SUMMARY

Fusion Technology, Safety and Environmental Aspects

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1. Introduction

The year 2002 was in the middle of successive governmental negotiation toward the start of the ITER Construction. The ITER Engineering Design Activities (EDA) continued until July 2001, and most of the highlighted topics were already reported at the last IAEA Fusion Energy Conference in Montreal or in other opportunities. However, the ITER EDA was followed by the Coordinated Technical Activities that provided a lot of qualitative achievements such as, the search for predictions on operation capabilities based on various data bases and analysis, optimization of the design based on its validating technology R&D.

As a consequence, at this conference, major contribution in the field of Fusion Technology was again from ITER, and its related topics occupied about 38% of the total number of contributions of 86. In ITER, physics analysis, predictions and heating/current drive technologies are highlighted.

Another key feature at this conference was the progress of study toward steady-state operation in both physics and technology research as well as their application to toroidal devices. Several tokamaks and helical devises are under construction or under design, and most of them incorporate super-conducting magnet for their coils.

Studies were made for various types of fusion reactors including Spherical Torus, Tokamaks, Helical systems etc., and their common understandings are progressing through their comparative study.

Looking in the near term, but beyond ITER, about 20% of the papers were devoted to the fusion materials and blanket development, with the neutron irradiation facilities for the research. Because of the importance of this field to be implemented in parallel with ITER, more contributions would be expected in future.

With these themes in mind, the remaining sections of this paper are arranged in the order of 2)ITER, 3)Toroidal Devices under Construction or under Design, 4)Reactor Technology, 5) Safety and Environment, and 6) Conclusion.

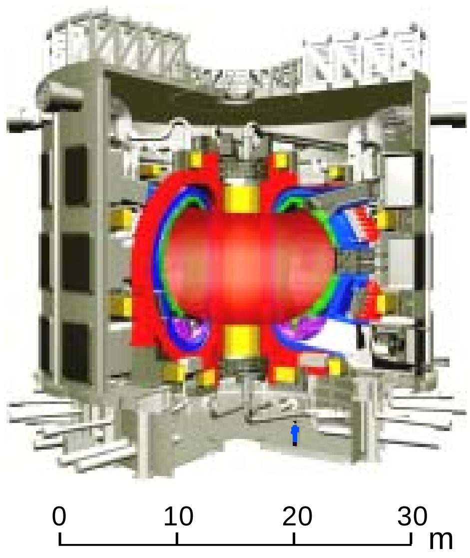


FIG.1. Cutaway view of ITER

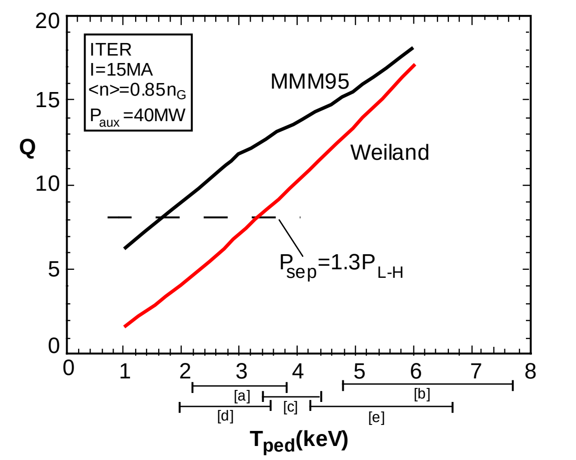
2. ITER

This chapter deals with not only ITER physics performance analysis, prediction, design and technology development, but also some technological activities for other purposes.

2.1. ITER Physics Design and Analysis

Confinement performance analysis based on empirical scaling demonstrates the feasibility of achieving $Q \geq 10$ in inductive operation, which is supported by dimensionless scaling analysis and theory-based transport modelling. Experiments of ASDEX-Upgrade, DIII-D and JET indicate that with peaked density profiles it is possible to maintain good global energy confinement ($H_{H98(y,2)} \sim 1$) up to and beyond the Greenwald density even though the loss power does not exceed the H-mode threshold power scaling. DT-helium elastic scattering, which was spectroscopically confirmed in JT-60U, was incorporated in divertor modelling, showing that the helium exhaust efficiency can be improved by a factor of 3-5, which results in significant improvement in core performance, e.g. Q = 10 to 14. Theory-based core modelling indicates the need of high pedestal temperatures (3.4 - 4.5 keV) to achieve $Q \geq 10$, which is in the range of projection with presently available pedestal scalings (FIG. 2).

There remains large uncertainty in the prediction of both the ITER pedestal temperature and ELM energy loss. The peeling-ballooning model appears to fit the variation of the edge pressure gradient with shape and other parameters, but understanding of the H-mode pedestal width scaling is still far from complete. However, ITER requires a transport barrier that is no larger a fraction of the minor radius than is common on existing machines. A scaling of pedestal pressure compared against experimental data in International Pedestal Database v. 3.



[a] MHD limit prdestal model, J.G.CORDEY, et al., this Conference, CT/R-02
[b] Thermal conduction pedestal model, J.G.CORDEY, et al., this Conference, CT/R-02
[c] MHD limit prdestal model, M.SUGIHARA, et al., Nucl. Fusion 40 (2000) 1743
[d] MHD limit prdestal model, A.KRITZ, et.al., EPS-29 (Montreux, 2002) D-5.001
[e] MHD limit prdestal model, M.SUGIHARA, et.al., 57th Annual Meeting of Phys. Soc. Japan (2002)

FIG.2. Q versus T_{ped} predicted for ITER by the Multi-Mode and Weiland models. Dashed line shows a value of Q compatible with P_{sep} = 1.3 x P_{L-H} , and horizontal bars show the ranges of T_{ped} predicted for ITER by different pedestal scalings.

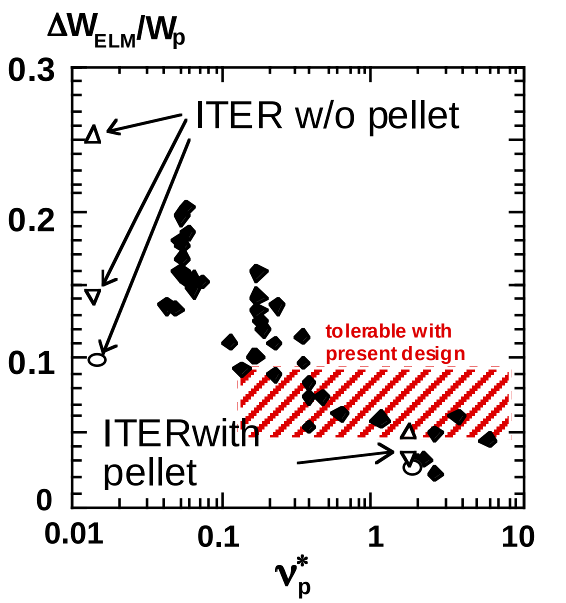


FIG.3. Dependence of fraction of pedestal energy loss $\Delta W_{ELM}/W_p$ on the pedestal collisionality ν^*_p. Experimental points are shown by closed diamonds together with predictions for ITER (open points) with and without pellet.

This scaling project a temperature of 5.3 keV for a pedestal density of 7 x 10^{19} m^{-3}, suggesting that Q ≥ 10 is achievable. More efforts on data analysis and theoretical work are called for to better understand the pedestal and ELM physics.

A scaling of the energy deposition time derived from target IRTV suggests that the convection is important for ELM heat transport to the divertor. Pellet injection from the high-field side would be useful in enhancing Q and reducing ELM heat load (FIG.3). The heat load of type-I ELM in high plasma current operation could be made tolerable by high-density operation and further tilting the target plate (if necessary).

The extensive divertor plasma modelling effort has led to an efficient power law parameterization of the plasma and neutral parameters at the interface between the edge and core plasma, making it finally possible to model the ITER core plasma performance in a way consistent with the divertor parameters. A variation of the x-point position offers some control over the achievable separatrix density. However, bifurcation of the divertor plasma parameters might limit the utility of this control method. More work is needed to elucidate the exact nature of this bifurcation and its relevance to the performance of the ITER divertor. The scalings of divertor parameters were incorporated into an integrated modelling of core, pedestal and divertor, confirming an inductive operation regime with Q = 12-16 and target heat load less than 10 MW/m^2.

On the erosion dominated areas (in the LFS divertor of most devices), experimental results indicate weak flux dependence, but no definitive conclusion can be drawn on carbon chemical erosion yields at high flux density in ITER. On the redeposition-dominated areas, contribution of high hydrocarbons to the total sputtered carbon atoms was found to dominate, and a large influence of surface temperature on the chemical yields was observed.

JET experiments show a strong sensitivity of the Toroidal Alfven Eigenmodes (TAE) damping rates to the shape and the magnetic shear. These studies confirm the theoretical prediction of strong stabilizing effect of the high magnetic shear at the plasma edge, and show that radially extended TAE modes should be stable in conventional scenarios in ITER and that conventional reactor conditions exist where fast particle losses due to Alfvenic instabilities should be small.

Steady state operation scenarios to achieve Q = 5 have been developed for ITER with modest requirement on confinement improvement and beta ($H_{H98(y,2)} \geq 1.3$ and $\beta_N \geq 2.6$). Stabilization of Resistive Wall Mode (RWM), required in such regimes, is feasible with the present saddle coils and power supplies with double-wall structure taken into account.

An international database consisting of scalar and 2-D profile data on Internal Transport Barrier (ITB) plasmas is being developed to determine the requirements for the formation and sustainment of ITBs and to perform tests of theory-based transport models in an effort to improve the predictive capability of the models. There is only limited agreement between the model predictions and the experimental results from DIII-D, JET and JT-60U.

Recent analysis shows a possibility of high power steady state operation with a fusion power of e.g. 0.7 GW at Q ~ 8. Achievement of the required $\beta_N \sim 3.6$, above the no-wall limit (2.8) and below ideal wall limit (3.8), by RWM stabilization is a challenge and further analysis is also needed on the reduction of the divertor target heat load. With a DT isotopic mixture of ~

0.2 or ~ 0.8, there is a possibility of simulating a reactor plasma condition with a fusion power of 1 GW, e.g. high β and power and particle control within the capability of the ITER hardware with $P_{heating}$ = 250 MW and $P_{heating}/R$ = 40 MW/m, which is within a factor of two of the DEMO values. Most normalized parameters can be tested in the range of DEMO reactors at DEMO-relevant parameters. The parameter on radiative cooling, P/R, can be projected with use of sophisticated divertor codes benchmarked against high power experiments in ITER. Modest confinement improvement (e.g. $H_{H98(y,2)}$ = 1.1) would enable investigation of self-heated plasmas (Q ~ 50) at a plasma current of 15 MA.
In tokamak plasma equilibria with hollow q-profile, Energetic Particle Modes (EPM) are most likely excited within the minimum-q surface. The non-linear EPM dynamics causes the radial displacement of an unstable propagating front (avalanche), associated with rapid fast-particle radial redistribution which stop at the minimum-q surface, suggesting the idea of an energetic-particle Internal Transport Barrier (ITB), analogous to that of the thermal plasma. The shape of the q profile determines the Alfven mode behavior, suggesting that good confinement of fusion products will set a limit on the maximum radial location of the minimum-q surface and of the value of q_{min}. More work is required to develop and validate theoretical models, before adequate predictions can be made for the stability boundaries in reversed shear ITER configurations.

Since 1992, the developments of ITER diagnostic system have been tackled in a coordinated programme involving all ITER partners, and a comprehensive diagnostic system composed of 40 measuring systems has been designed to study the inteface with the machine components. Considerable progress on component development has been made and, although the step to ITER diagnostics is substantial and many details are not yet developed, it is believed that the basic measurements required for plasma control and machine protection can be made to the required specification. Most of the measurements needed for the control of possible advanced modes can also be provided. However, in a few cases, for example the current profile, the required specifications are not yet fully met.

The first burning plasma experiment, including its diagnostic systems, is not expected to come into operation during this decade. However, many parts of the systems will have to be frozen in an early phase of ITER construction. This calls for not only a very aggressive research programme on some remaining components such as the diagnostic first mirrors and radiation effects on components to be mounted in the vessel, but also, an early solution of the interface issues on diagnostic systems including their distribution around the machine.

2.2. Superconducting Magnet

All the contributed papers are devoted to ITER. After the last conference, where most of the experimental data were reported on the Central Solenoid (CS) Model Coil, efforts were oriented to the Toroidal Field (TF) Model Coil test and comprehensive understanding by the review of the results from both model coils (see Figs. 4, 5).

The TF Model Coil follows the same design principles and criteria with the ITER TF coil. Namely it consists of five racetrack-shaped double pancake wound with a Nb_3Sn circular conductor embedded in machined grooves of a radial plate. Weight of TF Model Coil is 35 tons. The coil was successfully built and installed into the TOSKA facility at FZK. In 2001 (phase 1 test), the TF Model Coil was successfully energized to 80 kA versus the rated current

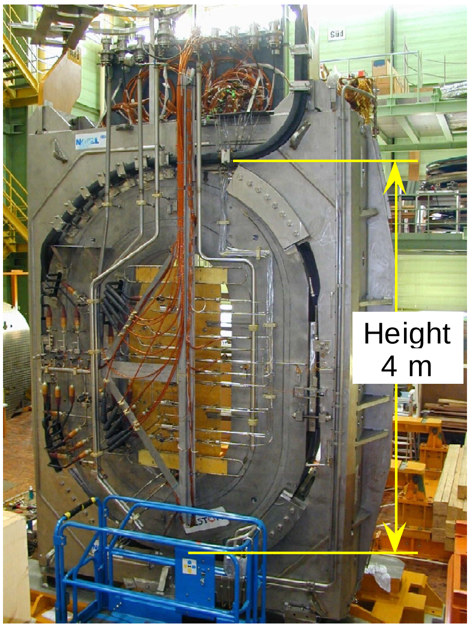


FIG.4. TF Model Coil assembled with LCT coil for the testing in TOSKA facility in FZK.

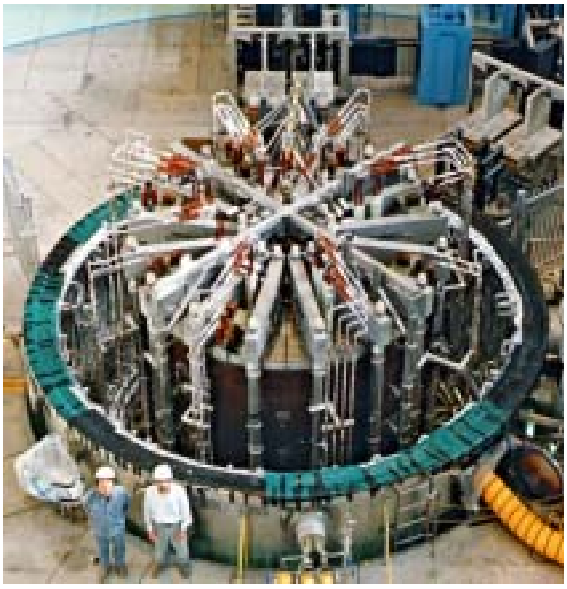


FIG.5. CS Model Coil assembly for the testing in JAERI.

of 68 kA in the ITER TF coil. The other superconducting characteristics of TF Model Coil were demonstrated to be in good agreement with the expectations. Phase 2 test (testing of TF Model Coil with the LCT coil, providing higher magnetic field and out-of-plane load) is now being performed to determine actual operation performances and mechanical characteristics.

The CS Model Coil and three insert coils with different types of conductors (two Nb_3Sn and one NbAl insert coils) had been tested in 1999 – 2001 in JAERI Naka Facility. All model coils have reached their full performance in terms of current (40 - 46 kA) and magnetic filed (13 T), and have performed satisfactory as regards the structural and electrical characteristics. This indicates that the design choices and most of the manufacturing processes proposed for the ITER magnets have been validated. However, the operational temperature margin of the Nb_3Sn superconducting conductor was found lower than expected (the difference is typically smaller than 1 K). Possible causes have been identified to be transverse strain on a strand and non-uniform current distribution among strands.

A twin-box joint concept developed at CEA was used in TF Model Coil and achieved low electric resistance and homogeneous current distribution. Prototype conductors for the ITER magnets have also been tested extensively at the SULTAN facility at CRPP aiming at in-depth characterizations that could not be performed in the model coils. Major contributions were made in transient stability study and AC loss characteristics. These results provided a valuable feedback to the ITER magnet design, with performance optimization and possible cost reduction.

Based on these findings, the conductor design of the ITER magnets is being optimized to fulfill the requirements. The technical specifications of the ITER magnets for procurement are being prepared jointly by the ITER International Team and the Participant Teams. This seems to ensure that all the experiences gained by these peoples during the model coil programmes are fully incorporated in these specifications.

2.3. Vacuum Vessel and In-vessel Components

Design and R&D activities for ITER vacuum vessel (VV), first wall (FW)/blanket and divertor between 2000 – 2002 are presented. Although the basic concept of the VV and in-vessel components of the ITER design has stayed the same, several detailed design improvements have been pursued to increase reliability, to improve maintainability and to reduce the cost such as revising the VV support system. The design of the First wall/blanket system are revisited to take into account fabrication methods and to provide cost benefits. Integration of the divertor supporting structure into the VV inner shell reduces its cross-section and promises to simplify its manufacture. Procurement specifications are now being prepared for ITER components whose delivery is on the critical path, such as the vacuum vessel.

As a continuous activity, several R&Ds were performed to improve the reliability of vacuum vessel. Among them, remotized NDT inspection of the field joint between the VV sector and the port extension was successfully performed using a robot to move a scanning device. Four different prototypical FW panels have been fabricated using DSCu and CuCrZr, and HIPing and brazing method (See FIG.6). Thermal fatigue tests have been successfully completed up to 5000 cycles at 1 MW/m^2. Fabrication of prototypical shield blocks was completed. Finally, a FW panel prototype with a central beam support was fixed onto the shield block (See FIG. 7).

A prototypical vertical target with CFC monoblock and W lamellae armor on the same PFC has been fabricated. A CFC-monoblock-armored mock-up with annular flow coolant tubes has been tested and it survived 1000 cycles - 20MWm^{-2} with no degradation of the thermal performance.

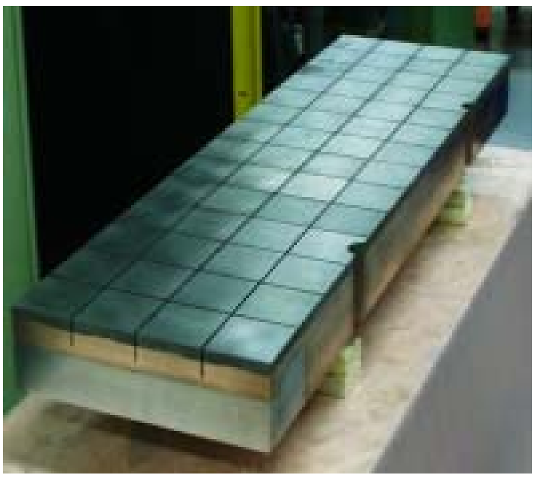


FIG.6. One of full-scale FW panel prototypes (Be brazed to DSCu layer) (EUPT)

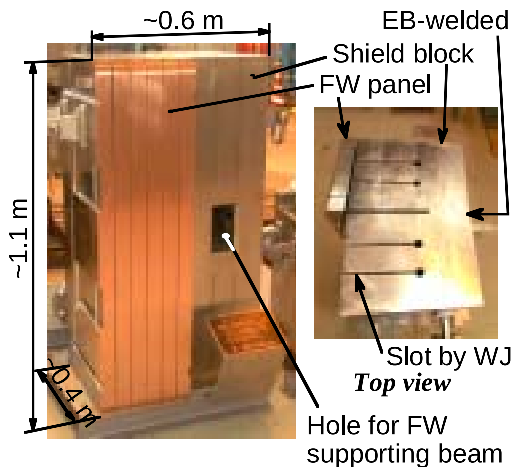


FIG.7. Assembled FW panel prototype and shield block (1/2 toroidal width) (JAPT)

2.4. NBI Technology

The negative ion technology is one of the key issues for the heating and current drive system in ITER. Progress of the developments on the negative ion accelerators, negative ion sources and negative-ion-based neutral beam injectors (N-NBI) has been reported by CEA-Cadarache, JAERI, and NIFS, respectively (See FIG. 8).

A single gap, single aperture accelerator for 1 MeV negative ion beam at CEA-Cadarche could sustain 940 kV by a main 350 mm gap, shortened from an original 627 mm gap. A D^- beam of 849 keV with a current density of 20 A/m^2 has been accelerated. The measured beam optics agreed with the prediction taking into account space charge neutralization in the accelerator.

A multi-aperture, multi-grid vacuum insulated beam source which is the reference design for the ITER NBI, has been developed to produce an H^- beam of 971 keV, 20 mA for 1 s at JAERI. Voltage holding capability was improved to sustain dc 1 MV for 8,500 s by using new stress rings in the accelerator. To improve the efficiency of the acceleration, a low-pressure negative ion source has been developed. An H^- current density of 310 A/m^2, the same level of the ITER NBI, was produced at very low pressure of 0.1 Pa, which is 1/3 of the ITER design pressure.

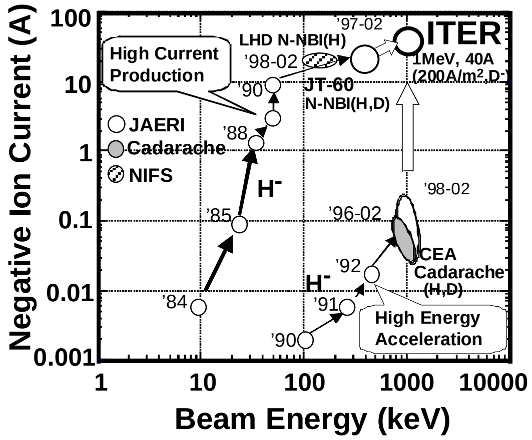


FIG.8. Progress of the negative ion beam technologies.

As for the integrated systems, N-NBIs for tokamak and helical device have been presented. In the N-NBI for JT-60U, beamlet deflection has been improved by correction of electric field distortion in the first acceleration gap. A maximum neutral beam power of 6.2 MW at 381 keV has been injected. A rated pulse length of 10 s at 2.6 MW was injected by the reduction of the heat load on the NBI port with a well converged beam. In the N-NBI for helical device, LHD at NIFS, a neutral beam power of 9 MW for 2 s at the beam energy of 160 keV was injected to LHD from three injector modules. Long pulse operation for 80 s with 0.5 MW by two ion sources has been performed. These progresses of beam technologies encourage the construction of the ITER NBI system.

2.5. RF Technology

Present status of 100GHz band gyrotron development was reported by Japan (JAERI), EU (FZK, CRPP, etc.), US (GA, CPI etc) and RF (IAP). Significant progress of gyrotron performance has been demonstrated during last two years as shown in Fig. 9. The 110GHz gyrotron were integrated to the ECH system and applied to fusion devices (JT-60U, DIII-D). Advanced gyrotrons (2MW, frequency tunable gyrotrons) were reported. These give a clear prospect for development of 1MW/CW gyrotron for ITER and advanced ECH system. Two activities on ICRF antenna were also presented.

Present high power gyrotrons were featured by an oscillation with very high order cavity mode, a depressed collector, an internal mode converter and a diamond window. On ITER 170GHz gyrotron, 0.9MW/9s, 0.3MW/60s (JAERI) 0.55MW/40s(IAP) were attained. The pulse duration was limited by a heat deposition of a stray radiation to the inner element. These performances will be improved by reducing a stray radiation in the gyrotron.

Long pulse operation was demonstrated on 140GHz gyrotron for W-7X: 0.9MW/3min, 0.5MW/~1000s. A higher power operation was attained on 110GHz gyrotron for JT-60U: 1.3MW/1.5s.

Aiming at higher power generation, a coaxial gyrotron was fabricated and tested. A power output of 2.2MW(1ms) was obtained at 165GHz as a proof of principle (FZK). Another advanced gyrotron is frequency step tunable gyrotron. A power output of 1MW/0.5sec was demonstrated at both 140GHz and 104GHz by changing applied magnetic field (IAP).

In the integrated EC system on JT-60U and DIII-D, 3MW level plasma power was injected using steering launcher at 110GHz. The electron heating and NTM suppression were proved using a feedback control of gyrotron power or mirror steering. High efficiency RF power transmission was demonstrated at GA and JAERI: 80% with 100m transmission line with 14 miter bends, and 95% with 12m transmission with 3 miter bends, respectively. These give a clear prospect for realization of EC system on ITER.

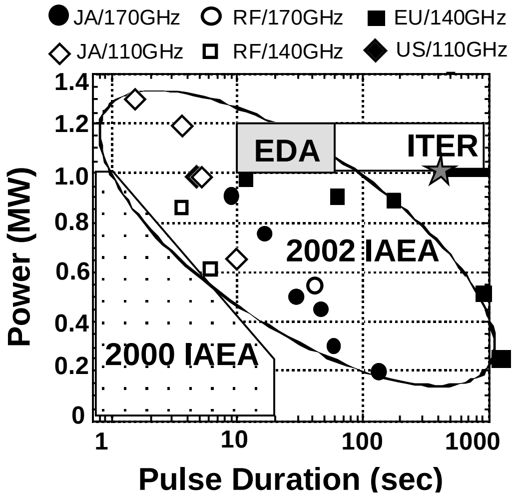


FIG.9. Status of Gyrotron performance. (Frequency>100GHz, Pulse duration>1s.)

A compact, 4-strap ICRF antenna has as been developed in Alcator C-Mod, and the maximum injected power and power density reached 3MW and 11 MW/m^2 respectively. A traveling wave ICRF antenna is being developed for LHD, motivated by the need to provide a capability for rotational transform profile control by RF current drive.

3. Toroidal Devices under Construction or under Design Study

Construction of the full superconducting tokamaks in Asia has been forwarded in the direction of steady state research; KSTAR in Korea, HT-7U in China and SST-1 in India. In Japan, a full superconducting tokamak of JT-60SC has been newly designed as the modification of JT-60U. For burning plasma physics, the design studies of FIRE and Ignitor using copper coils have progressed to enhance the capability for advanced tokamak regimes. To increase the plasma beta with spherical torus regimes, the design of NSST has been in progress for a performance extension phase succeeding NSTX. W7-X is the follow-up of W7-AS and is presently under construction. Compact stellarartor facilities of NCSX and QPS are currently in the conceptual design phase. All these directions and results are very important to pave the way for attractive fusion power plants.

3.1. Superconducting Tokamaks

The KSTAR project started from 1996 (I_p = 2 MA, B_t = 3.5 T, R_p = 1.8 m, a = 0.5 m, double/single null, 20-300 s, deuterium) is presently in the procurement phase. The engineering design phase including prototype fabrication has been finished in March 2002. The experimental building has been completed. The machine assembly work will begin in the model of 2003. The cooling test of the real-sized prototype TF coil will be carried out in 2002. The final engineering design of plasma facing components will be finished in 2002. The target date of the first plasma is scheduled to the end of 2005.

HT-7U (B_t = 3.5, I_p = 1.0 MA, R_p = 1.75 m, a = 0.4 m, double/single null, P_{IC} = 3.0 MW, P_{LH} = 3.5 MW, P_{EC} = 0.5 MW, 1000 s, deuterium) aims at wide integration of physics and technology for steady state advanced tokamak as well as investigation of power and particle handling. The physics and engineering design has been completed and significant progresses on R&D and fabrication have been achieved. The TF magnet system consists of 16 D-shape coils using NbTi cable-in-conduit conductor (CICC). The machine assembly will begin in 2003 and plan to get the first plasma in 2004.

SST-1 is a steady state superconducting tokamak being fabricated with the objectives of studying the physics of the plasma processes and the technologies in tokamak. Most of the components of SST-1 (I_p = 0.22 MA. B_t = 3 T, R_p = 1.1 m, a = 0.2 m, double/single null, hydrogen, P_{IC} = 1.5 MW, P_{LH} = 1 MW, P_{EC} = 0.2 MW, 1000 s) have been successfully fabricated and tested including TF and PF superconducting magnets with NbTi based cable-in-conduit conductor.

JT-60SC is a fully superconducting tokamak designed for the modification of JT-60 to enhance economical and environmental attractiveness in tokamak fusion reactors (B_t = 3.8 T, R_p = 2.8 m, a = 0.85 m, κ_{95} = 1.8, δ_{95} = 0.35, deuterium, I_p = 4 MA with a flat top of 100 s, P_{NBI+EC} = 44 MW/10 s to 15 MW/100 s). The modification program aims at realizing high-beta steady-state operation with β_N = 3.5-5.5 in the use of low radio-activation ferritic steel in low ν^* and ρ^* regimes relevant to the reactor plasmas. The design of JT-60SC is based on advanced R&D results including Nb_3Al and Ni-coated NbTi conductor. To demonstrate the applicability of Nb_3Al conductor to the TF coils, R&D using a full-side Nb_3Al conductor with react-and-wind technique has been carried out. For the PF coil system, a full-side NbTi conductor with low AC loss using Ni-coated strands has been successfully developed.

3.2. Copper Coil Tokamaks

The FIRE design study has been undertaken to define the lowest cost facility to address the key burning plasma and advanced tokamak physics issues identified in the ARIES studies. It was focused on the physics and engineering assessment of a compact, high-field, cryogenic-copper-coil tokamak (R_p = 2.14 m, a = 0.595 m, B_t = 6 to 10 T, I_p = 4.5 to 7.7 MA with a flat top time of 40 to 20 s for 150 MW of fusion power). The ELMy H-mode regime has been shown to produce Q~10 sustained under quasi-stationary conditions for 2 times current diffusion times (τ_{RC}). As a long term goal, the steady state high-beta advanced tokamak regimes will be explored with a high bootstrap fraction, a high beta and a moderate fusion gain for ~3 τ_{RC}.

Ignitor is a compact (R_p/a = 1.32 m/0.47 m) high field copper machine with higher field (13 T) and current (11 MA) capability in accordance with ignition mission. The simulation of the path to ignition conditions for L-mode and H-mode regimes using the JETTO code has shown that a modest injection of ICRH accelerate ignition so that the relevant burning plasmas can be investigated over times that exceed the current redistribution time. The BALDUR code analysis has shown that the particle density profile control is important to the approach to ignition when reversed shear conditions with peaked density profiles are produced through appropriate current ramping.

Next-Step Spherical Torus (NSST) is a performance extension stage ST (Spherical Torus) with a plasma current of 5-10 MA, $R_p = 1.5$ m, $B_t \leq 2.7$ T), which is designed to utilize the TFTR site. The design study has been carried out 1) to provide a sufficient physics basis for the design of CTF (Compact Test Facility), 2) to explore advanced operating scenarios with high bootstrap current fraction/high performance regimes, which can be utilized by CTF, DEMO and Power Plants, and 3) to contribute to the general plasma/fusion science of high beta plasmas.

3.3. Stellarators

WENDELSTEIN 7-X (W7-X) is presently being built at Greifswald. W7-X aims to confirm the favorable plasma properties and the high density and beta limits of the helical advanced stellarator and to demonstrate steady state operation. The magnetic configuration is defined by the current distribution in 50 non-planar superconducting coils with a current of 18.2 kA. The standard magnetic configuration is characterized by $R_p = 5.5$ m, a = 0.53 m, $B_t = 2.5$ T and an ι of 5/5 at plasma boundary. In order to keep the symmetry of the magnet configuration, the shape of all ten coils of a given type need to be fabricated within a dimensional accuracy of typically 2 mm. Design of the power supplies allows to adjust the magnetic induction with an accuracy of 30 mT and a field ripple of the order of 2 mT. Although series production of the superconductor has been released in May 2001, unexpected delay of the conductor has a serious impact on production of the non-planar and planar coils and hence on the overall project schedule.

Two compact stellarartor facilities in the U.S. are currently in the conceptual design phase: the National Compact Stellarator Experiment (NCSX) and the Quasi-Poloidal Stellarator (QPS). The primary feature of NCSX as a proof-of-principal scale device with $R_p = 1.4$ m and $B_t = 2$ T is a set of modular coils, which represent a major engineering challenge due to the complex shape, precise geometric accuracy and high current density of the windings. The QPS device is a smaller, concept exploration experiment with $R_p = 0.9$ m, $B_t = 1$ T, a small aspect ratio of 2.7. Instead of an internal vacuum vessel, the QPS modular coils will operate in a bell jar.

4. Reactor Technology

4.1. Materials Development and Irradiation Facility

Research activities on the blanket structural materials and high heat flux materials are reported. The lack of fusion neutron irradiation environment is being overcome through two ways. One is the construction of International Fusion Materials Irradiation Facility (IFMIF) to produce fusion-relevant neutron environment and the other is the analytical and experimental approach to the 14 MeV-neutron damage.

Progresses on IFMIF program are reported. During the “Key Element Technology Phase”(KEP, 2000-2002) of IFMIF, many tasks are completed mainly to reduce the uncertainties of key technologies. Improvements in ion source enabled 1000 h continuous operation of an intense D+ source. For the linear accelerator, a diacrode to be used in RFQ demonstrated the 1000 h full power operation, showing the capability to operate stably under IFMIF-relevant conditions. New libraries of nuclear data and advanced neutron transport calculations are now available to optimize the configuration of irradiation field to adjust the neutron spectrum to simulate the DEMO blanket and also to increase irradiation volume. Thus, preparations to enter the next stage "Engineering Validation and Engineering Design Activities" (EVEDA) are now made.

Reduced activation ferritic/martensitic steels (RAFS) and silicon carbide (SiC) fiber-reinforced SiC matrix composites (SiC/SiC composites) are the major research target for the blanket structural materials. Research activities reported on vanadium alloys were limited.
As shown in FIG. 10, RAFSs are considered as the baseline materials to be used in DEMO with a few R&D risks, while SiC/SiC composites are considered as the advanced materials which would enable ideal operation of fusion power plant.

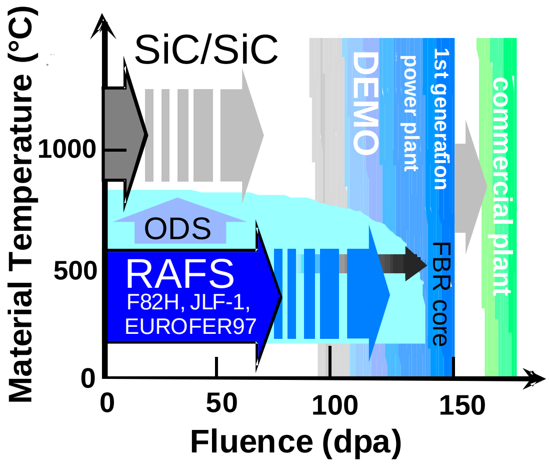


FIG.10. Conceptual operation domain of candidate structural materials for future fusion reactors.

RAFSs will be used for the blanket structural material. Their advantages are, the engineering maturity, higher resistance to swelling, high thermal conductivity, lower induced activity, etc. The Japanese candidate alloys are F82H (8Cr-2W) and JLF-(9Cr-2W), while the EU candidate is EUROFER(9Cr-1W). The material data on Japanese candidates before and after irradiation are being compiled into a database. The engineering maturity of RAFSs and knowledge about materials properties are sufficient to develop the R&D schedule of RAFSs to be in time for the DEMO operation. The R&D programs include material irradiation experiments in fission reactors and in IFMIF, and functional tests on blanket modules shall be performed in ITER in parallel.

To expand design window for the DEMO blanket, following issues shall be attacked:
- Softening during the service can limit their usage at high temperatures (>550-600°C).
- Embrittlement during the service at relatively lower temperatures (<350°C) would cause problems during maintenance.
- At the intermediate temperature range, accelerated swelling by He may exceed 1% even in ferritic steels.

Oxide Dispersion Strengthening (ODS) steels are promising to expand the limitation to a high temperature. Refinement of microstructure and optimization of heat treatment of steels are also promising to expand the limitation for lower temperature range.

Positive and negative effects of ferromagnetism on tokamak operation are reported. Reduction of toroidal field ripple has been demonstrated by JFT-2M operation with F82H plates in it, while calculations show plasma beta would be reduced slightly by the ferromagnetism.

A Blanket system using SiC/SiC composites is expected to provide very high thermal efficiency and extremely low induced radioactivity. However, issues still remain in their baseline properties before irradiation, such as the inexperience as industrial materials, lack of proven joining and hermetic sealing techniques. Their rather small deformability may require an innovative design methodology for the blanket. They are, however, making rapid progresses in these aspects through their application to recent advanced energy systems.

An advanced fabrication method (NITE process) of SiC/SiC composite is proposed, which improves the baseline properties. The method employs fine SiC powder with a particle size of 20 nm for sound matrix formation. Superior mechanical and thermal properties along with a good gas-leak tightness have been obtained.

Material R&D activities have several aspects. One is, of course, the development of individual material, such as RAFSs and SiC/SiC composite. These structural materials should satisfy several requirements simultaneously, namely not only strength and toughness but also fatigue property and hydrogen permeation etc. Studies on individual properties form another aspect. Since the current irradiation experiments are simulations for fusion environment, prediction of material behavior by analytical approach including modeling calculations is also important.

Studies were made on the plasma-wall interaction in simulated experiments using tokamak He discharges or by mixed beam irradiation. The tritium inventory of divertor due to co-deposition with sputtered carbon need more study and database.

4.2. Blanket and Neutronics

In the area of blanket and neutronics, well-organized development plans are being pursued for development of DEMO blankets. An advanced DEMO blanket has been proposed, applying supercritical water as a coolant to realize higher electrical power efficiency. Furthermore, achievements of design and technology developments have been presented. Supporting R&Ds have significant progress on fabrication technology development of advanced breeder and multiplier materials, and neutronics experiments for improvement of blanket neutronics evaluations on tritium production and activation. As an analytical work of neutronics, detailed 3-D analysis has been performed for EU water-cooled Lithium-Lead (LiPb) Test Blanket Module for ITER. As for the improvement of nuclear data, the activities under the European Fusion File project were overviewed.

Design of a solid breeder blanket cooled by supercritical water was proposed as an advanced concept of DEMO blanket, with higher thermal efficiency of more than 40 %, as shown in FIG. 11. Its design covered and integrated major critical issues. Significant progress of supporting technology R&D's have been made in the area of the first wall, box and pebble bed structure fabrication. Development of breeder and multiplier materials has also been made. The fabrication method of advanced TiO_2-doped Li_2TiO_3 pebbles by in-direct wet process was successfully developed. Beryllides have been studied as alternative of beryllium metal for DEMO blanket, because of their chemical stability in high temperature. Basic characterization

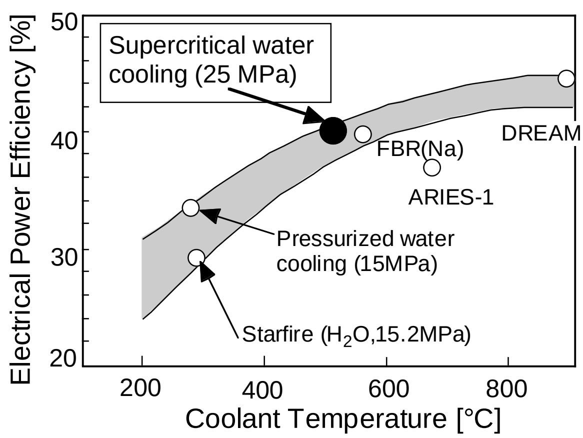


FIG.11. Electrical power efficiency of advanced blanket cooled by supercritical water

showed better characteristics than Be on compatibility with structural and breeding material, swelling and tritium inventory.

Neutronics experiments for a water-cooled solid breeder blanket have been performed. The tritium production rates (TPR) obtained by the experiments agreed with derived from the calculation results within 10 %. In activation measurement experiments, it has been clarified that sequential reaction rates are enhanced by taking into account the recoiled protons from the water. Three-dimensional neutron calculations have been made for the EU water cooled Lithium-Lead test blanket module for ITE, on the issues of tritium production, energy deposition and radiation damage. The results will be utilized for the tritium permeation analysis
and thermo-mechanical analysis of the module structure. The European Fusion File (EFF) Project has been continued to improve the quality of nuclear data for the fusion power plant design and operation studies. Important progress has been made on the primary tasks of nuclear data-file production and verification, development of calculation methods and nuclear data-file validation via integral measurements.

Thus, the development of blanket elements for DEMO has shown steady progress as a first step. For the second step, the development of integrated blanket modules of clearly defined targets is essential, in conjunction with the neutronics research for improvement of blanket designs.

4.3. Reactor Design

Reactor design activities cover varieties of types, spherical torus, tokamak, stellarator, and mirror, and purposes from neutron application to power reactor. In this conference, all the papers are in conceptual design level. Tokamak and STs designs are aiming at advanced concepts to drastically improve its economy. Stellarator designs are improving for more realistic studies.

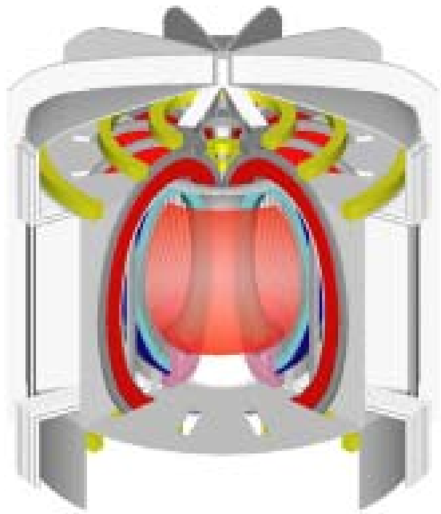
Designs on spherical tori covered various applications. Since spherical torus reactors can be made small in physical dimension but have drawback of economy, it is reasonable to pursue neutron application such as transmutation. These studies such as neutronics design follows previous regular tokamak design in the past and has not shown a marked progress. Application of fusion for transmutation must satisfy complicated criteria. Design of practical blanket that satisfies neutronics including tritium economy and heat removal will be one of the most important issue in these studies. Strong future design works will be needed to make these designs more realistic.

Possible ST power reactor design based on the current physical understanding was reported. It envisions helium coolant for blanket and pebble cascades for divertor. However these concepts will require further investigation. It is generally understood that the ST power reactors will have to handle high heat flux in both divertor and blanket, because of its smaller size than regular tokamak. Therefore, design constraints on ST reactor should be focused on these power handling components and devices.

Superconducting tight aspect ratio tokamak suggests a new possibly economical power reactor option (FIG.12). It takes advantage of thinner coil structure enabled by removal of central solenoid from conventional tokamak. This concept resolves inherent problems of power economy of normal conducting spherical tori and massive superconducting coils in regular tokamak. Further work will be expected on detailed design study as well as related experiments for this concept. Another possible concept to reduce the electromagnetic stress was also reported.

Stellerator designs increased reality based on recent progress of large device projects. Helias reactor design suggesting ignition experiments can be envisaged beyond Wendelstein 7X. Cost of helical power generation was assessed in some restricted parameter regions and lower aspect ratio and smaller dimension were suggested to compete tokamaks. Such an attempt to compare stellarator and tokamak on a common technical and economical basis is indeed valuable. However it is expected for stellarator to accumulate more physics and engineering database for such reliable comparisons.

A mirror machine for volumetric neutron source was also designed. Further improvement in concept will be needed for practical application in the future.



R=3.75m, a=1.9m, β_N=5.6, Bmax=19.6T, Ip=18.3MA, Pf=1.8GW

FIG.12. Appearance of the tight aspect ratio tokamak

4.4. Laser-driven Inertial Fusion Technology

Many progresses have been made in researches and developments for laser-driven inertial fusion energy including chamber concepts, laser drivers, target fabrication and its injection and power generation system study.

Feasibility studies on different types of first wall design have been pursued including dry-wall and wet-wall concepts of chamber that protect structures from high energy neutron and ions irradiations. Analyses and experiments on gas-protected concept for dry-wall design indicated feasibility of a design widow avoiding first wall vaporization and permitting target injection without damage to the fuel. For wet-wall design, analyses on ablation of the wall during the fusion pulse have shown optimum conditions about chamber radius and fusion power related to the power plant.

A prototype of an electron-beam pumped krypton fluoride (KrF) laser/amplifier has shown promising progresses of a focused intensity of $3x10^{18}$ W/cm^2 versus a target value of 10^{19} W/cm^2 and demonstrating total shots of 10^4 at a repetition rate of 1 Hz. For reflective optics for laser drivers, highly polished, grazing incidence aluminum mirrors have been developed. Their allowable laser fluence is well above the 5-10 J/cm^2 needed for Inertial Fusion Energy designs.

Many target fabrication techniques for a spherical capsule containing DT fuel have been developed including coating a foam-insulator on the capsule by using an emulsion process and an ablator material by using a fluidized bed. An experimental injection and tracking system is being constructed to develop technologies aiming at high accuracy. Models of thermo-mechanical effects on the target during injection have been developed.
System design study on KOYO-fast in Osaka Univ. has been pursued to examine the power plants. Design of a compact reactor of 240 MWe with the fusion pulse energy of 200MJ and

the pulse repetition rate of 3 Hz is proposed for the modular plant concept. Engineering design on the thermo-hydraulic issues is encouraged for future work.

5. Safety and Environment

Socio-economic study of fusion is attracting more interests than ever, because of its importance in the context of development strategy from ITER toward utilization of fusion as a viable energy source.

Global energy model that describes future energy mixture under environmental constraints revealed that early fusion introduction reduces world energy cost by hundreds Billion $ (FIG.13). Construction speed and initial tritium supply have strong impact on the share of fusion in the order of 20%.

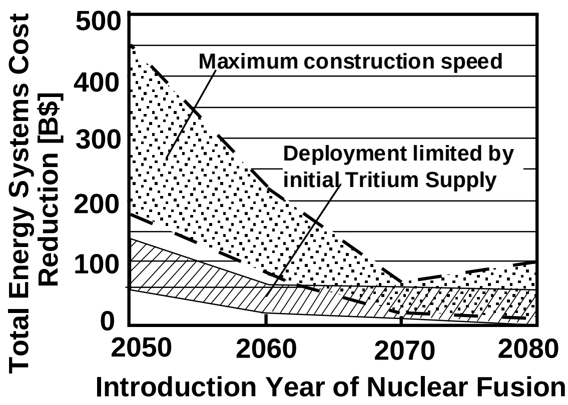


FIG.13. Future contribution of fusion energy

European study on various features of fusion energy from socio-economic aspects analyzes growing importance of social acceptance by local communities. It is important for fusion community to obtain the social understandings and such socio-economic studies should be pursued in more active way.

6. Conclusion

ITER Engineering Design Activity was completed in July 2001, and is now under the transition phase toward start of construction. During this period, integrated ITER design has been optimized in several points. ITER performance prediction and the results of technology R&D provide good confidence to start construction. It seems to be the fusion community's wish to conclude negotiations as soon as possible.

In both near term programs and future reactor studies, most of advanced tokamaks, STs or Helical systems stress economical approach, in particular, the importance of high beta, and steady state operation. The component technologies development for inertial fusion systems have also shown a steady progress.

Reactor technology in the field of blanket and materials for demonstration power plant showed a sound progress in both R&D and design basis. These progresses are discussed in relation to ITER and IFMIF programs in a coordinated manner. However, much more contributed papers to the Conference should be encouraged taking into account the importance of this field.

Fusion energy must show its practical applicability as early as possible to contribute to the global energy and environmental problems. Discussion to accelerate development of fusion was one of the general interests in the conference. The need for materials development program and the need for advanced plasma researches to support ITER operation and beyond seems to be an urgent issue to be discussed in the world fusion community.