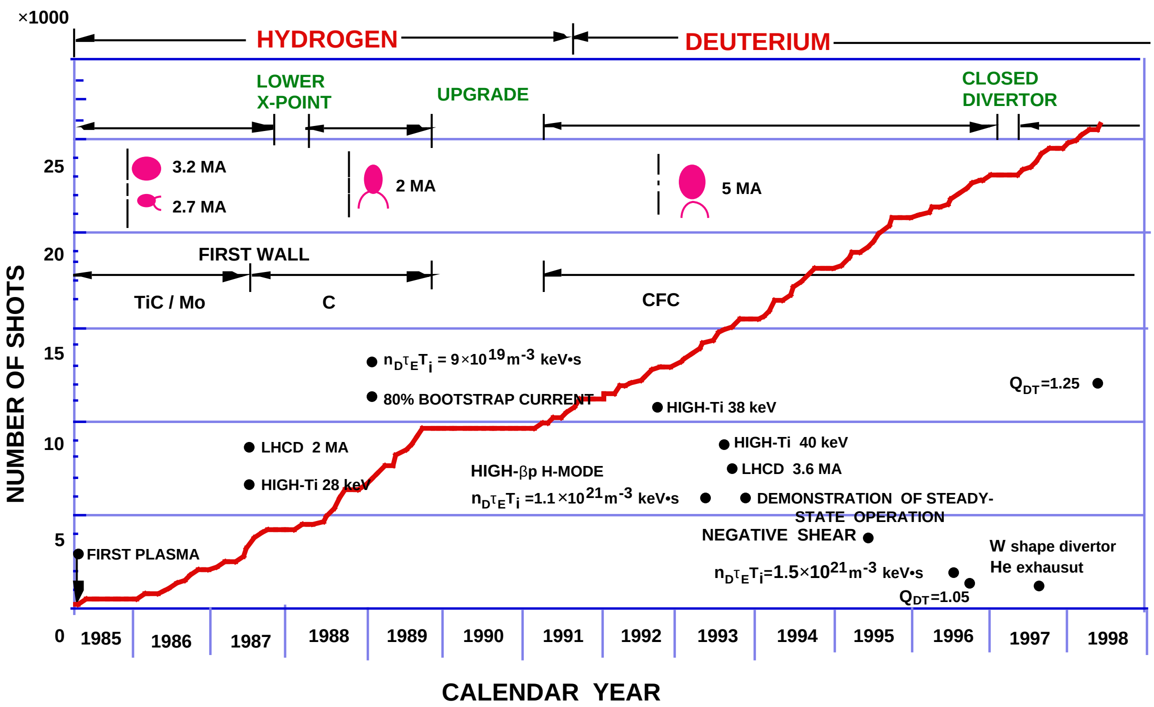

Fig.I.1-1 Progress in JT-60 operation.

1.1 Tokamak Machine

1.1.1 Toroidal Field Coil (TFC) and Poloidal Field Coil (PFC)

Since the occurrence of water leakage of cooling pipes for toroidal magnetic field coils (TFC) in 1992 (TC-9) and 1995 (TC-14), the coil layers with water leakage have been operated without water cooling. The cause for the water leakage was identified as some cracks which were found by a fiberscope observation system. Hence, in every maintenance period, the inside of the cooling pipes with cracks have been investigated by the fiberscope observation system [1.1-1] and airtight testing has been carried out for all the cooling pipes of TFC. By these examinations, it has been confirmed so far that there is no further growth of the cracks and no new cracks. On the other hand, as a protection system of TFC, a short circuit detection system has also been developed. This system aims to find precursor events as early as possible before occurrence of a short circuit between coil layers which finally lead to damage to the TFC. The configuration of the system is shown in Fig. I.1.1-1. This system consists of a Rogowskii coil and a set of six magnetic probes arranged around a TFC. The Rogowskii coil detects the short circuit phenomenon and the magnetic probes complement the Rogowskii coil data. With a successful result of preliminary tests on S/N ratio of the system, construction of this system has started on a full scale. In parallel with this development and taking into account that cracks might occur in neighboring coil layers in the future, influence of Fluorinert (Fluorocarbon) on the TFC construction materials have been examined as an alternative coolant for the TFC with cracks. This is because cooling of these coil layer cannot be expected without some coolant in this situation. The examination of results is favorable for usage of Fluorinert. Corrosion due to Fluorinert, which is a key for long term use as a coolant, was found to be permissible.

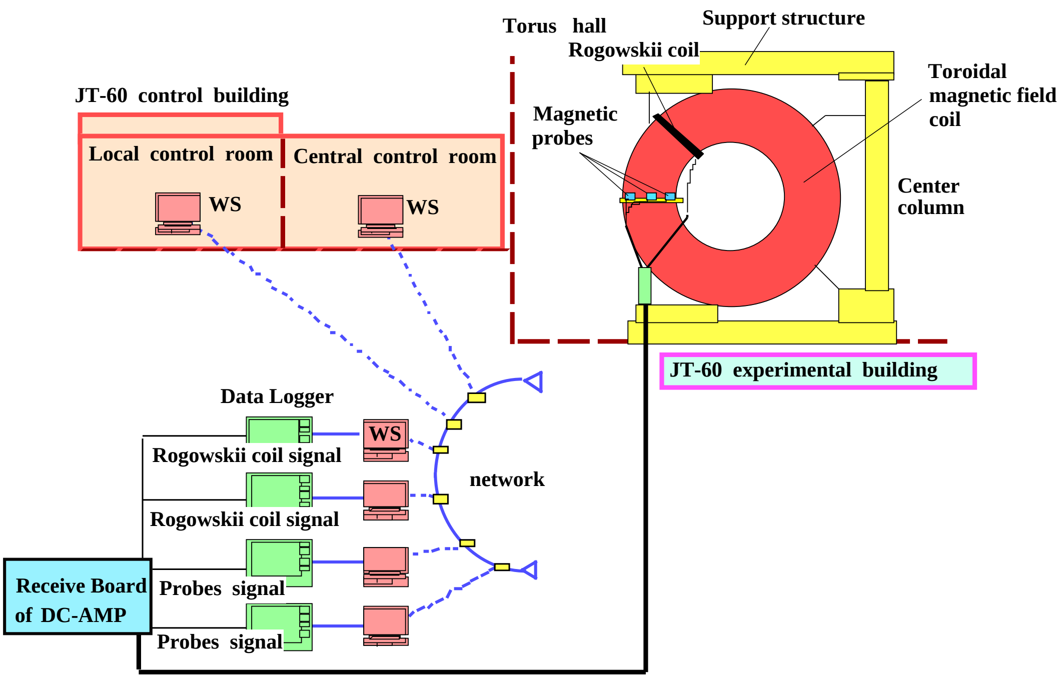

Fig. I.1.1-1 Short circuit detection system

In long pulse high triangularity operations, there is a risk that temperature of the VT-coil for controlling triangularity of the plasma may rise up to higher than the design value. For safe operation of this coil, interlock system with an optical fiber thermometer was installed in the VT-coil feeder.

1.1.2 In-vessel Inspection

The divertor modification was completed in May 1997. The first experimental campaign was started in June and ended in October. During this experimental campaign, operational parameters of JT-60U with the W-shaped divertor were : number of shots 1753, plasma current of 2.5 MA, NBI heating power of up to 25 MW, toroidal magnetic field of 4 T and number of disruption ~200. After this five months' operation, a routine inspection of the JT-60U vacuum vessel interior was conducted in November.

Severe erosions of outer divertor tiles and dome tiles were found in the routine inspection. These erosions were concentrated at the tile lip. Two outer dome tiles were broken in two, but still stayed on the dome plates. Fine cracks of outer dome tiles were also found. These show that heat flux higher than expected even reached the outer dome tile lip. The designed heat flux to the outer dome tiles was 1 MW/m^2 x 10 sec. These outer dome tiles were fabricated from isotropic graphite which is fragile at lower thermal shock compared with carbon fiber composite (CFC), because high heat flux was not expected on the outer dome tiles. So, the main cause for the tile break and cracks is probably thermal shock. These eroded and broken dome tiles were exchanged for CFC tiles with higher thermal shock resistance. The eroded divertor tiles were tapered according to each erosion.

Thick deposit of carbon was observed on the inner divertor tiles, but no severe erosion was found. Total amount of the deposit estimated by weight measurement was approximately 25g. In the previous divertor, recycling of the inner divertor was higher than the outer one, so exhaust for the divertor pump is located between inner divertor and dome to control recycling with the strong in-out asymmetry. The asymmetric deposit of the carbon must be related to the structure and/or the in-out recycling asymmetry.

Soundness of the W-shaped structure was almost demonstrated by this inspection. Plasma sprayed ceramic could be used as gas-seal between structures without extreme deformation. Insulation resistance of this plasma sprayed ceramic did not deteriorate. Other stainless and Inconel structures were also sound.

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1.2 Control system

The control system works in the JT-60 experiments according to the required schedule. The following developments were newly performed in this fiscal year to improve plasma control performances and operational efficiency.

(a) A precise long-time digital integrator, that can be applied to the 2000-s pulse discharge in ITER, has been completed, and 75 sets were built up for JT-60 magnetic measurements. (Refer to 1.2.1)

(b) A new advanced plasma control system has become ready for installation in JT-60. The computational time was greatly reduced to a hundredth of that of the former system. Design and development of a new discharge control system have been started. (Refer to 1.2.2)

(c) To improve the accuracy of plasma X-point height, a new function parametrization formula was introduced, and its new coefficients were derived as a result from the method of least-squares using the JT-60 new equilibrium data base. (Refer to 1.2.3)

(d) Corresponding to the modification of closed divertor, two feedback controls were installed: controls of neutral gas pressure and plasma electron density at the divertor region using an actuator of neutral gas feed into the divertor.

(e) A differential feedback control method was added to the existing neutron feedback control. This modification is a preparation toward high confinement experiments.

(f) A plasma equilibrium prediction code with human-friendly interfaces has been developed: This system calculates the full JT-60 equilibria throughout a pulse discharge to fit the given preprogrammed waveforms such as plasma total current, positions of the plasma geometrical center, etc. Using this tool, a physics operator could know if plasma configuration would be correct, coil currents would not exceed their heat capacity, etc.

For the maintenance of the control system, annual inspections were made on the computer system, control boards, and the signal processing system for plasma control during the shut-down period of November and December.

The workstation, that manages JT-60 discharge condition parameters (discharge conditions server), was superseded by a faster workstation with a large auxiliary memory. This increased the number of discharge conditions which were used in the past experiments. Human-friendliness for the screen layouts on the workstations were also appropriately improved in response to the requests from the JT-60 physics operators.

1.2.1 Development of a Precise Long-Time Digital Integrator for Magnetic Measurements

Magnetic field measurements are indispensable for control and diagnostics of a tokamak plasma. The existing methods for the measurement are (i) integration of time-derivative of magnetic field, and (ii) direct measurements of absolute magnetic field using rotating coils, Hall-element sensors, etc. The latter seems to be incompatible with 14-MeV neutron field, while the former has a problem of inevitable signal drift in an integrator, and thus it could not work for a long period of discharge (e.g., 2000 s for ITER). We chose the VF (voltage-to-frequency)-UDC (up-down counter) method for the development from the following view points: Its high accuracy is expected equivalent to analog integration. Wider dynamic range is allowed in a large digital accumulator, and drift can be compensated more precisely in a digital system.

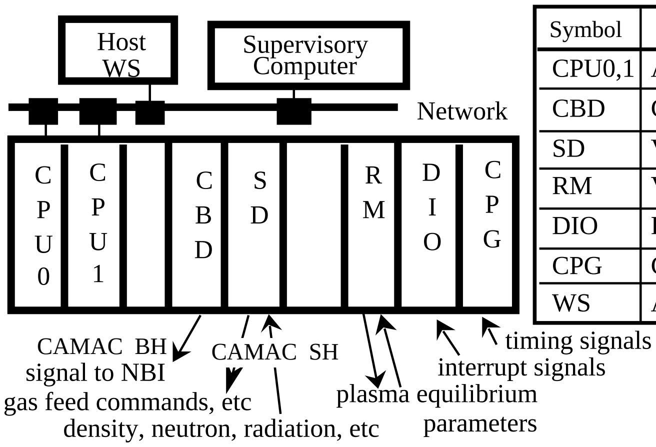
We built three trial models with new improvements added to reduce the integral errors in the VFC-UDC system arisen from the following causes: (1) VF conversion non-linearity, (2) production of deadbands, (3) slow base-line drift, and (4) stepped change at plasma instabilities. Finally, the third board showed good integration accuracy even for ITER with suppressing drift speed in the test environments of JT-60 experiments. [1.2.-1, 2]

On the basis of the technical experiences mentioned above, manufacturing of 75 VF converters have been completed. All the old VF converters will be superseded by the newly developed ones for JT-60 experiments in 1998. The outside view of a new VF converter is shown in Fig. I.1.2-1.

Fig. I.1.2-1 A new VF converter outside view.

1.2.2 Development of the JT-60 New Control Systems

Since requirements for modification of advanced plasma control and efficient discharge control increase, two control systems are being developed. Concerning a new plasma control system, we chose an Alpha-288 VME board (made by DEC. Ltd.) for main processors, and built



Symbol	Type	Function /Manufacturer
CPU0,1	AXPvme280	DEC 21066A, 288MHz, 64MB/ DEC
CBD	CBD8210	CAMAC Branch Driver / CES
SD	VSHD2992	CAMAC Serial Driver / HYTEC
RM	VMIVME5576	Reflective Memory (256kB) / VMIC
DIO	DSP8230	Digital Input (32 ch) / MTT
CPG	CPG241	Clock Pulse Generator / MTT
WS	Alpha WS 255	UNIX-WS alpha 255MHz / DEC

BH: Branch Highway SH: Serial Highway

Fig. I.1.2-2 Configuration of the plasma realtime control system.

a VME-bus system together with the I/O boards and a reflective memory module as shown in Fig. I.1.2-2. As a result from its performance test, this system can execute the realtime program within 0.1 ms. This period is approximately 100 times faster than the old mini-computer (10 ms). However, slow communication (1.4 ms) with the existing old subsystems through CAMAC highway is an obstacle for minimization of the total execution time.

This system will begin to work for JT-60 in May 1998, after several integration tests with the other subsystems.

Concerning a new discharge control system, we performed system design of the hardware configuration. According to the current design, this system is composed of two parts: one is a supervisor for compilation of discharge condition files and acquisition of result data from all the subsystems (DC/RD-SV), and the other is a supervisor for discharge sequential control (SQ-SV).

DC/RD-SV, as a master for discharge condition files before discharge, distributes the appropriate conditions to the corresponding subsystems after compilation procedures. After discharge, DC/RD-SV works as a master for result data acquisition. This receives result data from all subsystems, and builds an intermediate file before transferring it to the JT-60 database production server.

SQ-SV manages all the actions and events occurred in the JT-60 systems according to the discharge sequence by sending the command messages and receiving the replies. This part will be composed of VME modules due to the required fast on-line communications. DC/RD and SQ-SV's will be expected to come into operation in 1999.

1.2.3 A New Function Parametrization Formula

A function parametrization (FP) method has been adopted for the real-time control of JT-60U plasma position and shape, where sets of linear coefficients in the FP formulas are determined through the method of least squares (LS) on the numerically-prepared equilibrium database. On modification of the divertor structure, the number of linear coefficients was increased from 7 to 33 to improve the control accuracy.

Although the numerical examination using the database had shown a good accuracy (standard deviation σ ~2 cm), the experimental comparison of the FP method and the equilibrium analysis showed considerably large discrepancies of large offset bias and large standard deviation in X-point position for high-elongated plasmas. Investigation of these problems has determined that the following items could be the causes: an ill-posedness in the FP formula coefficients and negligence of ohmic-heating coil (OH-coil) effects.

It was found out that strongly correlated sensors or very insensitive sensors involved in the LS analysis caused the ill-posed problem that small amount of actual sensor noise or error could make large difference from the numerical results on the ideal database. We excluded probes of concern and recalculated the coefficient. As a result, the new coefficients set suppressed that the offset bias completely.

A lot of experimental results suggested that the effects of OH-coil field have strong influence on the error. We have then developed a new FP formula for X-point position $X_{\mathrm{P}}^{\mathrm{FPM}}$ that takes the effects of the OH-coil current (I_{OH}) as well as the divertor-coil current (I_{D}) as follows,

$$
\begin{aligned}
X_{\mathrm{P}}^{\mathrm{FPM}} = {} & C_0 + C_1 \frac{I_{\mathrm{D}}}{I_{\mathrm{P}}} + C_2 \frac{I_{\mathrm{OH}}}{I_{\mathrm{P}}} + \sum_{i=1}^{N_\omega} \left(C_{\omega i} + D_{\omega i} \frac{I_{\mathrm{D}}}{I_{\mathrm{P}}} + E_{\omega i} \frac{I_{\mathrm{OH}}}{I_{\mathrm{P}}} + F_{\omega i} \frac{I_{\mathrm{D}} I_{\mathrm{OH}}}{I_{\mathrm{P}}^2} + G_{\omega i} \frac{I_{\mathrm{D}}^2}{I_{\mathrm{P}}^2} \right) \frac{B_{\omega i}}{I_{\mathrm{P}}} \\
& + \sum_{i=1}^{N_\rho} \left(C_{\rho i} + D_{\rho i} \frac{I_{\mathrm{D}}}{I_{\mathrm{P}}} + E_{\rho i} \frac{I_{\mathrm{OH}}}{I_{\mathrm{P}}} + F_{\rho i} \frac{I_{\mathrm{D}} I_{\mathrm{OH}}}{I_{\mathrm{P}}^2} + G_{\rho i} \frac{I_{\mathrm{D}}^2}{I_{\mathrm{P}}^2} \right) \frac{B_{\rho i}}{I_{\mathrm{P}}},
\end{aligned}
\tag{1.2.3-1}
$$

where I_{P} is the plasma current, $B_{\omega i}$ and $B_{\rho i}$ tangential and normal components of magnetic fields at the *i*-th probe positions, N_ω and N_ρ the numbers of probes, and the C_1, C_2, $E_{\omega i}$, $F_{\omega i}$, $G_{\omega i}$, $E_{\rho i}$, $F_{\rho i}$ and $G_{\rho i}$ the coefficients determined by the method of LS.

The application of the new FP formula to the actual experiments are shown in Fig. I.1.2-3, where X-point positions are certainly detected within 2 cm, as expected in the numerical analysis. [1.2-3].

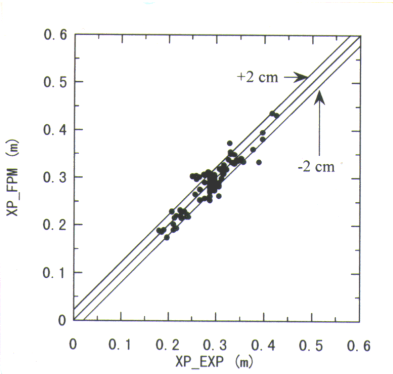


Fig. I.1.2-3 Correlation of X-point positions, $X_{\mathrm{P-FPM}}$ and $X_{\mathrm{P-EXP}}$. ($X_{\mathrm{P-FPM}}$: the FP formula on experimental data. $X_{\mathrm{P-EXP}}$: the equilibrium analysis).

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1.3 Power Supply System

The JT-60 power supplies were operated on schedule without any serious problems throughout this physical year, though fourteen years have passed since they were completed. In November and December 1997, annual maintenance of the power supplies were performed according to the regulations for electric equipment in Naka Fusion Research Establishment. The following maintenance works were conducted as the measures against aged deterioration of the devices: (1) Insulating supports for thyristor stacks in converters of the vertical filed coil power supply were replaced. The surface of the supports made by FRP had chemically changed and the electrical insulation resistance had been extremely decreased. (2) All of the insulated-signal transducers for the feedback control of the inverter unit in the un-interruptible power supply were replaced with new ones for preventive maintenance. (3) Water was found on the floor near the DC feeders from the thyristor converters in the poloidal field power supplies due to the leaks in the wall and roof of the JT-60 rectifier building at the typhoon in June. Hence, the repair work was done for the rectifier building.

1.3.1 Replacement of a New Coil Current Control System in the Toroidal Filed Power Supply

The toroidal field coil power supply (TFPS) is composed of six diode rectifier banks and a motor-generator(MG) with a large flywheel. Four banks of the rectifiers are directly connected to a commercial power grid, and the rest of them are powered from the MG. The toroidal field coil current must be controlled through the output voltage of the generator. Therefore the field control system of the generator is very important for the TFPS. However it became very difficult to maintain the integrity of the original control system, because 15 years have passed since the fabrication. Then we decided to replace the control system including I/Os to the new one which is based on the VME-standard (see Fig.I.1.3-1). The recent microprocessor is so powerful as to make the control system multi-functional. The limit function, coil fault detection, and real-time data display functions are introduced in the new system. This system may greatly enhance the reliability of the toroidal coil current control, and also make it possible to improve the control performance. Several tests using the analog simulator were carried out as a linkage performance test. After the completion of dummy load test, the original control system will be switched to the new one.

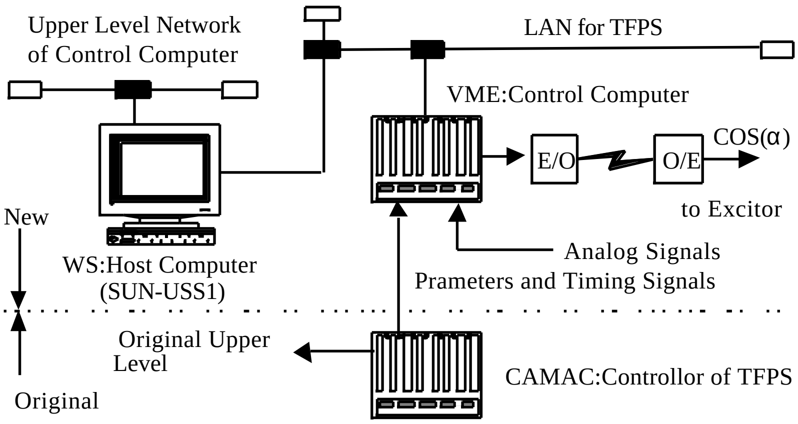

Fig.I.1.3-1. A new control system of the toroidal coil current.

1.3.2 Development of an IGBT Current-Driven PWM Converter

A 100-kW-class current-type PWM (pulse width modulation) converter based IGBTs (insulated-gate bipolar transistors) was developed and the feasibility of its application to a large magnet power supply for nuclear fusion device was investigated. Table I.1.3-1 shows the ratings of developed IGBT converter. Some adjustments for the circuit parameters were needed, but the target values of the rated performance were successfully achieved. Through the tests of the converter, the following issues to be solved were newly found: transient high voltages of a *LC* filter, distortion of an AC source current for low output voltage operation, and decrease in power factor owing to large current operation [1.3-2]. We are planning to optimize the feedback gains in the control of the reactive and output current in order to realize the rapid step response.

Table I.1.3-1 Ratings of the IGBT converter.

capacity	100 kVA
maximum voltage	200 V
maximum current	500 A
IGBT	1S–4P–6Arm
LC filter	2 mH–500 F
load	4mH–17mW

1.3.3 Development of a Water-cooled VCB for a Superconducting Magnet Power Supply

We started to develop a current interrupter which can carry a large current in steady state. The purpose of this development is to offer the key component of a quench protection circuit in superconducting magnet power supplies for fusion devices. As a candidate of the current interrupter, a water-cooled VCB was newly designed and its model test was conducted.

The target values of its performance were determined as follows: (1) continuous current-carrying capacity of 25 kA or more, and (2) current interruption rating of 50 kA or more. Since thermally critical parts of the VCB are contacting surfaces of its electrodes, a key issue of the design is how to remove the heat generated on the surfaces in the electrodes from the vacuum area. For the heat removal with good efficiency, the VCB was designed to possess a short fixed rod with a large coil outside the vacuum area and a fat movable rod where a water-cooling channel can be bored. Thus the new VCB has an up-down asymmetrical structure having the coil that provides co-axial magnetic field for stabilizing the current interruption property (Fig.I.1.3-2). Thermal characteristics of the VCB were analyzed by computer simulation. In addition, a model of the VCB was fabricated and tested to evaluate the characteristics. At the test of the model VCB, it was proved that the water-cooled VCB with a current-carrying capability of about 18 kA is feasible [1.3-3].

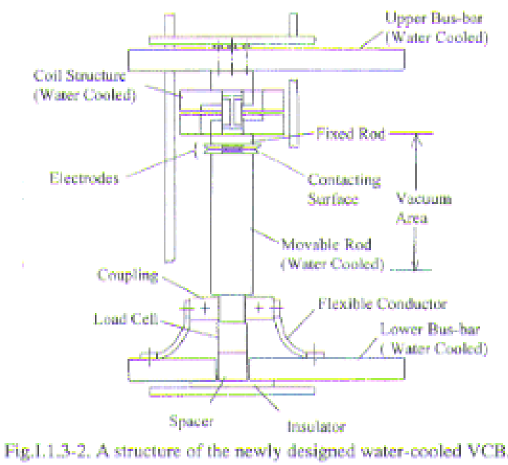

Fig.I.1.3-2. A structure of the newly designed water-cooled VCB.

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1.4 Neutral Beam Injection System

1.4.1 Positive-ion Based NBI System

Three beamline units out of fourteen of the positive-ion based NBI system have been modified as an exclusive use of gas pumping for a newly fabricated W-shape divertor in JT-60. The divertor experiments using the beamline cryopumps started in summer 1997. Neutral gas pumping at the divertor region with the cryopumps has been clarified to increase the JT-60 plasma performance. In a simulation experiment of helium ash exhaust, the divertor pumping for a mixed gas of deuterium and helium has been demonstrated with the NBI cryo-sorption pumps which is made through condensing argon gas onto the liquid helium cooled cryo-panels.

The neutral beam injection using the rest of the beamline (11 units) has been conducted with an injection power range of 20-30MW at a beam energy of around 90keV. A total injection power of 203MJ has been obtained in a long pulse beam injection near 10 sec at a moderate injection power of around 22MW, and thus quasi-steady state H-mode plasma near 10 sec has been achieved.

1.4.2 Negative-ion Based NBI System

The beam injection experiment into JT-60 with the negative-ion based NBI system (N-NBI) has been conducted, augmenting the beam power gradually, since the beam operation started in 1996.

Many problems have been experienced in the high voltage beam operation. Most of the problems occurred in the ion source and high voltage power supply were caused by a surge energy at the moment of the ion source breakdown. These have been solved step by step through altering the components of the power supply hardware and remodeling of the control system. Improving the ion source and power supply components, the beam power has increased gradually, and a neutral beam injection power, so far, has

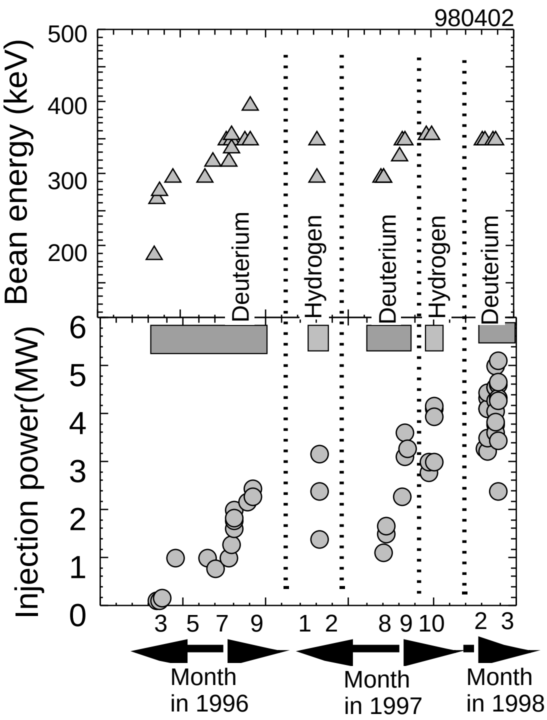

Fig. I.1.4-1 Time evolution of injection power and beam energy.

reached 5.2 MW at 350 keV with deuterium and 4.2 MW at 360 keV with hydrogen as shown in Fig. I.1.4-1. A negative-ion beam power per one ion source has reached 380 keV, 14.3 A with deuterium and 360 keV, 18.5 A with hydrogen.

The plasma characteristics concerning a NBI current drive efficiency and neutron yield from plasma with the NBI beam injection, so far, have been confirmed to be agreed with a theoretical prediction. The efforts for increasing beam power and energy will be continued aiming at the rated injection power of 10 MW for establishing the technical and physical bases of the ITER.

1.5 Radio-frequency Heating System

1.5.1 ICRF System

The ion cyclotron range of frequencies (ICRF) system for JT-60 was operated well at 102 MHz in FY 1997. Prior to the operation, the stub tuners were modified in April 1997 in order to improve their voltage stand-off capability [1.5-1], because we were afraid that the coupling of the antenna and the plasma would be degraded on the modification from an open divertor to a W-shaped divertor. High power can be coupled to the plasma even with degraded antenna-plasma coupling when the higher voltage stand-off is achieved at the antenna. Then we had to improve the voltage stand-off capability of the coupler system which consists of stub tuners, high power phase shifters, coaxial lines and antennas.

After careful optimization of the plasma shape for the ICRF experiment, we obtained similar antenna-plasma coupling with the W-shaped divertor as one with the open divertor. Plasma shapes for ICRF coupling with open and closed divertor are shown in Fig. I.1.5-1 and Fig. I.1.5-2, respectively. The gap, δ_0, between the antenna and separatrix was tried to kept constant in front of the antenna by means of adjusting a separatrix curvature and a vertical position of the plasma. We paid attention to keep at least 3 cm of the gap at the outer baffle plate of W-shaped divertor in order to reduce the plasma-wall interaction there, by means of adjusting the height of the X-point. Typical parameters for ICRF coupling were the plasma current = 1.7 MA, the toroidal magnetic field on the axis = 3.34 T, the triangularity = 0.28, the plasma volume = 80 m^3 and the gap δ_0 = 6 cm - 15 cm.

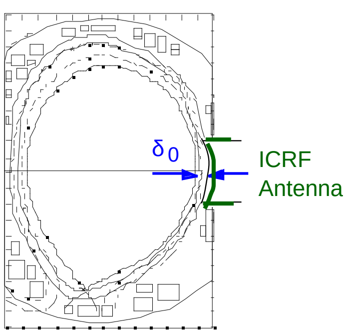

Fig. I.1.5-1 Configuration with open divertor for ICRF coupling

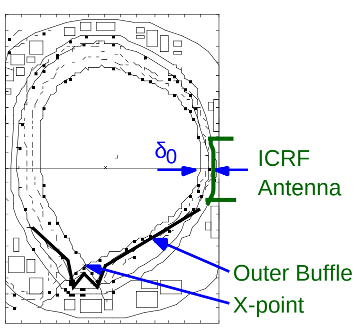

Fig. I.1.5-2 Configuration with W-shaped divertor for ICRF coupling

To evaluate the antenna - plasma coupling, the coupling resistance (R_C) is often used. R_C is defined as $R_C = 2PZ^2V_{max}^{-2}$, where P is the input power to the antenna, Z the characteristic impedance of the coaxial line in the antenna, and V_{max} the maximum RF voltage of the standing wave on the coaxial line in the antenna. When high R_C is obtained, high power can be coupled to the plasma with low RF voltage at the antenna. About 2.5 W of R_C was obtained with around 10 cm of δ_0. It is consistent to the result of the reciprocating probe measurement [1.5-2, 3] which shows that the gradient of the scrape off plasma density in the W-shaped divertor case is similar to that in the open divertor case. If the breakdown voltage of the antenna is 40 kV which was obtained in the antenna conditioning in vacuum, 6.4 MW will be coupled to the plasma with 10 cm of δ_0.

Coupled power of 1 MW for 1.5 sec and 5.1 MW for 50 ms were obtained after only 5 days' antenna conditioning after divertor modification, as shown in Fig. I.1.5-3. On September 22nd, the 6th's day of the antenna conditioning, 4 MW for 1.5 sec and 4.3 MW for 1 sec were achieved. Energy of 8.6 MJ was coupled to the plasma as a sum of three RF pulses in one plasma shot. After the antenna conditioning, ICRF power of 4 MW was routinely coupled to the plasma for the experiments on ICRF heating of negative magnetic shear plasmas [1.5-4] in September and October.

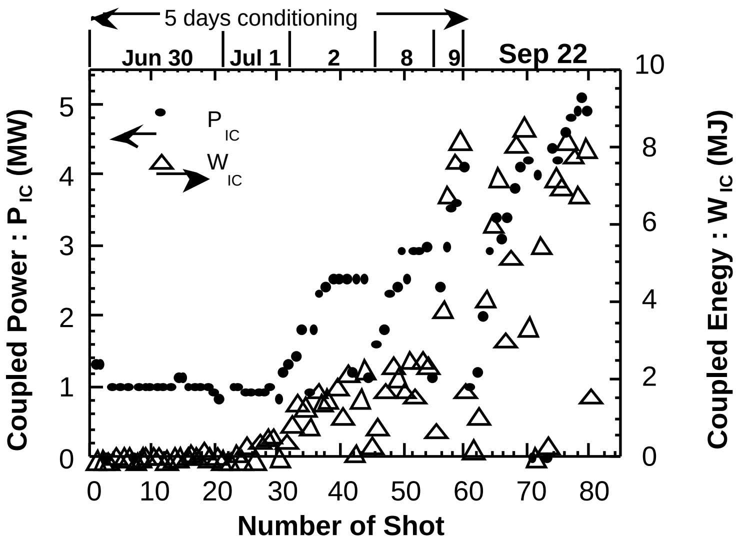

Fig. I.1.5-3 Progress of the coupled ICRF power and energy with W-shaped divertor.

1.5.2 LHRF System

The lower hybrid range of frequencies (LHRF) system in JT-60 was also operated with the W-shaped divertor plasma in FY 1997. At first, coupling properties of LH antennas were investigated. The coupling was good for the LH antenna C located at the upper inclined port of P-11, even with the W-shaped divertor plasma. It is a reason that plasma parameter in the scrape off layer in front of the antenna mouth was not changed after modification from the open divertor to the W-shaped divertor, due to the antenna mouth being far from divertor section. On the other hand, the distance between separatrix and the first wall around the LH antennas A and B located at horizontal port P-18 became shorter as ~8 cm for obtainment of the same low reflection coefficient. The reflection coefficient of the lower-side LH antenna A is plotted as a function of the distance named as δ_{344} in Fig. I.1.5-4. As shown in the figure, the plasma should be closer to the LH antenna for LHRF experiments under good coupling with the W-shaped divertor in comparison with open divertor. This leads to a limitation of plasma configuration and/or experimental conditions. Then the usage of the lower-side LH antenna system is planned to be utilized for the localized current profile control system. Even though number of LH antennas decreased from three to two, the current profile control and current drive were also available with the W-shaped divertor plasma. Injection of LHRF power highly contributed to reversed magnetic shear experiments, referring to section I.2.4.

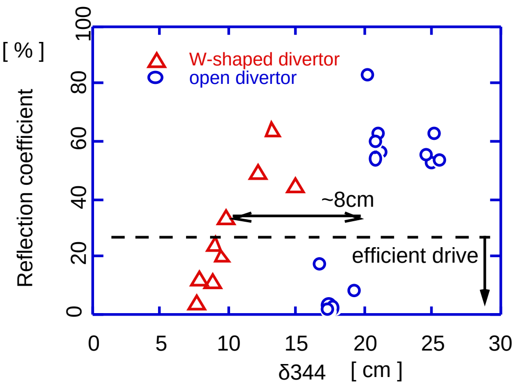

Fig. I.1.5-4 The reflection versus distance δ344.

It is important to establish a conditioning method for the LH antenna with a small number of shots as possible as we can. So, the power injection with pulse modulation was tried in order to avoid serious breakdowns in the LH antenna. This allowed effective conditioning by means of valuable plasma shots, because a small breakdown in the waveguide can not grow up within 1-10 msec and rf injection can continue in the same plasma shot. On the contrary, in the former conditioning shots, the rf power was stopped when a severe breakdown occurred. Moreover the pulse modulated injection can drive plasma current with the same efficiency as shown in Fig. I. 1.5-5, taking into account of the duty for the pulse modulation. This pulse modulated injection will be useful in the next generation tokamak, since the antenna mouth should be healthy without breakdowns.

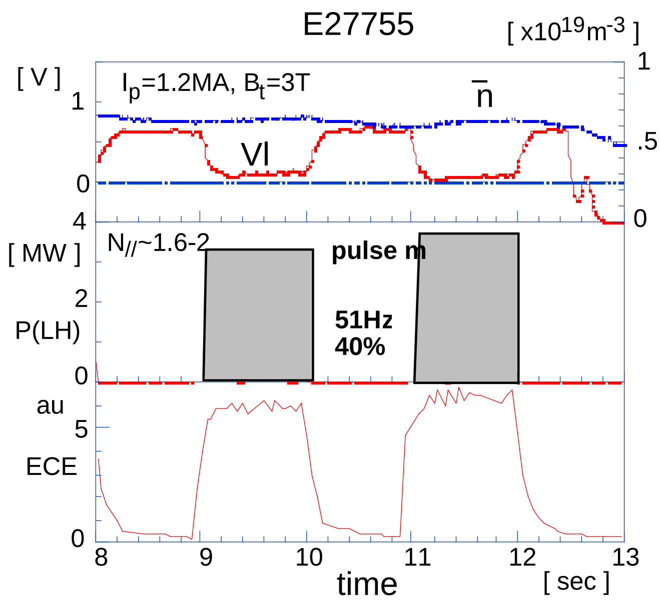

Fig. I.1.5-5 Typical shot in pulse modulation.

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1.6 Diagnostics System

New installations have been done for the following systems; infrared laser polarimeter, mm-wave Iinterferometer, Mach probe, In-vessel bolometer camera, fast response ionization gauge, core correlation reflectometer and CO_2 laser collective Thomson scattering.

1.6.1 Infrared Laser Polarimeter for Electron Density Measurement

An infrared laser polarimeter has been developed for electron density measurement in large tokamaks [1.6-1]. By using the infrared laser polarimeter, the first measurement of the tangential Faraday rotation of a CO_2 laser wave (wavelength ~10 μm) in a tokamak plasma has been successfully obtained in JT-60U [1.6-2], where the tangential Faraday rotation is approximately proportional to the product of electron density and the toroidal magnetic field.

1.6.2 Mm-wave Iinterferometer in Divertor Region [1.6-3]

A mm-wave interferometer has been developed for divertor diagnostics in JT-60U. Three lines of sight, which pass through the X-point horizontally, the inboard divertor and the outboard divertor were chosen. Two transmitter/receiver units with frequencies of 217 and 183 GHz were employed in order to eliminate the spurious vibration effect using a two color scheme. The two independent units were also arranged to enable two sight lines measurement without the vibration compensation. The measurements performed for several types of discharges indicated the feasibility of the system, and the rapid reduction of the electron density was first measured near the X-point at the transition of the confinement mode.

1.6.3 Mach Probe for the Plasma Flow in SOL

Multi-point measurements of temperature and density distributions in the SOL, i.e. at the midplane, near the x-point and at the divertor plates, were developed in the W-shaped divertor. In particular, Mach probes were installed at the midplane and near the x-point in order to evaluate the plasma flow and its direction. When the single null divertor was operated, it was found that the direction of the ion grad-B drift plays a critical role in determining the SOL flow direction. The plasma flows from the midplane to the x-point for the reversed field (the ion grad-B drift is directed away from the divertor), but it flows from the x-point to the midplane for the normal field direction, suggesting the flow reversal near the midplane. Here the particle source is very small compared to that near the divertor region for both cases. These observations show the existence of parallel ion convection at the SOL of the main plasma.

1.6.4 In-vessel Bolometer Camera for the Divertor Study [1.6-4]

Bolometer cameras were installed inside the W-shaped divertor chamber of the JT-60U. Each camera has a four-channel bolometer head of high temperature version using mica substrate.

Radiation profiles along the separatrix surface from x-point to the divertor tiles were measured with two cameras placed under the divertor dome. Other cameras view an x-point area from horizontal and vertical directions to observe the phenomena such as MARFE. Although the initial operations were successful in obtaining good quality and the time resolution of signals, the detectors were damaged during disruptions. Both thermal and electrical insulation of the camera are planned to be improved.

1.6.5 Fast Response Ionization Gauge for the Neutral Gas Pressure [1.6-5]

Fast response ionization gauges were installed to measure the neutral gas pressure profile at the divertor region, the pumping duct, and the main plasma edge. The gauge and its controller were developed by the ASDEX team. Its dominant advantage is that the sensor head can be applied in strong magnetic fields, so that the sensor head can be located very close to the plasma, which contributes to the fast time response. In front of the sensor head a chevron is placed in order to view the plasma indirectly and to provide thermalization of particles. The time response including the chevron is estimated to be about 3-4 ms, which is two orders of magnitude faster than conventional pressure gauges used in the vacuum vessel of JT-60U.

1.6.6 Core Correlation Reflectometer

A core correlation reflectometer system has been developed under the collaboration between JAERI and PPPL. The polarization of the launched wave is X-mode and the frequencies of the launched waves are 115, 130, 122.5±Δf GHz. The value of Δf can be changed from 2 to 18 GHz, so that the cut-off layers can be scanned through the region of the internal transport barrier (ITB) for the reversed shear plasma (see Section 2.1). Furthermore the frequencies can be scanned rapidly within a single shot, allowing radial correlation measurements of the fluctuations. The core correlation reflectometer system will help us to understand the physics of ITB plasmas. This system will work from the summer in 1998.

1.6.7 CO_2 Laser Collective Thomson Scattering [1.6-6]

A Collective Thomson Scattering (CTS) is nominated as a candidate in ITER for the measurement of bulk ion temperature and energy distribution of high energy α-particles. The CTS system using a pulsed CO_2 laser with small scattering angle ($\approx 0.5^\circ$) has been developed to measure ion temperature in JT-60U. Estimation of scattered power spectrum shows a reasonable signal to noise ratio for the ion temperature measurement. A pulsed CO_2 laser system (wave length: 10.6 μm, energy: 10 J, pulse length: 1 μs, repetition rate: 0.5 Hz) and a heterodyne receiver system with a hot CO_2 absorber cell as a stray light notch filter, which are developing in Oak Ridge National Laboratory, will be installed in a diagnostic room in 1999.

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1.7 Data Analysis System

1.7.1 Data Analysis Tools, Database and Computer System

Developments and improvements have been carried out for data analysis tools of the JT-60 analysis server. A new version of DAISY (DAta Illustration SYstem) has been developed, which has a new graphical user-interface using the X-Window technology. After the divertor modification of JT-60U, the plasma-boundary identification code, FBI, and the MHD equilibrium analysis code, SELENE, have been revised to incorporate the new divertor configuration. The software showing a time slice of experimental data, SLICE, has also been improved to incorporate new diagnostics and has been added as a function of file output. A new statistical analysis tool, SANDER (Statistical ANalysis and Database Exploring Routine), has been developed using the statistical analysis package, SAS. This tool can use both the experimental database on the JT-60 database server and the DARTS (DAtabase ReTrieval System) database on the JT-60 analysis server.

The JT-60 experimental database has enriched the content. Appropriately for the update of diagnostic systems, such as CO_2 polarimeter, heterodyne radiometer, divertor bolometer, and neutral gas pressure diagnostics, these diagnostic data have been added to the experimental database. Plasma equilibrium data by the new FAME, which is described in Sec. 1.7.2, have increased by twice in kinds and 10 times in time points. Calculated data by new RTP (real time processor) and fast sampling ($\geq 5\mu s$) data have also been added to the database.

Some subsystems and programs of JT-60 data processing system have been improved according to the demands of plasma diagnostic systems. Software for the acquisition system of the massive data, FDS (Fast VME Data acquisition System), has been developed to handle the increasing data on ISP (Inter Shot Processor) speedily. New RTP has been utilized to process the increasing amount of input-data and to realize an advanced feedback control. The CPU and the analog-to-digital converter have been improved and are about 10 and 5 times faster than the former ones, respectively.

ISP has an automatic data storage system of the cartridge tape library, CTL. It contains ~1300 cartridges and stores ~300GB. At present, all JT-60 data of ~4TB are kept in these cartridges. But the CTL does not have sufficient capacity to handle increasing amount of JT-60 data. The amount of data in a cartridge of ~250MB is too small to handle data in the JT-60 data processing system. The reliability of magnetic tape media is also a problem. Therefore, a new data-archiver with another media of more data storage capacity and reliability has been utilized. It is a UNIX file server with ~100GB RAID disks and ~900GB MO (magneto-optical disk) auto-exchangers. This archival capacity corresponds to the data of about one-year JT-60 experiment shots at a present level of data.

1.7.2 FAME System

The original system of FAME (Fast Analyzer for MHD Equilibrium) was developed in 1993 to provide about 130 MHD equilibria in time series which are enough for the non-stationary analysis of the experimental data of JT-60U within a shot interval. The new system, FAME-II, with a high processing speed using IBM RS/6000 SP has been utilized in FY 1997, succeeding the original system. The system is a MIMD type small scaled parallel computers with 7 CPUs and the maximum theoretical speed is 3.42 GFLOPS. The SELENE and FBI codes are tuned up taking the parallel processing into consideration as well as the original system. Consequently, the computational performance of the new FAME system becomes more than 3 times faster than the original system. The new system also has the file server system with the large capacity of the data storage of 50 GB.

Efforts of utility development and update have been concentrated on more effective use of the new FAME system. An equilibrium animating system has been developed on a workstation arranged in the central control room. The system can provide animations of MHD equilibrium analyzed by the FAME, incorporated with SLICE. In order to display typical equilibrium data such as an ellipticity, an internal inductance, and so on, as functions of time with other experimental data, the new FAME system recalculates equilibria at an interval of 10 ms during off-experimental hours in night and transfers the results to JT-60 database server.

1.7.3 Data Link System and Remote Participation in JT-60 Experiments

The remote participation in JT-60 experiments from PPPL was successfully carried out during reverse shear experiments in September, by utilizing the Data Link System and the video conferencing systems. Participants from both JAERI and PPPL jointly analyzed and discussed the JT-60 data together.

In September, the JAERI- DOE overseas line was upgraded from 128 kbps to 768 kbps using state-of-the-art frame-relay technology. It is beneficial to the remote analysis of JT-60 data from PPPL to use the Data Link System. The Data Link System provides standard JT-60 data analysis tools; DAISY, FBI, EQREAD (MHD equilibrium display code), and SLICE. All tools have been improved to keep the security. Seven projects have been approved as remote collaboration under the Cooperation among the Three Large Tokamak Facilities. These seven projects cover a wide range of research topics and the total number of participants amounts to about one hundred.

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2. Experimental Results and Analysis

2.1 Reversed Shear Experiments

2.1.1 Stability Improvement in Reversed Shear Plasmas with an H-mode Edge

The high performance reversed shear discharges with an internal transport barrier (ITB) encountered a beta collapse when q_{min} decreased to 2, which restricted the fusion performance and the duration of high confinement. The beta limit was β_N~2 (β_N is the normalized beta) at q_{min}~2, and the low-q_{min} region below q_{min} = 1.7 could not be reached [2.1-1]. It was observed that fluctuations of electron temperature grew explosively in the ITB region with a very fast growth time of order ~10 μs just before the collapse [2.1-2]. These observations of the beta limit and instability growth time well agree with calculated values for low-n kink-ballooning modes by using the ERATO-J code [2.1-3, 4]. Since this mode is destabilized by a large pressure gradient in the ITB layer, pressure profile broadening with combining the ITB with the H-mode in the peripheral plasma region was attempted in high triangularity discharges for a toroidal field of 3.5 T. The beam power during the current ramp-up was reduced to prevent the development of a strong ITB (or a steep pressure gradient), because strong ITBs appeared to make the H-mode transition difficult [2.1-1]. Then a high power heating at the current flat-top triggered the H-mode transition and also generated an ITB. The stability in the low-q_{min} region was successfully improved for the H-mode edge discharges, and a high β_N value of 2.3 has been achieved with q_{min} = 1.5.

2.1.2 Sustainment of internal transport barrier

To realize a steady state operation of high confinement reversed shear discharges, simultaneous sustainment of the ITB and the current profile are required. In JT-60U quasi-steady sustainment of ITB with an ELMy H-mode edge was obtained thanks to the enhanced stability in H-mode edge discharges for B_t=3.5 T (see Fig.I.2.1-1 [2.1-5]). Though the current profile was not kept stationary due to the lack of the active non-inductive current drive source, significant advance towards the steady state operation has been obtained as shown in Fig.I.2.1-1. Improved

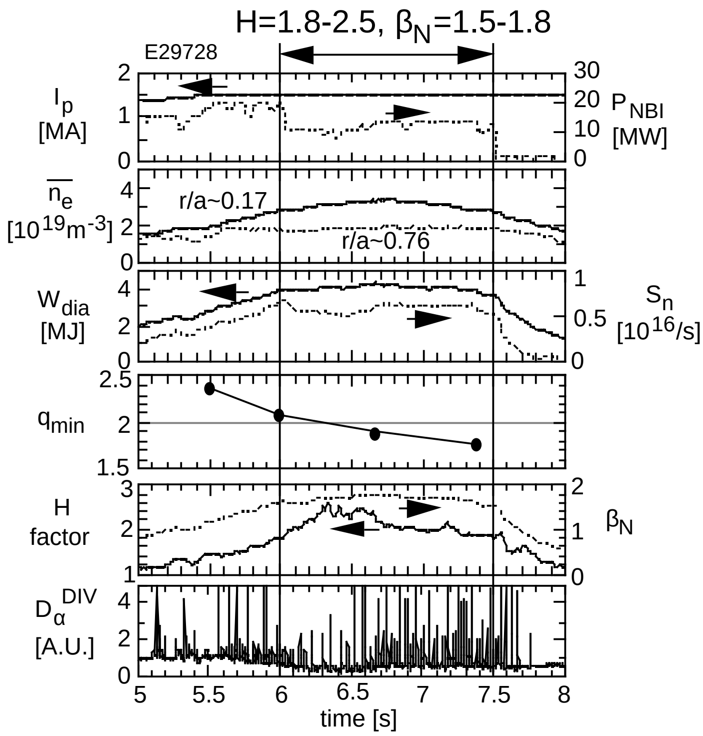

Fig. I.2.1-1. Waveforms of a reversed shear discharge with an ELMy H-mode edge where an ITB was sustained for 1.5 s with an H-factor of 1.8-2.5 and β_N of 1.5-1.8.

confinement with an H-factor of 1.8-2.5 and β_N of 1.5-1.8 is sustained for 1.5 s (or 4-5 times τ_E) in a high triangularity reversed shear discharge (I_p = 1.5 MA, q_{95} = 4.5, δ ~ 0.28). The ITB is established and developed sufficiently by the high power heating before t = 6 s as indicated by the rise of line averaged electron density $\bar{n}_e$ along the central chord (r/a ~ 0.17) , and the beam power is stepped down to avoid a collapse. Though the minimum q, q_{min}, continues to decrease, the reversed shear configuration and the ITB with steep gradients in n_e, T_e and T_i profiles are maintained until the end of neutral beam heating.

Long sustainment, reaching the time duration of 4.3 sec, of the ITB and the reversed shear configuration with an H-factor of ~ 1.7 and β_N of ~ 1.5 was also demonstrated using a feedback control of the beam power to maintain a fixed neutron emission rate.

2.1.3 Formation Condition of Internal Transport Barrier

The ITB, especially with a reduction in electron thermal transport, should be actively controlled in order to obtain steady-state burning plasmas. From this view point, onset conditions of the ITB were investigated by systematic scans of electron density, beam power and magnetic shear. Preliminary results have been obtained as follows.

(1) Under an almost fixed reversed magnetic shear profile, toroidal rotation shear seemed to alleviate the required heating power for the ITB formation.

(2) Even with similar electron density, heating power and toroidal rotation shear, a discharges with a reversed magnetic shear showed an ITB formation, while another discharge with a positive magnetic shear did not. (In different condition, there exists ITB accompanied by a positive magnetic shear.) This difference suggests that reversed magnetic shear mitigates the required conditions for the ITB onset.

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2.2 High β_p and High Triangularity Discharges

This section treats recent development of quasi-steady ELMing discharges with enhanced confinement, high-β stability and current drive capabilities, where increase in absolute values of fusion performance and sustainment time are emphasized in addition to the normalized parameters. After modification to the new W-shaped pumped divertor, a long heating time (9sec) of high power NB (20 -25MW) became possible without harmful increase in impurity and particle recycling. The total energy input reached 203MJ. In addition to it, optimization of the plasma shape, the pressure profile and selection of appropriate electron density enabled us to extend the pulse length with high performances. The new extension of the pulse length accelerates the understanding of roles of parameters with long time constants such as the current profile and particle recycling.

2.2.1 Sustainment of High Integrated Performance and High Fusion Gain

Toward the *simultaneous* achievement of i) high confinement , ii) high β limit, iii) high bootstrap fraction and iv) high efficiency of heat and particle exhaust in the *steady-state,* discharges have been optimized in JT-60U mainly based on the high-βp H-mode with q(0)>1. Up to I_p=1MA, an optimized pressure profile with high triangularity δ (=0.35) enabled the favorable integrated performance with H-factor ($=\tau_E/\tau_E^{ITER89PL}$) ~ 2.5 and β_N~3 under full non-inductive current drive (bootstrap ~60%) sustained for 2s (~10τ_E)[2.2-1,2]. In a 1.5MA/3.6T discharge with δ=0.35, a favorable integrated performance was sustained for ~2.6s (~8 τ_E) with Q_{DT}~0.2, β_N~2.5 and H-factor~2.5, bootstrap current ~50%, beam driven current ~25% (~75% non-inductive). In this discharge regime, a current profile with natural shear reversal close to the steady-state solution was observed because of sustainment of high values of βp. At a higher I_p =1.8MA (B_t=3.6T) with an ITER-like configuration (q_{95}= 3.4, δ= 0.3, κ =1.5), Q_{DT}= 0.27-0.3, β_N= 2.7-2.9, H~2.5, H/q_{95}= 0.74 was sustained for 0.7s (2xτ_E). With the new W-shaped pumped divertor [2.2-3], a long heating time with high power became possible. We obtained an ELMy H-mode with Q_{DT}~0.11, H-factor ~1.7, T_i(0)~10keV and β_N~1.8 sustained for 9 sec (~40τ_E) under a high NB power of 20-25MW (I_p=1.5MA, B_t=3.6T, δ=0.16). Even with the high total energy input up to 203MJ, no increase in impurity (carbon) and particle recycling was observed. Before the divertor modification, increase in carbon and recycling degraded performance at ~3 sec of high power (20-30MW) heating. In case of high δ (=0.3), the better performance with Q_{DT}~0.16, H-factor~2.3, β_N~2 and β_p~1.6 was sustained for 4.5 sec with 60-70% of non-inductive driven current at a relatively high density of ~45% of the Greenwald limit. Duration of the high δ equilibrium is limited (< 5sec) by heat capacity of the shaping coils. Without the divertor pumping, gradual increase in recycling degraded energy confinement.

2.2.2 β-limits in Long Pulses

For maximizing β_N in a long pulse, it is required to keep a sufficient stability margin against the ideal MHD instabilities. Experimentally, these ideal instabilities limit the transiently achievable β_N values. For this purpose, it is essential to control the heating profile to produce the optimum peakedness of pressure profile p(r) [2.2-2]. At a larger peakedness, the β_N-limit is lower due to the β_p-collapse which is consistent with the ideal kink-ballooning limit. At a smaller peakedness, β_N is limited at a low level by giant ELMs which is consistent with the ideal high-n ballooning limit. Therefore, a medium peakedness of the pressure profile has the highest β_N limit. Since the ELM-limit increases with δ, the β_N limit increases with δ.

However, in a long pulse discharges, another kind of the MHD instability appears and which limits the sustainable β_N much lower than the transiently achievable values. So far, even at the optimum p(r) with the transiently achievable β_N^{max} =3.2, sustainable β_N is ~2.5 for 2.6sec and ~2 for 5-9sec at I_p=1.5MA, B_t=3.6T , q_{95}~4 and l_i~1 with collisionality similar to ITER. This degradation is due to slowly (~100ms) growing resistive instabilities with mode numbers of (m/n)=(3/2), (2/1) etc. The detailed measurement of electron temperature by the heterodyne radiometer with high spatial resolution of 2cm showed growth and saturation of island width (~5cm) [2.2-4]. The neoclassical tearing mode is the candidate of this instability. By appearance of the resistive modes, values of sustainable $H\beta_N$ (H-factor × β_N) in quasi-steady phase lasting >2sec are smaller than the transiently achievable ones by ~50% (from 13 to 6.5 at δ~0.35). However, even in such long pulse discharges, the stability is also better at higher δ, for example, sustainable values of $H\beta_N$ increases with δ (from ~4 at δ~0.1 to 6.5 at δ~0.35). The threshold β_N for onset of the resistive modes increases with increasing electron density and with broadening of p(r). At a density higher than ~50% of Greenwald limit, β_N decreases because of confinement degradation at high recycling. Therefore, for sustainment of long pulse discharges, we kept the optimum set of δ, density and β_N to avoid both the resistive modes and confinement degradation.

On the other hand, at the edge, the confinement and steadiness of the H-mode are affected by giant ELMs and depth of the edge pedestal Δ_{ped}. The depth Δ_{ped} in the ELMing phase is 2-3 times larger than that in the ELM-free phase and reaches ~10-15cm with a roughly constant pressure gradient. With increasing recycling, the pedestal layer moves inward and the ELM frequency increases, which cause the gradual confinement degradation in a long pulse.

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2.3 H-mode Study

2.3.1 Scaling and Neutral Effect of L-H Tansition in the W-shaped Divertor in JT-60U[2.3-1]

The influence of the edge neutrals on the L-H transition condition was investigated in the W-shaped pumped divertor in JT-60U, which is an extension of the previous work published in Ref.[2.3-2]. In the dedicated experiment, the density was scanned in the range of (1.5-4.0)× $10^{19}M^{-3}$. It was found, however, that the amount of reduction in the H-mode threshold power in the modified divertor geometry was subtle, in comparison with the open divertor. The density exponent also remained in the range of 0.5 to 0.75, whilst it was 0.5 for the previous open divertor case. The signs of presumed geometry effect was found in the detailed analysis, where the slight reduction of threshold power was observed as the compression ratio was increased. As for the edge neutrals density, it was found that $\nu_i{}^{*95}$ (ion collisionality at 95% flux surface) stays around unity even under the condition that $n_0{}^{95}/n_e{}^{95}$ is considerably large in the modified divertor geometry, although the remarkable reduction of $\nu_i{}^{*95}$ was documented at a high $n_0{}^{95}/n_e{}^{95}$ in the open divertor. Here, $n_0{}^{95}$ is neutral deuterium density averaged in poloidal direction at 95% flux surface. However, it was also found that the consideration of the wall source at the outer buffle plate is necessary, whereas it was negligible in the open divertor geometry. In the basis of DEGAS calculation, the contributions of each source regions, which are divertor and outer buffle plate, can be comparable. This means that the poloidal averaging may possibly produce misleading results, as we have not resolved where in the edge poloidal section of the plasma being the most influential on the L-H transition.

2.3.2 Scaling of H-mode Power Threshold with the Edge Nondimensional Quantities[2.3-3]

We have employed the well established nondimensional treatment, and thereby described the conventional nondimensional quantities in terms of the relevant edge variables. .The nondimensional formulae for the H-mode threshold power P_{th} are also transformed to $P_{th}=[\rho^{*95}]^{\alpha_{\rho}95}[\beta^{95}]^{\alpha_{\beta}95}[\nu^{*95}]^{\alpha_{\nu}95}R^2 n_e{}^{95}[T_i{}^{95}]^{3/2}$,where $\alpha_{\rho}{}^{95}/2-2\alpha_{\nu}{}^{95}+\alpha_{\beta}{}^{95}+3/2=0$. A postulated hypothesis regarding the significance of the ρ^{*95} dependence, based on the ion orbit loss theory, is that H-mode becomes more accessible and sensitive to the fast ions with ρ^{*95}. The obtained scaling satisfies the constraint written above, with its value being 0.1. In addition, it is quite consistent with our global scaling result of $P_{th}=n_e{}^{0.5}B^{1.0}R^{1.5}$. As indicated in equation of obtained scaling, ρ^{*95} has a positive contribution to the threshold power against our hypothesis. A speculated reason for the apparent inconsistency on ρ^{*95} is that the above procedure does not separate the bulk plasma transport from the transition physics. Therefore, it would be necessary to take the contribution of nondimensional confinement scaling into account.

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2.4 Current Drive Experiments

Optimization of the operation of Negative-ion-source Neutral Beam Injector (NNBI: designed parameters of 500 keV and 10 MW with two ion sources) progressed and the injection power of 3.2 MW/source at 350 keV was achieved. The maximum injection power increased up to 4.2 MW using two ion sources. Utilizing the NNB and the lower hybrid (LH) waves, following new results were obtained.

The NNB driven current was well identified for the first time by reconstruction of the current density and loop voltage profile using the equilibrium code (EFIT) and the internal magnetic measurements form the motional Stark effect spectroscopy (MSE). A centrally-peaked profile of plasma current driven by the NNB was confirmed. The total driven current and a current density profile are consistent with the predictions with the ACCOME code [2.4-1]. The efficiency of the NNB driven current increased with the central electron temperature as expected from the ACCOME code. The controllability of the plasma current profile was also confirmed by comparing the driven current profiles by the NNB (350 keV) and the PNB (80 keV).

LHCD was applied to a reversed magnetic shear plasma in order to sustain internal transport barriers by keeping a hollow current profile. It was demonstrated in a plasma of $I_P = 1$ MA, $B_T = 3.5$ T that internal transport barriers in the electron, ion temperatures and electron density were maintained by applying LHCD. The internal transport barriers have been prolonged for about 2 seconds so far by LHCD. Although the period is not very long, this should be attributed to that an LH driven current profile was not fully optimized for the reversed magnetic shear configuration. A preliminary result of the current profile analysis from the MSE data suggests that some amount of current was driven also in the central region of the plasma, while the contribution of the LH current was dominant to make a hollow current profile outside the internal transport barriers [2.4-2].

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2.5 W-shaped Divertor and SOL plasmas

The principal goal of experimental campaigns using a new semi-closed W-shaped divertor is the demonstration of a cold-and-dense or detached divertor plasma with the enhanced energy confinement. The divertor was designed to control recycling neutrals [2.5-1] with an exhaust system using cryo-pump units. To increase the neutral density at the strike point, the divertor targets are inclined at angles of 70 and 60 degrees, and joined to baffle plates at the divertor throat. A private dome separates neutral transport between the inner and outer divertors.

2.5.1 Divertor Plasma and Detachment

At relatively high line averaged electron plasma density of a main plasma ($\bar{n}_e$), a large peak in the divertor electron density (n_e^{div}) profile was observed near the strike point on the divertor plate. Comparing to the open divertor case, the local n_e^{div} was a factor of 2 larger at the same $\bar{n}_e$, and divertor temperature (T_e^{div}) was reduced. These results indicate that the inclined divertor target and the private dome are effective in condensing neutrals near the separatrix. Consequently, onset densities of the divertor detachment and MARFE in the W-shaped divertor were reduced [2.5-2]. The inner private pumping with gas puffing at the plasma top preliminarily demonstrated that the concentration of charged carbon ions (C^{3+}) can be reduced in the divertor region[2.5-3].

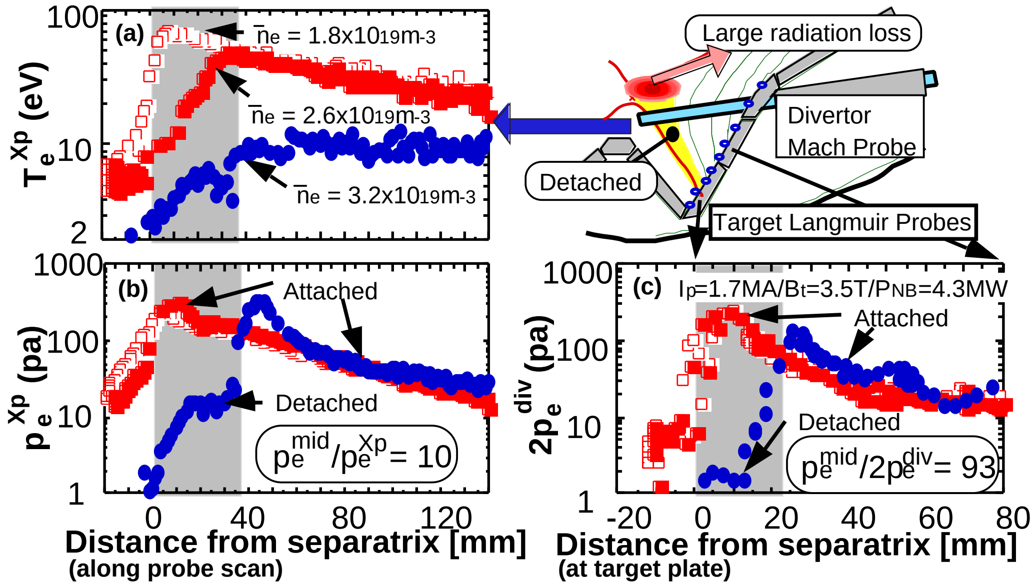

Fig. I.2.5-1 Plasma temperature and pressure profiles for attached and detached divertors.

Divertor plasma detachment simultaneously occurred at all points along the separatrix field line between the x-point and divertor target, which was measured from newly installed divertor Mach probe and target Langmuir probes (see Fig.I.2.5-1). Here the electron temperature profile in the outer flux surfaces becomes flat and density increases. Radial diffusion of particle flux may be enhanced upstream from the x-point. A large pressure loss factor of ~90 at the outer divertor target is obtained, which is larger than < 20 for the open divertor [2.5-2]. Due to the lower T_e^{div} and larger neutral density elastic and charge-exchange collisions may increase near the target.

2.5.2 Control of Recycling Neutrals [2.5-2]

Particle recycling generally does the most important contribution to fueling the main plasma. A change in the divertor geometry affected the distributions of neutral sources and neutral density. Neutral ionization fluxes at the main plasma edge and divertor ($\Phi_{D\alpha}^{main}$ and $\Phi_{D\alpha}^{div}$) were deduced from integrating the D_α signals. For an attached divertor plasma, an increase in $\Phi_{D\alpha}^{div}$ with $\bar{n}_e$ was similar for both W-shaped and open divertors. The value of $\Phi_{D\alpha}^{main}$ also increased with $\bar{n}_e$, but $\Phi_{D\alpha}^{main}$ for the W-shaped divertor was a factor of 2 - 3 smaller than that for the open divertor. The maximum neutral compression ratio, $p_{n0}^{div}/p_{n0}^{main}$, was measured to be 1000 (=1.2Pa/1.2 mPa).

Neutral particle distribution was investigated by a two-dimensional neutral transport code. Above the baffle plates, number of neutrals from the divertor source decreased significantly:

leakage of neutrals from the divertor to the main chamber was small compared to the open divertor. On the other hand, neutral sources, in particular, from the inner and outer baffle plates became dominant. The ionization source inside the separatrix originating from the baffle plates also had a larger contribution for fueling inside the separatrix (40% of the divertor source). As a result, a small reduction in the neutral density at the main plasma edge by the factor of 2 - 3 was obtained. This reduction in the edge neutral density is not as large as was predicted in design calculations.

Two (i.e. first and second) SOL regions with different characteristic lengths were observed in the n_e^{mid} profile measured by a midplane reciprocating probe. The decay length of the second SOL region was 3 - 4 times larger than the *e*-folding length of the first SOL, and was similar before and after the baffle plate installation. Quantitative evaluations of the ion flux to the outer baffle plates and the local recycling flux gave a good agreement, which suggests that the neutral source at the baffle plates is produced due to the interaction with the second SOL plasma.

2.5.3 Core Plasma Confinement of ELMy H-mode Plasmas [2-5-2]

A similar degradation in the H-factor (confinement enhanced over ITER89P-L mode scaling) of the ELMy H-mode plasma was observed at high density for the open and W-shaped divertors. The decrease in the edge neutral density (by a factor of 2-3) had no effect on the energy confinement. The reduction of the H-factor is due to the decrease in the fast ion slowing down time and in the thermal energy with the increase in $\bar{n}_e$. The total plasma pressure at the edge pedestal decreases at higher $\bar{n}_e$, which is caused by a reduction in the width of the pedestal region (from r = 0.95 - 0.99 to 0.97 - 0.99) with almost the same pressure gradient. Here the pressure gradient is not reduced. Effective fueling method inside the separatrix (e.g. a continuous pellet injection) should be implemented for the high density operation.

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2.6 Particle Transport and Exhaust with the W-shaped divertor

2.6.1 Steady-state Helium Exhaust [2.6-1]

When neutral beams of 60 keV helium atoms were injected to ELMy H-mode plasmas for 6 sec, efficient He exhaust was realized with He pumping using Ar frosted cryopumps for the W-shaped pumped divertor (see Fig.I.2.6-1). The He source rate (equivalent to 0.6 Pa•m^3/s) is balanced with its exhaust rate in a steady state, and high He exhaust capability (τ^*_{He}/τ_E=4) is successfully demonstrated, where τ^*_{He} is an effective He exhaust time. The enrichment factor of He is obtained about 1.0, which is 5 times larger than the ITER requirement (0.2). The exhaust rate increased with the electron density in the main plasma. Even without He pumping, an enrichment factor of 0.5 was obtained thanks to the W-shaped divertor. It seems that the reflection of He neutral particles near the inner strike point is enhanced by the W-shaped divertor. These results strongly support divertor designs in ITER.

In detached ELMy H-mode plasmas, τ^*_{He} is comparable to one in attached plasmas because recycling particle flux is enhanced at the inner strike point in a high density operation. Helium exhaust in detached plasmas is allowable for an ITER divertor operation scenario. The inner leg pumping worked well for He exhaust due to the inboard-enhanced He flux and deuterium flux, when the ion grad-B drift is directed to the target. The in-out asymmetry with He and deuterium flux profiles strongly affects the He exhaust capability.

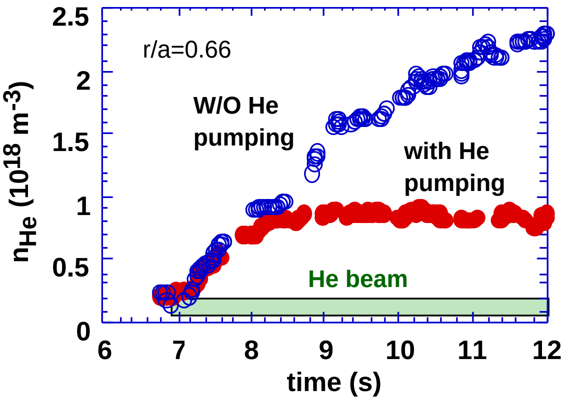

Fig. I.2.6-1 Time evolution of measured He density with and without He pumping. The He density with He pumping reaches a saturation level at 1.2 s after the start of the He beam injection.

2.6.2 Particle Transport in Reversed Shear Plasma [2.6-2]

The particle diffusivity and the convection velocity in the reversed shear plasma were separately evaluated based on the perturbation technique using modulated helium gas-puffing. The particle diffusivity in the region of the internal transport barrier (ITB) was reduced by about a factor of 2 compared with that in the core region surrounded by the ITB. The inward pinch was measured in the region of the ITB, while the outward convection velocity was observed in the core region. These results indicate that both of the particle diffusivity and the convection velocity largely related to the formation of the ITB.

2.6.3 Particle Balance and Neutral Particle Behavior [2.6-3]

The pumping rate of the W-shaped divertor was estimated from the quantitative analyses of the particle balance with and without divertor pumping. Furthermore, in order to understand the divertor pumping characteristics, the neutral particle behavior was analyzed using a neutral particle transport code, DEGAS. The ratio of the divertor pumping rate to the particle flux onto the divertor plates was estimated to be in the range of 0.5-2.5% for the density range of 2-4.3x10^{19}m^{-3}, and its strong dependence on the distance between the strike point and the pumping duct was observed. In the simulation of neutral particle behavior, a particle source from the outer baffle plates was found to be ~5% of the divertor source. The estimated pumping rate in experiment was a factor of three smaller than the predicted one. This difference might come form the effects of the structure under the baffle plates, which will be investigated in a future work.

2.6.4 Volume Recombination in the Divertor Plasma [2.6-4]

Understanding of the volume recombination for detached plasmas is important in tokamak fusion reactor design, because the detached divertor regime is attractive to reduce the ion flux incident to divertor plates. Balmer-series lines of deuterium atoms were observed and the population distribution for excited levels of the deuterium atoms was investigated. The ratio of the recombination sink to the ionization source was estimated from the ratio of the D_ε line intensity to the D_α line intensity for partially-detached divertor plasmas. While the onset of the recombination was correlated with the plasma detachment, the recombination sink was estimated to be about 1 % of the ionization source. This suggests that the recombination is not a principal cause of the detachment.

2.6.5 Behavior of He Atoms in the Divertor Region [2.6-5]

In JT-60U, it has been found that the Doppler width of He I line emitted from the divertor region increases with the increase in the electron density [2.6-5]. The atom temperature corresponding to the Doppler width is up to 1.7 eV for detached plasmas. Understanding of the He atom behavior is important to establish an effective system for He exhaust. Thus the Doppler broadening has been reproduced by numerical calculations using a neutral particle transport code. The broadening is attributed to elastic collisions with H^+. For an L-mode discharge a probability of penetration of He atoms from the outer divertor tiles into the main plasma was estimated to be 7 %, but this probability drops down to 4.5% in a calculation with neglecting the elastic collisions. Thus the elastic collision is expected to largely affect He contamination in main plasmas.

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2.7 Fast Ions and Alfvén Eigenmodes

The Toroidicity-induced Alfvén Eigenmodes (TAEs) and high frequency modes observed in ICRF-heated low-q discharges were analyzed in detail using the NOVA-K code (PPPL). It was shown that TAEs appeared before giant sawtooth crash were excited inside q=1 surface [2.7-1] and high frequency modes observed after the crash were the Ellipticity-induced Alfvén Eigenmodes (EAEs) excited at the q=1 surface [2.7-2]. The interaction between TAEs/EAEs and NNB-injected ions was investigated, and the EAEs were stabilized with the NNB. The stability analysis using the NOVA-K code suggested that the stabilization mechanism was beam ion Landau damping. It was also shown that the q-profile derived from the change in TAE mode frequencies agreed with that obtained from the motional Stark effect spectroscopy [2.7-3]. Chirping modes were observed in ICRF-heated weak magnetic shear plasmas [2.7-4]. The TAEs were excited with the NNB. Both burst and continuous modes with low toroidal mode numbers were observed in a low β_h regime of $<\beta_h>\leq 0.1$-0.2%, here, β_h is the beta value of energetic ions and $<\beta_h>$ is the volume-averaged one. The amplitude of magnetic fluctuations of the burst modes is about ten times as large as that of the continuous modes. Accompanying these bursting activities were 2-3% drops in the neutron emission rate. This small drop indicates that the loss of the co-injected NNB ions is small.

The ICRF coupling with plasmas in the W-shaped divertor was optimized by adjusting a gap between the first wall and the separatrix. The coupling resistance was similar to that in the open divertor and the ICRF power of ~4 MW was applied to reversed-shear plasmas. By changing the NB power and the ICRF power and by replacing the ICRF power by the NB power, the formation of the ITB was investigated. It was shown that the NB power of 4-5 MW was necessary to sustain the ITB in the electron density profile. The effect of magnetic shear on an increase in electron temperature was also investigated in the ICRF-heated reversed and normal shear plasmas. It was shown that the heating profile was hollow due to an expansion of banana orbits and/or enhanced ripple transport in the reversed shear plasma [2.7-5]. A new scaling including plasma current was obtained for the temperatures of ICRF-driven tail ions, which was based on a diffusion model of fast ion losses. The measured tail temperatures were well described by this scaling.

Enhancement in the ionization cross-section of the NNB was evaluated at 350 keV/amu. Measured shine-through of the NNB was lower than that calculated by assuming the single-step ionization process. This result shows that the multi-step ionization process is needed to be taken into account. The experimentally obtained enhancement factor of the ionization cross-section agrees with that predicted using the enhanced cross-section evaluated by Janev [2.7-6].

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2.8 Plasma Control and Disruption

2.8.1 Plasma Control

Preparatory work for the equilibrium control of plasmas under the modified divertor geometry started early in this year, including the compilation of over 800 numerical equilibrium data necessary to evaluate the function parametrization coefficients. The discharge procedure was also investigated deliberately with the equilibrium calculation code MEUDAS, considering the baffle and dome structures. In order to cope with various technical limitations, "startup operation group" was organized, and detailed physics operation procedure was discussed, including the heat load onto the in-vessel components and commissioning program of various divertor diagnostics. Accordingly, the nominal 15 s stable discharge was produced on the 2nd day of the campaign with its discharge waveforms as well as its equilibrium shape and positions exactly as predicted with the calculation code. Subsequently, the dynamic range of physics operation was confirmed, and the halo current experiment was conducted to establish the database to increase the plasma current.

Various real-time feedback schemes were prepared in 1997, in addition to the conventional density, neutron rate and divertor radiation feedback controls. The CO_2 laser interferometer data, which provides the averaged density near the magnetic axis, and neutral pressure signal in the divertor as well as the divertor density information have been made available as feedback tools on ZENKEI-1bR computer. The neutron feedback algorithm was also modified to implement the differential control, which turned out to be an effective method of stability control in the reversed shear high performance experiment.

2.8.2 Disruption Studies

The disruption studies performed in 1997 focused on the investigation of halo current characteristics, which is also an urgent ITER Physics R&D issue. The most dangerous disruption caused by vertical displacement event (VDE) was experimentally simulated, in which a plasma was actively controlled to move downward. Ranges of the measured total halo current normalized by initial plasma current (I_h/I_{p0}) and toroidal peaking factor (TPF) were 0.05 to 0.26 and 1.4 to 3.6, respectively, in the ranges of I_p = 0.7-1.8 MA, B_T = 2.2-3.5 T, κ = 1.3-1.6 and q_{95} = 2.8-7.0. The maximum TPF×(I_h/I_{p0}), corresponding to the maximum local halo current, was 0.52 so far, which was lower than that of the maximum value of the ITER data base of 0.75. The upper boundary of TPF×(I_h/I_{p0}) tended to decrease with the increase in I_{p0}. Other parameter dependencies of TPF×(I_h/I_{p0}) on B_T, κ and q_{95} were not clear. We confirmed that the upper boundary of TPF×(I_h/I_{p0}) decreased with the decrease in the vertical shift velocity ($-dZ_j/dt$). On the other hand, the TPF×(I_h/I_{p0}) clearly decreased with the increase in the stored energy just before the energy quench ($W_{dia}^{eq.}$) and the line integrated electron density at the peak of halo current. These stored energy and density dependencies of the halo current may be explained by a increase of halo resistivity, probably caused by a large amount of impurity generation during disruptions. The magnitude of the halo current decreased by about 40% by applied a strong pulse gas puff (H_2 of 50 Pam^3/s x 0.1s) during VDE. Perfect avoidance of the halo current has been demonstrated by maintaining the plasma vertical position during the current termination [2.8-1].

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3. Design Progress of the JT-60SU

3.1 Optimization for Steady-state Advanced Operation

The JT-60 Super Upgrade (R_p=4.8 m, B_t=6.25 T, $I_p \leq$10 MA [3.1-1, 2, 3]) has been designed as a superconducting tokamak for establishing an integrated scientific basis of a steady-state tokamak reactor and for contributing to an advanced steady-state scenario in ITER. Figure 1 shows a schematic drawing of the JT-60SU machine, where the diameter of the cryostat is 22 m and the total weight of the device including the cryostat is ~11000 tons. After 10 years D-D operation and installing the extra shield made of reduced activation ferritic steel, a steady-state D-T operation with Q_{DT}~5 is considered as an optional scenario.

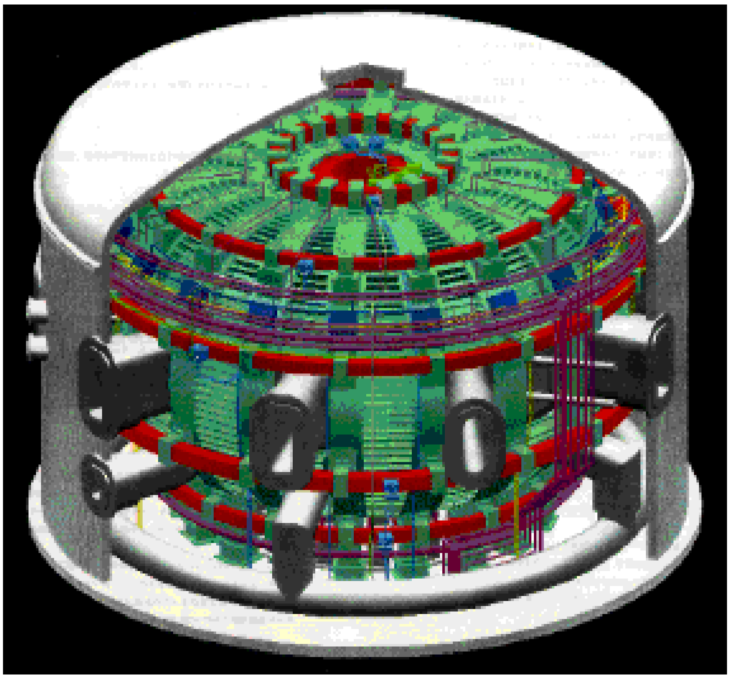

Fig.I.3.1-1 Schematic drawing of JT-60SU

JT-60SU is designed to have a high Greenwald density limit (>1x10^{20}m^{-3}) by selecting B_t/R~1.1-1.3 in order to perform a steady-state operation research at high density regime. At I_p=5-6MA, a fully non-inductive discharge can be expected at <n_e>~0.88x10^{20} m^{-3} by using 60 MW of CD power, which is much higher than the ITER scaling law for H-mode power threshold (~40 MW). In addition to a 750 keV N-NBI system for core heating and current drive, 150-220 GHz ECH system is adopted to provide flexible current profile control for establishing an advanced steady-state operation scenario with a stable reversed shear configuration in JT-60SU.

Ten units of independent PF coil system are adopted in JT-60SU to have a capability to produce a wide variety of plasma shaping (elongation κ_X up to 2.0 and triangularity δ_X up to 0.8 for DN divertor) for improving the β-limit in the steady-state operation scenario. Vertical Displacement Event (VDE) in JT-60SU has been also investigated by using Toroidally Symmetric Plasma Simulation (TSPS) code in which the Grad-Shafranov equation and a linearized equations of plasma motion taking into account the effects of eddy current on the vessel and baffle plates are iteratively solved [3.1-4]. Fast vertical position control system composed of two sets of normal conductors (10 turns) located near the vacuum vessel is adopted in JT-60SU for suppressing VDE. TSPS code has indicated that the VDE can be suppressed by fast vertical position control with the power supply of 200 V when the disturbance is moderate (a rapid change in βp during a minor disruption $\Delta\beta p \leq -0.6$ for 10 ms).

The ideal and resistive stability of the reversed shear scenario on JT-60SU is investigated using the equilibrium assuming a correlation between plasma pressure and magnetic shear scale length observed in JT-60U experiments. Stability analysis has indicated that growth rate of n=2 tearing mode is slightly reduced with increasing β_N, while an n=2 ideal global mode becomes unstable suddenly at around β_N=2.

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3.2 Progress in Engineering Design

Significant progress on the engineering design of JT-60SU has been made. For R&D of Nb_3Al superconductor, which is employed for TF coils because of its better mechanical and J_c properties than $(NbTi)_3Sn$ superconductor, almost all important engineering techniques for producting Nb_3Al strand is thought to be established. It has been demonstrated to make a 11km Nb_3Al strand with Jc=650 A/mm^2, RRR=131 without any wire breaking. An Nb_3Al strand with a low AC loss (a filament diameter of 31μm) with Jc=701A/mm^2 is also developed. Fe-Cr-Mn steels (C:0.02-0.2wt%, Mn:15wt%, Cr:15-16wt%, N:0.2wt%) with a lower induced-radio-activity than 316SS has been developed as a material of structure components for JT-60SU[3.2-1]. It has been confirmed that the developed high manganese steels have excellent mechanical properties and high resistance within standard temperature of JT-60SU vacuum vessel. By using this steel as the vacuum vessel, a rapid decay of the radioactivity of the machine than 316SS can be realized after two years DT operation. A fine modification of TF coil design was made for reducing a local stress on radial disk. By increasing the length of the wedge part of coil case, the maximum local stress is reduced to 643 MPa.

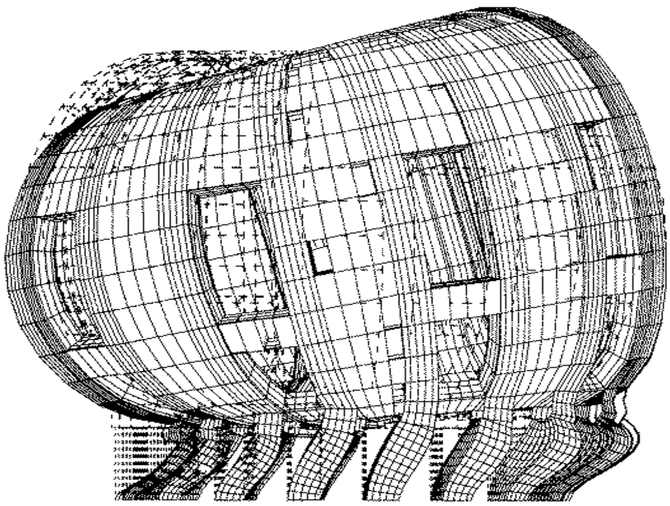

Fig.I.3.2-1 Model of superconducting coils for vibration analysis in JT-60SU; n=1 vibration with 16.2Hz

In addition to a safety design of a low tritium inventory (<100 g) in the DT optional operation scenario, a dynamic analysis of JT-60SU machine at emergency events such as earthquake and short circuit of TF coil is also performed. With respect to the safety of a fusion reactor; confinement of tritium, the eigen mode frequency on vibrations of the machine should be 10-20 Hz to reducing displacement of each component. JT-60SU superconducting coil systems (TF:2340 tons, EF:520tons, shear panels:660 tons and CS is not included) are modeled to analyzing their dynamic behaviors during an earthquake as shown in Fig.3.2-1. Fundamental frequency of vibration in JT-60SU coil systems is around 10 Hz, which is larger than ~1.5 Hz in ITER [3.2-2]. It is found that a weight reduced design of TF coils, shear panel connection and favorable design of supporting system contribute to realize a higher fundamental frequency for vibration. Preliminary analysis applying EL Centro wave form with 0.326 gal on the basement of machine room has indicated that the maximum displacement of the TF coil is within 2 mm.

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II. JFT-2M PROGRAM

Objectives of the JFT-2M program are (1) advanced and basic researches for the development of high-performance plasmas for nuclear fusion and (2) contribution to the physics R&D for ITER, with a merit of flexibility of a medium-size device. In the closed divertor experiments, effects of reduction of gas back-flow from a divertor region have been investigated by progressively changing a degree of closure at the divertor throat. It was found that more closed divertor geometry can further extend a coexistent regime of the high confinement and a dense & cold divertor plasma. It was also found that divertor biasing which produces $E_r \times B$ flow in the SOL from the inside to the outside, enhances the divertor function significantly with baffle plates. A compact toroid (CT) injection system has been installed in collaboration with the Himeji Institute of Technology (Prof. T. Uyama) for the development of the advanced fuelling for a fusion reactor, such as ITER. Encouraging results were obtained with initial CT injection experiments, such that reduction of radiation loss power was observed after the CT injection into OH plasmas. In JFT-2M H-mode, H-factors are different according to the beam injection angle, i.e. H(CO)>H(CO+CTR). It was found that larger H-factors with co-NBI is due to the increased E_r sheared region. A heavy ion beam probe system, which was developed by the National Institute for Fusion Science (Prof. Y. Hamada), has been installed for clarifying mechanism of improved confinement more definitely through fast measurements of the electric field. Preionization experiment using the FWCD combline antenna was attempted, showing that the loop voltage at the plasma current start-up decreased from 22 V to 14 V by the FW preionization. Test of newly installed EC antenna for current drive and MHD control was carried out in the vacuum. For the development of structure material for a DEMO reactor, such as low-activation ferritic steel for SSTR, design studies of the Advanced Material Tokamak Experiment (AMTEX) program have progressed, where toroidal field ripple reduction by ferritic inserts and high performance plasma production in a ferritic vacuum vessel will be tested.

JFT-2M was operated in accordance with the experimental plan of FY 1997. Periodical check-ups of the tokamak, heating and power supply system were done in January and February, 1998. Operation was restarted in March. The JFT-2M operations were carried out smoothly on schedule in FY 1997, counting 2369 experimental shots.

1. Experimental Results and Analyses

1.1 Closed Divertor

The neutral gas pressure or particle recycling level around the core plasma should be low for a good confinement. On the other hand, a high recycling condition and high gas pressure in the divertor chamber are required in order to form a cold divertor plasma for the reduction of heat load onto a divertor plate and to pump the fuel or diverted impurities as well. Divertor configuration of many tokamak machines have been modified to a closed configuration with baffle plate in order to decouple main and divertor chambers, but the decoupling is still insufficient.

We proposed a measure of tdegree of divertor closure, δ/λ, where δ is a distance from the separatrix to the baffle plate and λ is a scale length of the particle flux.[1.1-1]. For a quantitative comparison of the results obtained with various divertors, this measure is better to put together with each data set because even if the data obtained with the same machine or the same divertor configuration, the degree of closure can be varied according to the various plasma configurations, and the data set could vary with this.

In the JFT-2M tokamak, the degree of divertor closure has been modified step by step, i.e. δ/λ»1 (open divertor) → δ/λ=1.3 (closed divertor: CD1) → δ/λ=0.8 (closed divertor: CD2) as shown in Fig.II.1-1. (The degree of divertor closure of other machines are grater than unity in general.) In the extremely closed case, CD2, the decoupling of divertor and midplane pressures has been much improved (Fig.II.1-2). With an open divertor configuration, the main chamber pressure measured by a penning gauge at outer midplane, P_{mid}, as well as the recycling light is increased with increasing divertor pressure, P_{div} (the correlation factor is almost unity), but there is much less correlation with a closed divertor as shown in Fig.II.1-2.

This enabled extension of a coexistent regime of H-mode and a dense and cold divertor plasma with a strong gas puff in the divertor region. Typically, H factor of 1.6 was obtained at $\bar{n}_e/n_e^{GW}$~0.6 with n_e=3x10^{19}m^{-3}, T_e=5 eV in the divertor. Furthermore, a new quasi-steady-state improved confinement mode compatible with dense (3x10^{19} m^{-3}) and cold (4 eV) divertor appeared when a strong gas puff was applied to high density H-mode plasma.

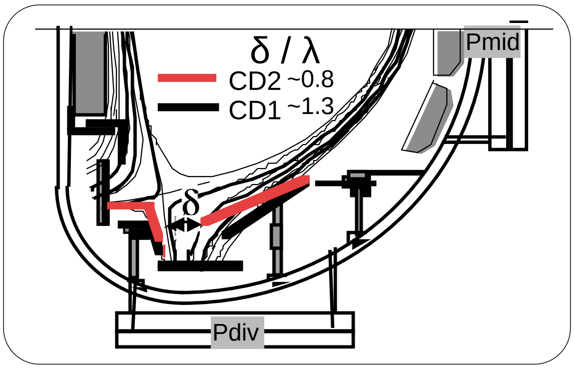

Fig. II.1-1 Variation of divertor closure on JFT-2M

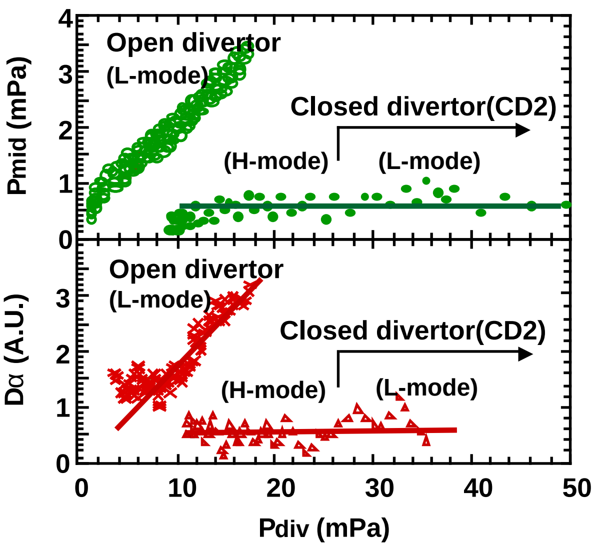

Fig. II.1-2 Correlation between midplane pressure, P_{mid}, and divertor pressure, P_{div}, with open and closed divertor (The recycling light from outer edge of the main plasma, D_α, also shows similar trend as P_{mid})

References
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1.2 Compact Toroid Injection

A dense and fast compact toroid (CT) injection is considered to be a core fueling method for a fusion reactor. The CT experiments on the JFT-2M tokamak started in November 1997 for the purpose of realizing a high confinement H-mode at high density using the CT injector (HIT-CT1) developed by the Himeji Institute of Technology. The penetration depth of the CT into the tokamak magnetic field is determined by the balance between the kinetic energy of the CT and the toroidal magnetic field (B_T) pressure. A fast framing camera is used to observe the extent of CT penetration into the JFT-2M vacuum toroidal field. Figure II.1-3 shows pictures of CT at different B_T; (a) B_T =0, (b) B_T =1 T with the counter clock-wise (CCW) direction and (c) B_T =1 T with clock-wise (CW) direction. The CT travels straight and crashes into the inside wall at B_T =0. At B_T =1 T, the CT is able to go to the extent of plasma region ($r/a \geqq 0.7$) and moves up or down on the poloidal plane, according to the direction of B_T. Initial plasma injectionexperiments were performed at B_T =0.9 - 1.3 T with a single null plasma configuration. The stored energy W_{MHD} increased after the CT injection accompanied decreases in the radiation loss P_{RAD} and the loop voltage V_L. In an ELM-free H-mode plasma, Giant ELMs occurred 5 ms after the CT injection. The ELM free H-mode did not disappear just after the CT injection, because the life time of the CT is in the order of 50 μs in tokamak magnetic field, which was confirmed from observation by the fast framing camera.

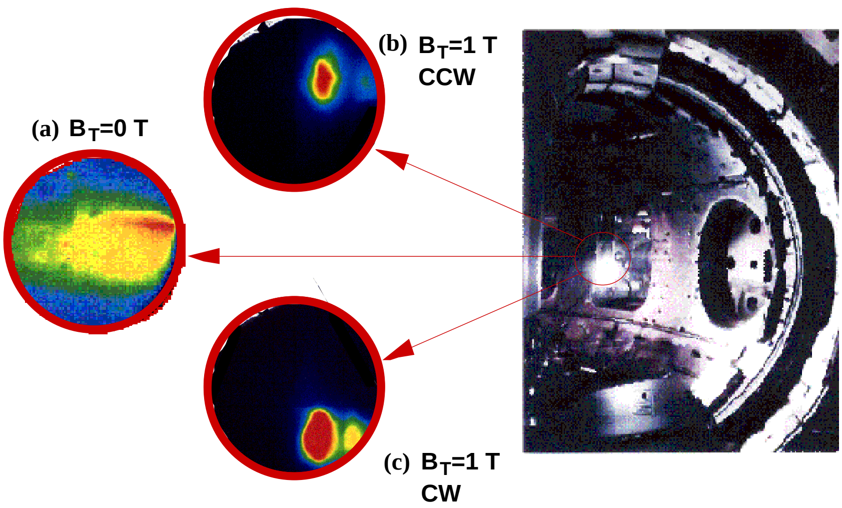

Fig.II.1-3 Pictures of the CT inside the JFT-2M vacuum vessel taken by a fast framing camera. (exposure time: 50ms)

1.3 H-mode Study and Development of Heavy Ion Beam Probe System

The improvement of the H-mode confinement in the JFT-2M tokamak shows the difference between CO- and CTR-NB heating. The improvement in CO is larger than that in CO+CTR or

CTR H-mode. The reason of τ(CO)> τ(CO+CTR) is considered to be not the effect of the sheared toroidal rotation, dv_φ/dr in the core. The v_φ at the last closed flux surface shows a finite value (about 20-40 km/s and the direction is depending on the beam injection angle) in a divertor H-mode. The finite toroidal rotation at edge is playing a role for the observed τ(CO)> τ(CO+CTR). Since the radial electric field formed by the poloidal rotation is negative, then the electric field shear at edge is increased by the positive electric field of $v_\varphi(CO)\times B_\theta$. On the contrary, the electric field by $v_\varphi(CTR)\times B_\theta$ is negative, then it reduces the electric field shear. Figure II.1-4 shows the difference of the radial electric field and its shear between the case of CO and CO+CTR H-mode. In the case of CO H-mode, the sheared region of the radial electric field is increased by the finite toroidal rotation just inside separatrix. Therefore, the difference of the improvement between CO- and CO+CTR H-mode, which comes from the difference of the pedestal height, is related to the electric field shear at edge[1.3-1].

The time resolution of the radial electric field measurement is 16.67msec. It is impossible to know the causality between the formation of a radial electric field and that of a transport barrier. Then we are preparing the Heavy Ion Beam Probe (HIBP) diagnostic system collaborating with National Institute for Fusion Science. Now we are doing the calibration of the system.

References

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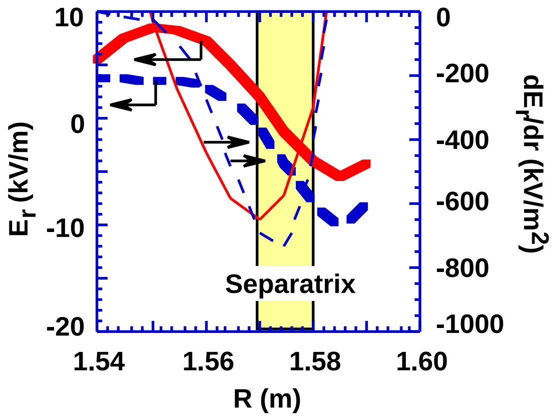

Fig.II.1-4 Thick solid and broken lines show E_r in the case of CO and CO+CTR H-mode, respectively. Thin solid and broken lines show dE_r/dr in those cases.

1.4 Radio-frequency Experiments

1.4.1 Pre-ionization by fast waves

Fast wave start-up assist is one of the high priority issues of the ITER physics R&D. Experiments of pre-ionization and start-up assist by 200 MHz fast waves were carried out using a travelling-wave-type antenna (combline antenna, developed by General Atomics). This antenna is well suited for this objective because of its load insensitivity. Plasma production was observed over a wide range of the toroidal field, 0.5 - 2.2 T. Figure II.1-5 shows reduction of the peak loop voltage from 22 V to 14 V using

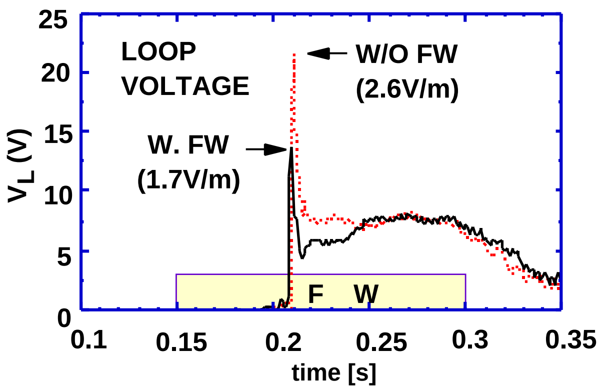

Fig.II.1-5 Time evolution of loop voltage with and without fast wave start-up assist.

fast wave pre-ionization at B_T=1.3 T.

1.4.2 Testing of new electron cyclotron wave antenna for current drive

A new antenna for ECCD installed in the JFT-2M tokamak can launch the wave in the HE_{11} mode with variable injection angle (toroidal/poloidal: ±25/20 degrees from perpendicular injection, respectively). The direction of the beam and beam divergence were measured using the three horn reflector antennas newly settled on the wall in the opposite side. The measured full half width of the rf beam (160 kW, 3 ms from one gyrotron) was 12 degrees in the toroidal direction and 10 degrees in the poloidal direction as expected.

1.5 Advanced Material Tokamak Experiment (AMTEX) Program

1.5.1 Ripple reduction by ferritic inserts and ferritic vacuum vessel [1.5-1]

There are some advantages using a ferritic steel as a material for a DEMO fusion reactor. The reason is a possibility to reduce radioactive waste, good thermal properties and high swelling resistance. Therefore, the ferritic steel is proposed for blankets in SSTR. However, its magnetism is worried about because of an error field in a magnetic fusion device. At the same time, its magnetism is considered to be used to reduce toroidal field (TF) ripple. If the ferritic steel is positioned appropriately, such that it strengthens the magnetic field between the toroidal field coils (TFC), where the toroidal magnetic field is weaker than that just inside the TFC, the TF ripple can be reduced. In ITER, it is indeed planned to use a ferritic steel to reduce the toroidal field ripple (to reduce fast ion losses). In order to examine the effects of ferritic steel on ripple reduction and plasma properties, AMTEX will be carried out in JFT-2M. In the first phase of AMTEX, in order to test TF ripple reduction, the ferritic boards are added between the nonmagnetic material vacuum vessel (VV) and the TFC. The equi-ripple-amplitude region in the cases of the nonmagnetic VV without (present VV) and with ferritic board are shown in Fig.II.1-6.

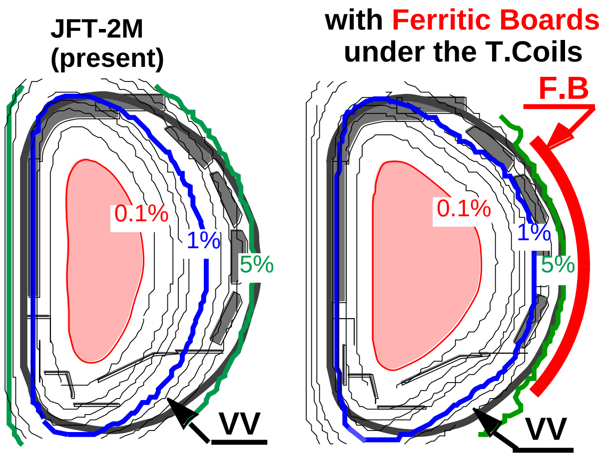

Fig.II.1-6 Equi-ripple-amplitude region in the cases of the nonmagnetic VV without (present VV) and with ferritic board.

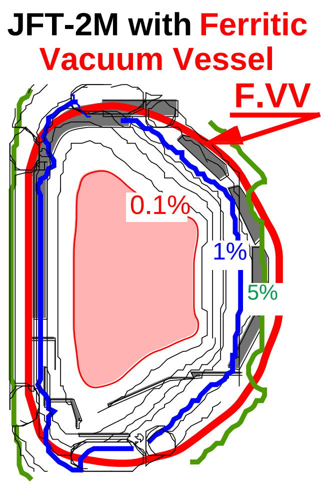

Fig.II.1-7 Equi-ripple-amplitude region in the case of the ferritic VV.

Computational results show that the ripple is reduced in the whole plasma region of low field side by the appropriate setting of the ferritic board near the VV. The ripple amplitude can be reduced by a factor of 3: ripple amplitude is reduced from 1.8 % to 0.6 % on the plasma boundary. In the second phase of AMTEX, plasma properties with the ferritic steel VV will be tested after replacing the present VV with the ferritic steel VV. The equi-ripple-amplitude region in the case of the ferritic VV with realistic horizontal port is shown in Fig.II.1-7. The ripple in the case of ferritic VV with realistic horizontal port is comparable with that in the case of nonmagnetic VV with the ferritic board.

1.5.2 Preliminary experiment with ferritic inserts

Preliminary experiments with ferritic board insertion ($0.5m^{H}$ x $0.15m^{W}$ x $24\sim48mm^{D}$) just inside only two TFCs (ripple reduction) and at only one toroidal section between a pair of TFCs (ripple enhancement) were carried out. In those experiment, we could not find any effect of the FB insertion on the global plasma parameter. The error field in the order of several ten Gauss may be below a limit of the allowable error field.

1.5.4 Irradiation of vanadium alloy in tokamak

The vanadium alloy is one of the candidate material for a DEMO reactor. Since its brittle fracture by hydrogen and oxygen absorption is worried about, the Vanadium alloy was exposed in JFT-2M in collaboration with GA, MIT and Hokkaido University. Initial results show that the hydrogen absorption and the brittle fracture of the Vanadium alloy are smaller than those of pure vanadium and titanium alloy.

References

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2. Operation and Maintenance

2.1 Tokamak Machine

The operation and management of the tokamak device went very smoothly. After all joints of the toroidal coil coolant were replaced last fiscal year, no water leak was observed and there was no problem in insulators of magnets. An anomalous waveform of the Q poloidal coil power supply appeared possibly due to a noise. It was solved by fixing the potential of the control output signal. Damaged divertor probes and MI cables were replaced. Old equipments were renewed, and check and maintenance works were carried out in the gas fueling, the vacuum pumping, the cylister cooling system of the poloidal power supply, and the coil secondary cooling system.

In collaboration with the Experimental Plasma Physics Laboratory, installation of the CT injection device, the power supply, and the vacuum pumping system was completed in October. The CT injection device was developed by the Himeji Institute of Technology. The power supply and the vacuum pumping system were designed and fabricated by JAERI. After tests of the power supply with dummy loads and standalone tests of the CT injector, the injector system was connected to the JFT-2M device. After optimization of the CT plasma production, CT plasma injection experiments were started.

As a part of the Advanced Materials Tokamak Experiment (AMTEX) program, the toroidal field ripple reduction by ferritic inserts is being planned. Investigation was done for the installation of ferritic steel boards between the vacuum vessel and the toroidal coils. Technical assessments were made on the mounting structure and shape of the ferritic steel boards.

2.2 Neutral Beam Injection System and Radio-frequency Heating System

The systems of Neutral Beam Injection (NBI) heating, Electron Cyclotron Heating (ECH) and Fast Wave (FW) current drive were operated without a major problem. These systems were efficiently used for the experiments. A turbomolecular pump of a magnetic levitation type was damaged due to vibrations of the gate valve in a shut-down operation. To prevent such a vibration problem, the exhaust manifold support was reinforced and a bellows was installed to absorb vibrations. Old equipments of the acceleration-power-supply control system were replaced and check and maintenance works were carried out. Inspection, maintenance and aging of the ECH system were carried out in order to increase injection power and pulse duration. A power combiner was installed for the traveling wave antenna (combline antenna) of the FW current drive system, and it was used in the experiments.

2.3 Power Supply System

The toroidal coil power supply (MG) ran smoothly and contributed to the experiments. Furthermore, following works and improvements were done for increasing the efficiency of operations; inspection and maintenance of commutators, improvements of interlocks of the main circuit switch and reduction of time for stopping the MG.

III. THEORY AND ANALYSIS

The principal objective of theoretical and analytical studies is to improve the understanding of physics of tokamak plasmas. Remarkable progress was made on the physical understanding of the reduced transport and the stabilities not only of ideal MHD modes but also of kinetic ballooning mode in reversed shear plasmas. Progress was also made on the neoclassical transport calculation by the Matrix Inversion method and on the scaling law of an offset nonlinear form for the ELMy H-mode confinement. A five point model for the scrape-off layer and divertor plasmas was developed and the inside/outside divertor asymmetry was investigated.

The main purposes of the NEXT (Numerical EXperiment of Tokamak) project, which began in 1996, are researches on complex physical processes in core plasmas, such as transport and MHD, and in divertor plasmas by using recently-advanced computer resources. The Next project also includes the development of simulation models suitable for a large, high temperature tokamak with reactor parameters, and the development of simulation technology on massively parallel computers.

1. Confinement and Transport

Recent experimental and analytic progress in the JT-60U was reported in [1-1]. Especially the confinement and transport in reversed shear (RS) plasmas were investigated. Ion and electron thermal diffusivities in a quasi-steady-state RS plasma were obtained, which become very small inside the thin ITB (internal transport barrier) layer and are the same level or smaller than the neoclassical ion thermal diffusivity in the core region enclosed by ITB. This transport feature is similar to that in the transient phase. The stability of high-n toroidal drift modes was analyzed. The E×B shearing rate becomes of the same order of magnitude as the linear growth of the dominant mode around the ITB. It was found that the anomalous transport cannot be enhanced by the steep pressure gradient of the ITB.

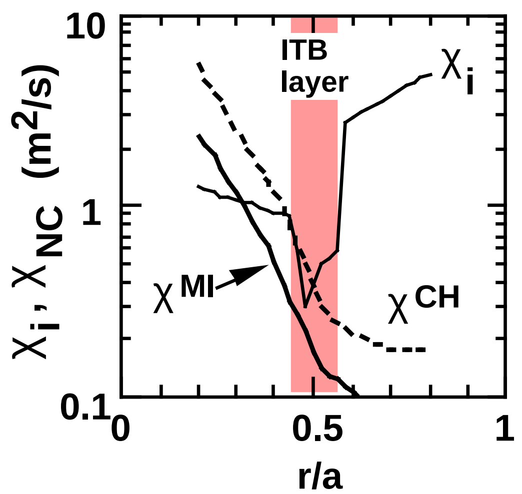

Fig.III.1-1 Profiles of ion thermal diffusivity, χ_i, and neoclassical diffusivity, χ_{NC}, evaluated by MI method and by Chang-Hinton's formula in the reversed shear plasma with internal transport barrier (ref. [1-1]).

In the improved confinement plasmas, such as RS plasmas, the ion thermal diffusivity is reduced to the neoclassical level. The accurate estimation of the neoclassical transport coefficients is required. A Matrix Inversion (MI) method for calculating the bootstrap current was modified to calculate the neoclassical ion thermal diffusivity including the effect of impurity [1-2]. It was found that the ion thermal diffusivity calculated by the MI method is about half of that calculated from the Chang-Hinton formula for a typical hot-ion H-mode in JT-60U.

The role of convective heat losses in tokamak plasmas was studied analytically and numerically [1-3]. A the natural forms for the energy confinement scaling law was suggested. An analytical formula of the ion temperature limit in the steady-state neutral beam injection discharges caused by convective heat losses was obtained. A new technique for the determination of convective heat losses from experimental measurements was applied to the analysis of a JT-60U NBI discharge. The particle source and convective heat losses calculated directly from the experimental data were in good agreement with ASTRA calculations. The convective heat loss obtained from the data was about 50% of the absorbed NB power in the core region.

An offset nonlinear scaling was developed for the ELMy H-mode confinement by analyzing the ITER H-mode database ITERH.DB2 combined with JT-60U data [1-4]. The offset part of the stored energy is determined by the MHD stability of the ELM, and the incremental confinement time of a nonlinear function of heating power is determined by gyro-Bohm-like transport in the core plasma. This scaling predicts a lower confinement time for ITER than that predicted by general power-law scaling.

A non-linear Fokker-Planck code was applied to the study of a JT-60U hot ion plasma in which the experimentally measured carbon impurity temperature T_C reached up to 45 keV with 90 keV deuterium beam injection [1-5]. A non-Maxwellian deuteron distribution function is obtained numerically and the deuteron bulk temperature T_D, which has not been determined experimentally, is evaluated from the slope of the energy spectrum. It was found that T_D can exceed T_C, indicating that the T_C measurement does not lead to overestimation of the ion temperature. The deuteron effective temperature based on the average energy was found to be almost the same as T_C. The DD fusion reactivity is also around a value given by the Maxwellian distribution with its temperature equal to T_C. Consequently, the T_C may possibly be regarded as an equivalent ion temperature.

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2. Stability

Ideal MHD stabilities were studied for the negative shear plasmas with steep pressure gradient observed in JT-60U [2-1]. In order to improve the ideal β-limit, effects of the location of a minimum value of the safety factor, q_{min}, and of the location of the maximum pressure gradient on ideal MHD stability were investigated. When the maximum pressure gradient is inside the $q = q_{min}$ surface, the ideal β-limit is determined by the $n = 1$ kink-ballooning mode. When the maximum pressure gradient is on or outside the $q = q_{min}$ surface, the infernal and high n ballooning modes become more unstable than the $n = 1$ mode. It is found that the ideal β-limit is improved when the maximum pressure gradient is inside the $q = q_{min}$ surface and the edge pressure gradient is high. The experimentally observed β-limit in negative shear plasmas in JT-60U is consistent with the numerically obtained β-limit determined by the $n = 1$ mode.

The Mercier criterion in a reversed shear plasma of a tokamak was studied numerically [2-2]. A reversed shear plasma has negative magnetic shear and negative pressure gradient in the inner region of a plasma. In the negative shear region, stabilizing terms due to the parallel current and the magnetic well produced by the poloidal current change to destabilizing ones. As the value of $(q_0-q_{min})/q_{min}$ increases, the destabilizing effects increase and the Mercier criterion can be violated. Here, q_0 and q_{min} are the safety factor on the plasma axis and the minimum value along the minor radius, respectively. In JT-60U reversed shear plasmas, the value of (q_0-q_{min})/qmin becomes large. The violation of the Mercier criterion seems to be consistent with the observation of MHD activity localized near the internal transport barrier.

Research on the asymptotic matching analysis of resistive MHD stability has been performed. A new code MARG1D for solving the Newcomb equation in the ideal MHD region and computing the matching data was developed [2-3]. The Newcomb equation is solved as a boundary value/eigenvalue problem to which the finite element method can be applied. The extension of the MARG1D to the two dimensional toroidal problem is now under development.

The kinetic ballooning mode (KBM) at the internal transport barrier (ITB) with negative magnetic shear in a tokamak was analyzed numerically by using a kinetic shooting code. The eigenvalues (growth rates and real frequencies) of a KBM equation were calculated by carefully checking the convergence of solutions at large shooting distances. The second stability regime for negative magnetic shear, predicted by Hirose et al. [Hirose A., Elia M., Phys. Rev. Lett. **76** (1996) 628], was shown to disappear. The mode with comparatively low toroidal mode number and the real frequency ~100 kHz was found to be destabilized only around the ITB for the JT-60U parameters. These characteristics are consistent with the experimental observations of the mode inducing the mini β collapse in the vicinity of the ITB. The KBM is considered to be a possible candidate for the experimentally observed MHD activities, whereas lower frequency drift type modes might be responsible for the thermal transport [2-4].

References
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[2-4] Yamagiwa M., Hirose A., Elia M., " Kinetic Ballooning Modes at the Tokamak Transport Barrier with Negative Shear", Phys. Plasmas **4** (1997) 4031.

3. Divertor

A five-point model for the scrape-off layer (SOL) and divertor plasmas was developed to study the inside/outside divertor asymmetry induced by the divertor biasing [3-1]. Effects of divertor biasing on the asymmetry were studied for low and high recycling states. In the low recycling state, the biasing has a little influence on the asymmetry. On the other hand, in the high recycling state, the biasing substantially controls the asymmetry. The energy loss due to the ionization and impurity radiation plays an important role to cause the heat flux asymmetry. The divertor plasma has higher density, lower temperature and lower heat flux at the anode-side plate compared with those at the cathode-side plate.

This five-point model was used to study the characteristics of JT-60U divertor plasmas [3-2]. A high-recycling divertor is formed when the particle flux from the main plasma exceeds $(1\sim2)\times10^{22}$ s^{-1} for the safety factor of ~5 and the heating power of (1~20) MW. At the onset of the high recycling, the density at the SOL mid-plane is $n_{mid} \approx 0.5\times10^{19}$ m^{-3}.

References
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4. Numerical Experiment of Tokamak (NEXT)

4.1 Development of Computational Algorism

One of the main obstacles for the global particle simulation such as kinetic MHD instabilities in a tokamak is the large discrepancy between the characteristic time scale of the mode and the transit time of the electron motion parallel to the magnetic fields. To resolve such difficulties, a particle-fluid hybrid model for the simulation of the kinetic MHD instabilities was developed [4-1], which treats electrons as a fluid and retains the electron inertia effect. The model was applied to the nonlinear simulation of the m/n = 1/1 internal kink mode and was confirmed to agree well with the previously performed gyro-kinetic particle simulation, while only consuming 1/8 to 1/4 of the CPU time.

The above approach was extended to the electrostatic drift wave problem in a slab geometry and a new system of gyrokinetic Vlasov-Maxwell equations was derived [4-2]. In the formulation, the motion of the high energy transit electron is averaged over the periodic unperturbed orbit. The resultant equations for the high energy electrons involve only the E×B nonlinearity, and the adiabatic response to the low frequency fluctuation is renormalized in the field equation. The numerical experiments verified the efficiency of this simulation model.

4.2 Transport and MHD Simulation

In order to study the transport in a tokamak, a particle based global simulation code which has a full toroidal geometry was developed. To treat the delicate toroidal coupling problem under the weak/reversed magnetic shear configuration, the toroidal metric (r, θ, φ) with a nonuniform grid for the radial direction is employed and the electrostatic potential is solved via Fourier mode expansion both for poloidal/toroidal directions [4-3]. Recently, the code was developed so that the effect of the self-generated radial electric field, i.e. the (0,0) mode, which will be nonlinearly derived through the Reynolds stress, can be simulated. The (0,0) mode driven by semi-global "radial mode" of ITG instability which also shows the semi-global radial extent is excited. Such a (0,0) mode induces a fluctuating "zonal flow". The flow disintegrates the semi-global ITG vortices into small species and reduces the fluctuation level and transport [4-4].

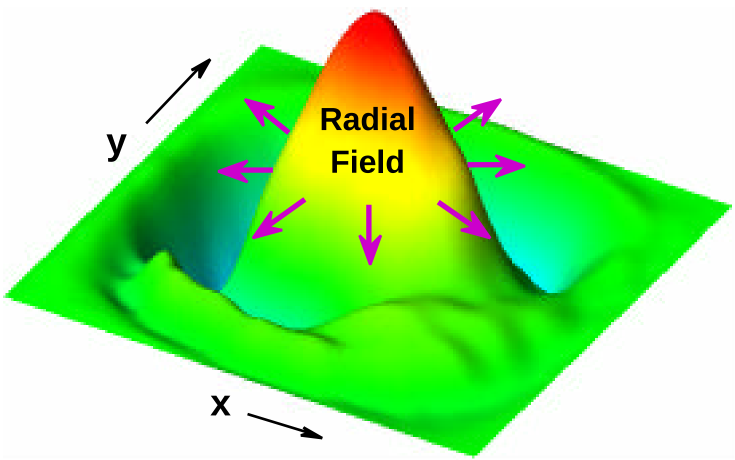

Fig.III.4.2-1 The electrostatic potential profile on a toroidal cross-section. m/n=0/0 mode (radial electric field) induced by the nonlinear interaction grows to the same level as the 1/1 mode.

We have been developing the non-linear MHD code using a full set of resistive MHD equations. To this end, we compared the linear results obtained by a linear version of the code to those obtained by the FAR code developed by Oak Ridge National Laboratory in USA. The eigenvalues and eigenfunctions calculated by both codes were in close agreement and showed that the finite beta effect is stabilizing on the toroidal tearing mode, while compressibility has little effect [4-5].

The effects of the density gradients on the kinetic m/n = 1/1 internal kink mode were investigated by the electromagnetic gyrokinetic-particle code. The first reconnection process was confirmed to be similar to that for the uniform density case. However, after the first reconnection, it was found that self-generated radial electric fields are induced by the nonlinear interaction, and the combination of the growth of 0/0 mode and the attenuation of 1/1 mode produces a vortex structure in the density profile [4-6]

4.3 Divertor Simulation

Simulation codes have been developed for the purpose of understanding physical processes in the divertor plasma. PARASOL is a particle code to verify the physical model for SOL plasmas, such as sheath conditions and heat transport. SOLDOR is a fluid code to predict the plasma parameter accounting for interactions with the neutral particles. IMPMC is a 2D Impurity Monte-Carlo code to analyze the impurity behavior in the divertor plasma. The assumptions widely used in impurity fluid codes, (i.e. instantaneous thermalization of impurity ions and simplified evaluation of self-sputtering outflux) was examined with the IMPMC code. It was found that they could not be applied for impurity ions with low charge states near the plates, and they lead to overestimation of the impurity influx into the main plasma [4-7].

4.4 Massively Parallel Computing

In the particle-fluid hybrid simulation, the computational cost for the electron fluid and electromagnetic fields is comparable to the cost of ion particle-pushing. High performance of the vector computation is required for the fluid equations and Maxwell's equations, while a large amount of memory is necessary for the ion particles [4-1]. Therefore, heterogeneous computing is an effective parallel computing technique for the hybrid simulation. We have demonstrated the efficiency of heterogeneous computing using the Hybrid3D code by connecting a vector parallel computer (VPP300) with a scalar parallel computer (SR2201); the fluid equations and Maxwell's equations are solved on the VPP300 and the equations of motion for ions are solved on the SR2201. The necessary performance of network between two computers was estimated, and it was shown that 10 parallel network system of 800 Mbytes HIPPI is sufficient for the present hybrid simulation. This research was performed in collaboration with the Center for Promotion of Computational Science and Engineering, JAERI.

References

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[4-4] Kishimoto Y. et. al., "Discontinuity Mode for Internal Transport Barrier in Reversed Magnetic Shear Plasma", 1998 International Sherwood Fusion Theory Conference, Mar 25, 1998, Atlanta, Georgia.

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[4-6] Matsumoto T., Tokuda S., Naitou H., "Density Gradient Effect for m = 1 Kinetic Internal Kink Mode", in Proc. of 1997 Workshop on MHD Computations - Numerical Methods and Optimization Techniques in Controlled Thermonuclear Fusion Research - (Institute of Statistical Mathematics, Tokyo, 1998) p.88.

[4-7] Shimizu K., Takizuka T., Sakasai A., "A Review on Impurity Transport in Divertors", J. Nucl. Mater. 241-243 (1977) 167.

IV. FUSION INTERNATIONAL COOPERATIONS

In the area of fusion research and development, Japan is recognized as one of the leading nations of the world together with Europe, USA and Russian Federation. Fusion reactor development is a long-term project which requires large resources both in man-power and in fund. It covers also broad area of science and technology. International cooperation has been recognized quite efficient in avoiding unnecessary duplication and in enhancing world's fusion program. JAERI is carrying out various international cooperation in fusion through multilateral cooperation under International Energy Agency (IEA) in Organization for Economics Cooperation and Development (OECD), International Atomic Energy Agency (IAEA), and bilateral cooperation such as Japan-US cooperation. The multilateral and bilateral cooperation carried out in JAERI are summarized in Table IV. 1-1 and IV. 1-2.

1. Multilateral Cooperations

1.1 IAEA

Under the coordination of International Fusion Research Council, IAEA holds various conferences such as the International Fusion Energy Conference and Technical Committee Meeting (TCM). IAEA also undertakes the Engineering Design Activity (EDA) in the ITER program.

1.2 IEA

Fusion Power Coordinating Committee (FPCC), which is organized under IEA, coordinates the research and development programs for member nations, selects the important areas and reviews the cooperation activities.

Cooperation under the IEA Implementing Agreement among the Three Large Tokamak Facilities pursues personal exchange, holding expert meetings and information exchange among JT-60, JET in EU, and TFTR in USA. Currently six tasks, namely "High-β_p Plasma Research", "Disruption Studies", "Divertor Plate Technology", "Neutral Beam Current Drive Research", "Remote Participation in Experiments" and "impurity Content of Radiative Discharges", have been successfully continued. In connection with the Remote Participation in Experiments, Provisional Guidelines for Remote Research on the JT-60 under Co-operation among the Three Large Tokamak Facilities were approved by the Committee. The collaboration among the Three Large Tokamak Facilities has made remarkable achievements and significant contributions to improving tokamak plasma performance and to providing sufficient basis for fusion energy development including ITER.

In the Implementing Agreement on Plasma Wall Interaction in TEXTOR, a plasma-wall interaction research cooperation is carried out utilizing the facility of the TEXTOR tokamak built in

Forschungszentrum Julich, Germany.

The agreement for cooperation on fusion materials research is investigate the irradiation damages by applying neutrons from a fission rector to fusion materials. In order to develop fusion materials after a prototype reactor, a conceptual design of a 14 MeV intense neutron source (Fusion Materials and Irradiation Test Facility : IFMIF) is carried out by for parties of Japan, USA, EU and Russia.

The agreement for cooperation on environments, safety and economics is to carry out their evaluation researches which are ongoing with particular emphasis upon environments and safety.

The agreement for cooperation on fusion reactor engineering is to carry out research cooperation and information exchange in terms of neutron engineering, tritium breeding blanket and so on.

Multilateral Cooperation	
IAEA	• ITER (International Thermonuclear Experimental Reactor) /EDA Project [Japan, USA, EC, Russia] • Information Exchange on Large Tokamaks • Information Exchange on Atomic and Molecular Data • International Conferences
IEA	• Three Large Tokamak Cooperation [JT-60(J), TFTR(US), JET(EU)] • Plasma Wall Surface Interaction Program [Japan, USA, EU, Canada] • Program of Research and Development on Radiation Damage in Fusion Reactor Materials [Japan, EU, Canada, Switzerland, USA] • Joint Program for Environmental, Safety and Economic Performance of Nuclear Fusion Technology [Japan, USA, EU, Canada] • Cooperative Program on Nuclear Technology of Fusion Reactors [Japan, USA, EU, Canada]

Table IV.1-1. Multilateral cooperation in fusion international cooperation at JAERI

2. Bilateral Cooperations

On Japan-US cooperation, Coordinating Committee of Fusion Energy (CCFE) is formed to synthetically coordinate the cooperation activities under Agreement between the government of Japan and the government of the United States on cooperation in Research and Development in Energy and Related Fields. The Japan-US cooperation consists of four frameworks of exchange program, joint program, joint project and plasma physics. In particular, broad joint projects based on agreements and annexes have produced fruit results, playing a leading role in world's fusion research and development.

On Japan-EU cooperation, Agreement for Cooperation between the Government of Japan and the European Atomic Energy Community in the field of controlled thermonuclear fusion was concluded February 1988. Based on this agreement, a joint experiment is carried out in which lower hybrid (LH) wave launcher module built at JAERI are installed into the LH test facility in Cadarache Institute.

With Canada, JAERI carries out information exchange and expert meeting on tritium technology and tokamak research through Atomic Energy Canada Ltd. (AECL). With Australia, information exchange and expert meeting are carried out by holding workshops mainly in the area of diagnostics, experiment and theory for toroidal plasmas. With Russia, information exchange and expert meeting on plasma and fusion are planned under Agreement between the government of Japan and the government of Russia in Research and Development in Science and Technology.

Bilateral Cooperation	
Japan-US	• Doublet III Project • HFIR Joint Irradiation Experiment Program • Fusion Fuel Processing Technology Development Program • Cooperation in Fusion Research and Development • Data Link Program
Japan-EU	• Cooperative Activities Concerning a lower Hybrid Antenna Module
Japan-Canada	• Cooperation in the Field of Controlled Nuclear Fusion
Japan-Australia	• Cooperation on Diagnostics, Experiments and Theory
Japan-Russia	• Cooperation in Fusion Research and Development

Table IV.1-2. Bilateral cooperation in fusion international cooperation at JAERI

3. Cooperative Program on DIII-D (Doublet III) Experiment

3.1 Highlights of FY 1997 Research Results

Integration of the advanced tokamak concept, which encompasses to acquire the improved confinement in a steady-state with high normalized-beta and a large non-inductive current fraction, and the highly-evolved divertor functions was emphasized in the 1997 experimental campaign at DIII-D. Accordingly, the H-factor of 2 was sustained for 2 seconds in a reversed shear plasma with ELMs under the divertor pumping, by means of the effective modification of edge pressure gradient, mainly in terms of the shaping and edge density control . Furthermore, based on the highest performance discharge in 1996, which recorded the equivalent QDT of 0.32, the product of normalized beta and H-factor was sustained at 7 over 1 s. Intensive studies on the physics of internal transport barrier were also pushed forward.

As to the development of advanced divertor, the geometry and pumping capabilities are modified in multiple steps i.e., the baffle plates and pumps will be installed both at upper and lower divertor in 1999. In the baffled upper pump experiment performed in 1997, high density H-mode plasmas at a density of 1.5 times the Greenwald limit was obtained with the low-field side pellet injection as shown in Fig. IV.3.1-1, which made direct contribution to the ITER Physics R&D. In addition, RI-mode operation was first undertaken, and normalized beta of 4 and H-factor of 3 to 4 were simultaneously obtained at 60% of the Greenwald density.

On the other hand, studies of ECCD experiment was remarkably progressed in 1997. The effective heating and central current drive was demonstrated with 2 MW of 110 GHz EC waves.

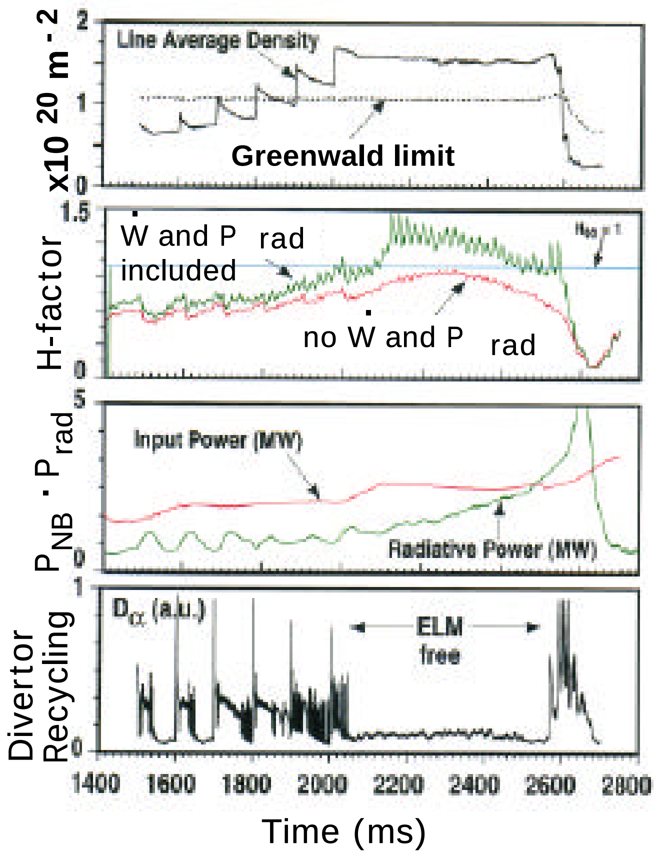

Fig. IV.3.1-1 Discharge waveforms of RI-mode

4. Collaborative Activities Concerning Fusion Technologies

4.1 Collaborative Activities on Environmental, Safety, and Economics Aspects of Fusion Power

Designated by the Government of Japan, JAERI has been participating in the IEA Implementing Agreement on a Cooperative Programme on the Environmental, Safety and Economic (ESE) Aspects of Fusion Power. This collaborative activity is carried out by Canada, EURATOM, JAERI, MINATOM and USA since 1992 and extended for another five years in 1997. JAERI is coordinating the tasks on Transient Thermofluid Modeling and Validation Tests and Safety System Study Methodology. Two new tasks on "Socio-Economics Aspects of Fusion Power" and on "Radioactive Waste from Fusion Power" are being considered for collaboration.

4.2 Collaborative Activities on Research and Development of Plasma Wall Interaction in TEXTOR

An IEA implementation for a collaboration program of research and development on plasma wall interaction in TEXTOR is expected up to December 2002. TEXTOR management is under KFA (Forschungszentrum Juelich GmbH), ERM/KMS (Ecole Royale Militaire) Brussels and FOM (Stichting voor Fundamenteel Onderzok der Materie) Nieuwegein under the TEC (Trilateral Euregio Cluster). Japan is a member of the executive committee and NIFS organizing Japanese programs as a cooperation center of Japan. JAERI has been joined the program as a Japanese technical committee member. In this fiscal year, four staff members visited TEXTOR to exchange informations on the design of dynamic ergodic divertor and on plasma edge diagnostics and discuss experimental results.

4.3 Collaborative Activities on Technology for Fusion-Fuel Processing between US-DOE and JAERI

Research and development of technology for Fusion Safety has been carried out at the Tritium Systems Test Assembly (TSTA) of Los Alamos National Laboratory since 1995.

Following the first experiment carried out in FY 1996, the second and third tritium release experiments, which were associated with a 1 Ci tritium release, were carried out to obtain data such as, 1) diffusion process of tritium in a room, 2) conversion rate of tritium gas to tritiated water, 3) tritium behavior when the ventilation system runs for tritium removal from room, and 4) adsorption of tritium on wall materials, etc. The experiments were successfully carried out on December 16, 1997 and March 3, 1998 [4.3-1].

Beta Scintillation Detector (BSD) is a newly proposed technique for measuring total tritium concentration in gases. A new BSD was installed at TSTA to explore this technique. As a result of a series of experiments using various kinds of tritium gas mixture with other hydrogen isotopes, helium and nitrogen, the response of the BSD to tritium was found to be almost unaffected by presence of other gases. This shows that the BSD is a particularly promising technique for tritium accountancy in fusion fuel processing.

For decontamination study, measurement of adsorption isotherms of hydrogen isotopes, particularly pure tritium on Molecular Sieve 5A (MS5A), Molecular Sieve 4A (MS4A) and Activated Carbon (AC) at liquid nitrogen temperature (77 K) was carried out using the constant volume method, in which a measured amount of tritium was adsorbed on the sample stepwise and equilibrium pressure was measured as a function of adsorbed amount. Obtained result with MS5A is shown in Figure VII.4.3-1. The adsorption isotherms of pure tritium on MS5A, MS4A and AC at 77 K can be expressed with the two sites Langmuir equation, and the Langmuir coefficients were obtained as the functions of the reduced mass. These data will be utilized effectively in development of the blanket tritium recovery system and the helium glow discharge exhaust gas cleanup system.

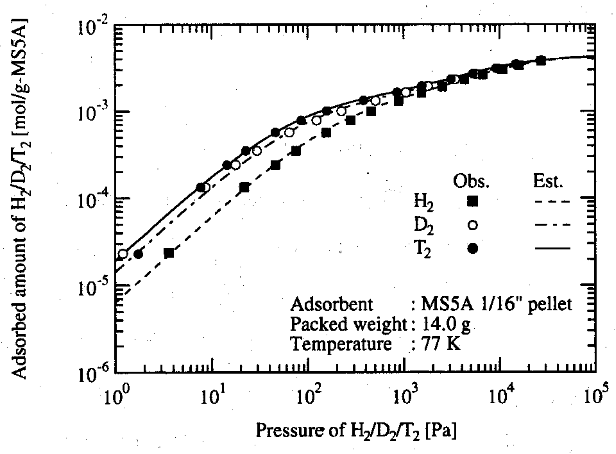

Fig. VII.4.3-1 The adsorption isotherms of H_2, D_2 and T_2 on MS5A at 77 K.

References

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4.4 Collaborative Activities on Research and Development of Plasma Facing Components between US and Japan

The JAERI-SNL collaborative activities on the divertor mock-ups have been carried out under the US-Japan Fusion Cooperation Program. In FY 1997, a critical heat flux (CHF) experiment and a thermal cycling test of a Be divertor mock-up were performed in Sandia National Laboratory (SNL). The objective of the experiment is to clarify heat transfer behavior of a cooling tube for the divertor plate over CHF region (post-CHF regime). It was found that the post-CHF regime was clearly appeared in lower flow velocity (~ 1 m/s). The Be divertor mock-up was developed at JAERI. Be armor tiles were bonded onto a Cu heat sink with a HIP method. In the thermal cycling test, a surface heat flux of 3 MW/m^2 was cyclically loaded on the mock-up to evaluate the thermal fatigue behavior. The mock-up successfully withstood a heat load of 3 MW/m^2, 10 s for 1,000 thermal cycles without failure.

4.5 Collaboration between JAERI and CEA-Cadarache for Lower-Hybrid Antenna Modules

Cooperative activities have been started to obtain a detail outgassing database during a high power and a long pulse RF operation for a launcher design in a future LHCD system from 1993. RF power test was performed at CEA-Cadarache RF Test Facility which allowed high power injection up to 500 kW, under quasi-continuous operation at a frequency of 3.7 GHz. In the first step, the outgassing rate of Dispersion Strengthened Copper waveguide module during RF injection was identified. In the second step of the collaboration from 1995, outgassing rate with mouth modules made of Carbon Fiber Composite (CFC) has been measured to develop heat resistant LH antenna front and the aimed data base was obtained. It was concluded that no external pumping for the antenna is necessary with the appropriate antenna design.

4.6 Collaborative Activities on Research and Development of Plasma Facing Components between EU and Japan

Under the Japan-EURATOM Fusion Agreement, two collaborative activities have been performed. One is the JAERI-CEA collaboration and the other is the JAERI-KFA collaboration. The critical heat flux (CHF) benchmark experiment was performed at CEA. To investigate the influence of the heat flux profile upon CHF, the experiments were carried out with a flat profile and a peak profile which simulates the real heat flux on the ITER divertor plate. It was turned out that the incident CHF with the ITER modified profile is 20 ~ 40 % higher than that with flat profile.

High heat flux experiments on mock-ups developed by KFA were performed at JAERI. In the high heat flux tests of CFC/Cu and B_4C/TZM mock-ups, the surface heat flux was stepwise increased to evaluate critical performance. Both mock-ups successfully withstood up to heat loads of 12 ~ 15 MW/m^2, 15 s.

4.7 Collaborative Activities on Technology for Tritium Transfer between AECL and JAERI

The objectives of cooperation in the field of controlled nuclear fusion between JAERI and Atomic Energy of Canada Limited (AECL) are to conduct information and personnel exchanges to develop fusion technologies on tritium handling, breeding blanket, and plasma physics. In 1997, a technical meeting was held at the Tritium Process Laboratory, JAERI to discuss the technical items for loading and shipping of tritium from Chalk River Laboratory of AECL. Procedures for accountancy and calibration were discussed and general information was exchanged for mutual understanding of this program. The third shipment of tritium is planned in 1998 based on the contract for purchasing tritium for research and development of tritium handling technology for fusion, which was renewed in March 1998.

5. Other Activities

The mutual information and personal exchanges between JAERI and fusion research institutes in Asian area are rapidly increasing during this several years under significant development on fusion research in this area, especially in China and Korea. These exchanges are performed under STA scientist exchange program (in 1997, three scientists from China for one year and three JAERI scientists to China for two weeks), the scientist invitation program (in 1997, one senior scientist from China for one month), STA and JAERI fellowships and so on. A new framework to make more fruitful cooperation between JAERI and these countries on fusion research field should be prepared under Science and Technology Cooperation Agreement between Japan and these countries.

APPENDIX

A.1 Publication List (April 1997 - March 1998)

A.1.1 List of JAERI reports

1) Ajima T., Kurihara R., Seki Y., et al., JAERI-Data/Code 97-034, "Improvement of TRAC-BF1 Code to Analyze the Ingress of Coolant Event (ICE)", (1997) (in Japanese).

2) Aoki I., Seki Y., Sasaki M., JAERI-Data/Code 97-042, "Development of Dynamic Simulation Code for Fuel Cycle of Fusion Reactor (1. Single Pulse Operation Simulation)", 1997) (in Japanese).

3) Aoyagi T., Sato M., Sakata S., et al., "Development of New CICU", JAERI Tech 97-073 (1997).

4) Furuya K., Sato S., Hatano T., et al., JAERI-Tech 97-022, "Design of ITER Shielding Blanket", (1997).

5) HatanoT., Sato S., Fukaya K., et al., JAERI-Research 97-017, "High Heat Flux Testing of HIP bonded DS-Cu/316SS First Wall Panel for Fusion Experimental Reactors", (1997).

6) Hayashi T., Miya N., Kikuchi M., et al., JAERI-Research 97-007, "The Design Study of the JT-60U Device (No. 9) - Fuel Handing, Confinement and Clean Up System for JT-60U", (1997) (in Japanese).

7) Hiranai S., Yokokura K., Moriyama S., et al., "development of Temperature Measuring System of ICRF Antenna for JT-60", JAERI-Tech 98-006 (1998).

8) Ikeda Y., Ushikusa K., Seki M., et al., "ECR Discharge Cleaning on JT-60U", JAERI-Research 97-075 (1997).

9) Ishii Y., "Ideal MHD Stability in Negative Shear Plasma", JAERI-Research 97-047, Review of JT-60U Experimental Results from February to November,1996 , p.14-17.

10) Kanari M, Hatano T., Sato S., et al., JAERI-Research 98-004, "Characterization of HIP Bonded DS-Cu/SS316L Joints for Fusion Experimental Reactors", (1998).

11) Kurihara K. and JT-60 Control Group, "Improvement of Equilibrium Control Algorithm", in "Review of JT-60U Experimental Results (Feb. to Nov.," JAERI-Research 97-047(1997) pp.129-132.

12) Kurihara K., Kawamata Y., Akiba K., et al., " Development of the Real-time Plasma Shape Visualization System in JT-60," JAERI-Research 97-055 (1997) (in Japanese).

13) Kurihara K., "A Study on Current Density Distribution Reproduction by Bounded-Eigenfunction Expansion for a Tokamak Plasma," JAERI-Research 97-084 (1997) (in Japanese).

14) Kurihara K. and Kawamata Y., "Development of a Precise Long-time Digital Integrator for Magnetic Measurements in a Tokamak," JAERI-Research 97-072 (1997) (in Japanese).

15) Maki K., Sato S., Kawasaki H., JAERI-DATA/Code 97-002, "Development of Displacement Cross Section Set for Evaluating Radiation Damage by Neutron Irradiation in Materials Used for Fusion Reactors", (1997).

16) Matsukawa M., Aoyagi T. and Miura Y. M., "General Tokamak Circuit Simulation Program GTCSP," JAERI-DATA/Code 97-017 (1997) (in Japanese).

17) Miura H., Sato S., Enoeda M., et al., JAERI-Tech 97-051, "Design of Test Blanket System for ITER Module Testing", (1997).

18) Miura Y. M., Matsukawa M., Miyachi K., et al., "Development of a Current-type PWM Converter with High Power Factor - I," JAERI-Tech 98-001 (1998) (in Japanese).

19) Miya N., Kikuchi M., Ushigusa K., et al., "The Design Study of the JT-60SU Device (No. 8) -Nuclear Shielding and Safety design-", JAERI-Research 98-012 (1998).

20) Nagaya S., Onizawa M., Kawai I., et al., "Assessment and counterplan for JT-60U spectroscopic system", JAERI-Tech 97-062(1997).

21) Neudatchin S.V., Takizuka T., Shirai H., et al., "Analysis of Space-time Structure of Internal Tansport Barrier in JT-60U", JAERI-Research 97-052 (1997) .

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A.2 Scientific Staffs in the Naka Fusion Research Establishment (April, 1997 - March, 1998)

Naka Fusion Research Establishment

KISHIMOTO Hiroshi (Director General)

OHKAWA Tihiro (Scientific Consultant)
SEKIGUCHI Tadashi (Scientific Consultant)
TANAKA Yuji (Scientific Consultant)
MIYAMOTO Kenro (Invited Researcher)
KAWASAKI Sunao (Invited Researcher)
SHIMAMOTO Susumu (Invited Researcher)
TOMABECHI Ken (Invited Researcher)

Department of Administrative Services

KOMAKI Akira (Director)

Department of Fusion Plasma Research

AZUMI Masafumi (Director)
NAGAMI Masayuki (Deputy Director)
TAKAHASHI Ichiro (Administrative Manager)
SHIMADA Michiya

Tokamak Program Division

NAGAMI Masayuki (General Manager)

IDE Shunsuke	ISHIDA Shinichi	KITAI Tatsuya (*15)
KURITA Gen-ichi	MORI Katsuharu (*15)	NAGASHIMA Keisuke
NAKAGAWA Shouji (*15)	OGURI Shigeru (*15)	TOYOSHIMA Noboru
USHIGUSA Kenkichi		

Plasma Analysis Division

KIKUCHI Mitsuru (General Manager)

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KOIWA Motonao (*31)	MATSUDA Toshiaki	NAITO Osamu
NAKAMURA Yukiharu	NEUDATCHIN Sergei V. (*23)	
OHSHIMA Takayuki	POLEVOI Alexei (*11)	SAITO Naoyuki
SAKATA Shinya	SATO Minoru	SHIMIZU Katsuhiro
SHIRAI Hiroshi	SUZUKI Mitsuhiro (*33)	TAKIZUKA Tomonori
TSUGITA Tomonori		

Large Tokamak Experiment Division I

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CHIBA Shinichi	FUKUDA Takeshi	HAMANO Takashi
ISAYAMA Akihiko	ISEI Nobuyuki	INOUE Akira
IWASE Makoto (*36)	KAMADA Yutaka	KASHIWABARA Tsuneo
KAWANO Yasunori	KITAMURA Shigeru	KOKUSEN Shigeharu
KRAMER Gerrit Jakob (*41)	KUSAMA Yoshinori	MENG Yuedong (*8)
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NEMOTO Masahiro	NEYATANI Yuzuru	NISHITANI Takeo
OIKAWA Toshihiro	SAKUMA Takeshi	SHITOMI Morimasa
SUNAOSHI Hidenori	TAKEJI Satoru	
TCHERNYCHEV Fedor Vsevodovich (*9)		TOBITA Kenji
TSUCHIYA Katsuhiko	TSUKAHARA Yoshimitsu	UEHARA Kazuya
URAMOTO Yasuyuki	YOSHIDA Hidetoshi	ZHAO Junyu (*8)

Large Tokamak Experiment Division II

YOSHINO Ryuji (General Manager)

ASAKURA Nobuyuki
DaCOSTA Olivier (*2)
FUJITA Takaaki
HATAE Takaki
HIGASHIJIMA Satoru
HOSOGANE Nobuyuki
ITAMI Kiyoshi
KONDOH Takashi
KONOSHMA Shigeru
KOOG Joong San (*36)
KUBO Hirotaka
NAGASHIMA Akira
SAKASAI Akira
SAKURAI Shinji
SHINOHARA Kouji
SUGIE Tatsuo
SUZUKI Shingo (*36)
TAKENAGA Hidenobu

Plasma Theory Laboratory

HIRAYAMA Toshio (Head)

DETTRICK Sean (*41)
HUDSON Stuart (*41)
HAYASHI Mitsuru (*36)
ISHII Yasutomo
ISHIZAWA Akihiro (*36)
OZEKI Takahisa
MATSUMOTO Taro
SUGAHARA Akihiro (*31)
TOKUDA Shinji
TUDA Takashi
YAMAGIWA Mitsuru

Experimental Plasma Physics Laboratory

KIMURA Haruyuki (Head)

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KAWAKAMI Tomohide
KAWASHIMA Hisato
LIU Wandong (*49)
MAEDA Mitsuru (*14)
MAENO Masaki
MIURA Yukitoshi
OGAWA Hiroaki
OGAWA Toshihide
OASA Kazumi
SATO Masayasu
SENGOKU Seio
SHIINA Tomio
YAMAUCHI Toshihiko

Department of Fusion Facility

FUNAHASHI Akimasa (Director)
SHIMIZU Masatsugu (Deputy Director)

Fusion Facility Administration Division

TAKAHASHI Ichiro (General Manager)

JT-60 Facility Division I

KIMURA Toyoaki (General Manager)

ADACHI Hironori (*25)
AKASAKA Hiromi
ARAKAWA Kiyotsugu
FUKUDA Hiroyuki (*15)
FURUKAWA Hiroshi (*32)
KAWAMATA Youichi
KURIHARA Kenichi
MATSUKAWA Makoto
MIURA M. Yushi
NOBUSAKA Hiromichi (*15)
OKANO Jun
OMORI Shunzo
OMORI Yoshikazu
OOBA Toshio (*32)
SEIMIYA Munetaka
SHIMONO Mitsuru
TAKANO Shoji (*33)
TERAKADO Tsunehisa
TOTSUKA Toshiyuki

JT-60 Facility Division II

SAIDOH Masahiro (General Manager)

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HIRATSUKA Hajime
HONDA Masao
ICHIGE Hisashi
KAMINAGA Atsushi
KODAMA Kozo
KOMURO Ken-ichi (*28)
MASAKI Kei
MIYATA Hiroshi (*6)
MIYO Yasuhiko
MORIMOTO Masaaki (*27)
OKABE Tomokazu
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SANTO Masahide (*6)
SASAJIMA Tadayuki
SASAKI Noboru (*6)
TAKAHASHI Shoryu (*6)
YAGYU Jun-ichi

RF Facility Division

YAMAMOTO Takumi (General Manager)

ANNOU Katsuto
IKEDA Yoshitaka
ISAKA Masayoshi

ISHII Kazuhiro (*32)
HIRANAI Shinichi
HIROI Toshikazu (*42)
KAJIYAMA Eiichi (*28)
KIYONO Kimihiro
MORIYAMA Shinichi
SEKI Masami
SHINOZAKI Shin-ichi
TERAKADO Masayuki
YOKOKURA Kenji

NBI Facility Division
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AKINO Noboru
EBISAWA Noboru
HONDA Atsushi
ITOH Takao
KAWAI Mikito
KAZAWA Minoru
MOGAKI Kazuhiko
OHGA Tokumichi
OHSHIMA Katsumi (*28)
OHMORI Ken-ichiou
OOHARA Hiroshi
SEKI Hiroshi (*32)
TAKENOUCHI Tadashi (*47)
TANAI Yutaka (*32)
TOYOKAWA Ryoji(*28)
USUI Katsutomi
YAMAZAKI Haruyuki (*6)
ZHOU Capin(*40)
GRISHAM Larry (*37)
HU Liquen (*8)

JFT-2M Facility Division
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YAMAMOTO Masahiro (Deputy General Manager)
HASEGAWA Koichi
KASHIWA Yoshitoshi
KIKUCHI Kazuo
KOMATA Masao
OKANO Fuminori
SAWAHATA Masayuki
SHIBATA Takatoshi
SUZUKI Sadaaki
TANI Takashi
UMINO Kazumi (*32)

Department of Fusion Engineering Research
OHTA Mitsuru (Director)
NAGASHIMA Takashi (Deputy Director)
MURASAWA Michihiko (Administrative Manager)

Blanket Engineering Laboratory
TAKATSU Hideyuki (Head)
ABE Tetsuya
ENOEDA Mikio
FURUYA Kazuyuki
HARA Shigemitsu (*6)
HATANO Toshihisa
KASAI Satoshi
KANARI Moriyasu (*36)
KIKUCHI Shigeto (*48)
KURODA Toshimasa (*19)
NAKAMURA Jyun-ichi (*35)
SATO Satoshi
YANO Atsushi (*35)

Superconducting Magnet Laboratory
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ANDO Toshinari
AZUMA Katsunori (*6)
HAMADA Kazuya
HANAWA Hiromi (*32)
HIYAMA Tadao
ISONO Takaaki
ISHIO Koutarou (*13)
KATO Takashi
KAWANO Katsumi
KOIZUMI Norikiyo
MATSUI Kunihiro
NAKAJIMA Hideo
NUNOYA Yoshihiko
OSHIKIRI Masayuki (*32)
SAWADA Kenji (*26)
SEKI Syuichi (*32)
SHIMBA Toru (*10)
SUGIMOTO Makoto
TAKAHASHI Yoshikazu
TAKANO Katsutoshi (*32)
TANEDA Masanobu (*20)
WAKABAYASHI Hiroshi (*32)
YAMAMOTO Kazutaka (*48)

NBI Heating Laboratory
OKUMURA Yoshikazu (Head)
AKIBA Masao
BANDOURKO Vassi (*11)
BOSCARY Jean (*41)
DAIRAKU Masayuki
EZATO Koichiro (*36)
FUJIWARA Yukio
GILANYI Attila (*36)
HANADA Masaya
JIMBOU Ryutaro (*6)
MIYAMOTO Naoki (*30)
MIYAMOTO Kenji
NAKAMURA Kazuyuki
SAWAHATA Osamu (*32)
SUZUKI Satoshi
SUZUKI Takayuki (*6)
WATANABE Kazuhiro
YOKOYAMA Kenji

RF Heating Laboratory

IMAI Tsuyoshi	(Head)	
IKEDA Yukiharu	KASUGAI Atsushi	KATO Yasushi (*32)
KOARAI Tohru (*32)	MAEBARA Sunao	NUMATA Hideyuki (*29)
SAKAMOTO Keishi	SHIHO Makoto	TAKAHASHI Koji
TSUNEOKA Masaki	WATANABE Akihiko (*29)	ZHENG Xaodong (*36)

Tritium Engineering Laboratory

NISHI Masataka	(Head)	
ARITA Tadaaki (*42)	HAYASHI Takumi	ISHIDA Toshikatsu (*19)
ISOBE Kanetsugu (*24)	ITO Takeshi (*17)	IWAI Yasunori
KAKUTA Toshiya (*19)	KAWAMURA Yoshinori	KOBAYASHI Kazuhiro
MARUYAMA Tomoyoshi (*27)		NAKAMURA Hideki (*48)
NAKAMURA Hirofumi	O'HIRA Shigeru	SHU Weimin
SUZUKI Takumi	TADOKORO Takahiro (*6)	YAMADA Masayuki
YAMANISHI Toshihiko		

Reactor System Laboratory

SEKI Yasushi	(Head)	
AOKI Isao	AJIMA Toshio (*6)	KURIHARA Ryoichi
NISHIO Satoshi	UEDA Shuzo	

Reactor Structure Laboratory

TADA Eisuke	(Head)	
AKOU Kentaro (*19)	ITOU Akira (*10)	KAKUDATE Satoshi
NAKAHIRA Masataka	OBARA Kenjiro	OKA Kiyoshi
TAGUCHI Kou (*32)	TAKAHASHI Hiroyuki (*6)	
TAKIGUCHI Yuji (*48)	TAKEDA Nobukazu	

Department of ITER Project

MATSUDA Shinzaburo	(Director)
SEKI Masahiro	(Prime Scientist)
SHIMOMURA Yasuo	(Prime Scientist)
FUJISAWA Noboru	

Administration Group

SHOJI Kuniaki	(Leader)

Project Management Group

SEKI Shogo	(Leader)

Joint Central Team Group

SEKI Shogo	(Leader)	
ANDO Toshiro	EBISAWA Katsuyuki (*48)	HATTORI Yukiya (*6)
HIROKI Seiji	HORIKIRI Hitoshi (*39)	HOSHI Yuichi (*10)
IIDA Fumio (*6)	IIDA Hiromasa	IIZUKA Takayuki
IOKI Kimihiro (*27)	INOUE Takashi	ITOH Kazuyoshi (*42)
ITOH Mitsuyoshi (*10)	KATAOKA Yoshiyuki (*6)	KAWAI Shigetaka (*26)
KOBAYASHI Noriyuki (*48)	KODAMA Tetsuhiko (*27)	MARUYAMA So
MATSUMOTO Hiroshi	MIZOGUCHI Tadanori (*6)	MOHRI Kensuke (*19)
MORIYAMA Kenichi (*43)	MITA Yoshiyuki (*34)	MIKI Nobuharu (*48)
NAKASHIMA Yoshitane (*10)	NAKAMURA Hiroo	OIKAWA Akira
OKUNO Kiyoshi	ONOZUKA Masanori (*27)	OSANO Katsuharu (*5)

OZAKI Fumio (*48) SAJI Gen SATO Kouichi (*1)
SHIBANUMA Kiyoshi SUGIHARA Masayoshi
TAKAHASHI Kenji (*27) TAKIGAMI Hiroyuki (*48) TANAKA Shigeru
YAMADA Masao (*27) YAMAMOTO Shin YONEKAWA Izuru
YOSHIDA Hiroshi YOSHIDA Kiyoshi
YOSHIMURA Kunihiro (*48)

Home Team Design Group
TSUNEMATSU Toshihide (Leader)
ARAKI Masanori HASHIMOTO Masayoshi (*10) ITOH Yutaka (*6)
KITAMURA Kazunori KOIZUMI Koichi MIURA Hidenori (*19)
MIYAMOTO Masanori (*18) ODAJIMA Kazuo OHMORI Jyunji (*48)
OHNO Isamu (*10) OHKAWA Yoshinao OZAWA Yoshihiro (*6)
SENDA Ikuo (*48) SHIRAI Tetsuo (*44) SHOJI Teruaki
TADO Shigeru (*26) TAKASE Haruhiko (*48) YAGENJI Akira (*4)

Safety Evaluation Group
INABE Teruo (Leader)
ARAKI Takao (*48) MARUO Takeshi MITSUI Jin (*42)

*1 Atomic Data Service Corp.
*2 Ecole Polytechnique (France)
*3 Fuji Electric Co., Ltd.
*4 Hazama-gumi Ltd.
*5 Hitachi Information Systems, Ltd.
*6 Hitachi Ltd.
*7 Hitachi Nuclear Engineering Co., Ltd
*8 Institute of Plasma Physics Academia Sinica (China)
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*31 Research Organization for Information Science Technology
*32 Nuclear Engineering Co., Ltd.
*33 Nuclear Information Service Co.
*34 Obayashi Corp.

*35 Osaka Vacuum Ltd.
*36 Post-Doctoral Fellow
*37 Princeton Plasma Physics Laboratory (USA)
*38 Shimizu Corparation
*39 Shinryo Corporation
*40 Southwestern Institute of Physics (China)
*41 STA Fellowship
*42 Sumitomo Heavy Industries, Ltd.
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