MEETING REPORT

SUMMARY OF THE ALL-UNION SEMINAR ON TOROIDAL SYSTEMS IN CONTROLLED FUSION, DUBNA, USSR, FEBRUARY 20-24, 1989

The All-Union Seminar on Toroidal Systems in Controlled Fusion was held at Dubna, USSR (130 km from Moscow), February 20-24, 1989. More than 100 specialists from the USSR and 16 invited specialists from the United States, United Kingdom, Federal Republic of Germany, German Democratic Republic, People's Republic of China (PRC), Poland, and Czechoslovakia participated. The managers of the International Thermonuclear Experimental Reactor (ITER) project were also present: Yu. A. Sokolov (USSR), J. Gilleland (United States), K. Tomabechi (Japan), and R. Toschi (Europe). The Joint European Torus (JET) Undertaking was represented by P. Rebut. R. Janev, head of the Atomic and Molecular Data Section of the International Atomic Energy Agency (IAEA) also attended.

The choice of Dubna as the place for the seminar was not accidental: An international symposium on toroidal systems was held there in 1969, after which active research on tokamaks was begun.

The seminar was dedicated to the 80th anniversary of the birth of L. A. Artsimovich, who was head of fusion studies in the USSR until his death in 1973. This year is also the 70th anniversary of the birth of M. S. Rabinovich, who was head of the stellarator program at the Lebedev Institute for many years until his death in 1982 (this study is being continued at the Institute of General Physics). The Artsimovich memorial lecture was presented by V. D. Shafranov (USSR) and the Rabinovich memorial lecture by I. S. Shpigel (USSR).

Thirty-five invited papers were presented on the following topics:

1. experimental results from modern tokamaks and stellarators
2. plasma heating
3. current drive
4. equilibrium and transport processes in toroidal systems (theory)
5. new facilities.

The paper distribution was as follows: USSR – 21, United States – 2, United Kingdom – 2, ITER – 4, PRC – 1, Poland – 1, and IAEA – 1. Different aspects of ITER (physics, nuclear technology, engineering) were discussed at a special session.

The experimental results obtained from T-10 (K. A. Razumova), Tokamak Fusion Test Reactor (TFTR) (K. McGuire), DIII-D (R. Freeman), JET (P. Rebut), Axially Symmetric Divertor Experiment (ASDEX) (F. Soeldner), and Divertor and Injection Tokamak Experiment (DITE) (D. Robinson) were widely discussed. Different aspects of energy and particle confinement were also studied. O. S. Pavlichenko reviewed results from stellarators. The theoretical aspects of plasma magnetic confinement and limit parameters were discussed by B. B. Kadomtsev, O. P. Pogutse, L. M. Degtyarev, Yu. N. Dnestrovskii, and D. P. Kostomarov. Plasma heating problems were considered by A. V. Gaponov-Grekhov et al., V. V. Alikaev, V. E. Golant and V. N. Budnikov, and V. M. Kulygin et al. G. V. Pereverzev reviewed current theoretical studies on noninductive current drive methods. The results obtained from toroidal systems in the United Kingdom and the program for the future were presented by D. Robinson. P. Rebut presented "JET: Status and Plans" and "Next Step View – Impact of JET Results."

Results of plasma physics research at the Plasma Physics and Laser Microfusion Institute were presented by E. Wolowski (Poland). Huo Yu Ping (PRC) presented "Global Properties of Tokamak Plasma Studies by Magnetic Helical Perturbations." The relationship between atomic processes and controlled fusion was discussed by V. V. Afrosimov and A. N. Zinoviev and by R. Janev. The paper by V. V. Orlov and A. V. Kashirskii was devoted to the important problem of fusion power production safety. N. A. Monoszon presented the main engineering solutions used in the design process and construction of large Soviet tokamaks.

In the following, papers devoted to the Soviet T-15 and TSP tokamaks, which recently began operation, and those on ITER are considered in more detail.

The startup of the T-15 tokamak and its physics program were discussed by V. S. Strelkov. The first experiments were performed in December 1988. A plasma current of ~120 kA and a pulse duration of 50 ms were obtained. The superconducting magnetic system was cooled almost to operating temperature ($T_{inlet} = 11.5$ K, $T_{outlet} = 13.5$ K), and the superconducting state of the Nb_3Sn alloy toroidal field (TF) coil

TABLE I

Main Parameters of the T-15 Tokamak

Parameter	Stage 1	Stage 2
Major radius (m)	2.43	2.43
Minor radius (m)	0.5 to 0.7	0.5 to 0.7
Magnetic field on chamber axis (T)	3.5	Up to 5.0
Plasma current (MA)	1.4	Up to 2.3
Current pulse duration (s)	5	5
Total flux (V·s)	15	17
ECRH power (MW)	6	6
Neutral injection power (MW)	9	9
Auxiliary heating pulse (s)	1.5	1.5

at a 100-A current was observed. The main parameters (see Table I), the choice of auxiliary heating methods, and the structural design were determined by the main tasks of T-15, experience obtained from the construction of T-10 and T-7, and experimental results on ohmic heating and auxiliary heating by neutral beam injection (NBI), and by radio frequency (rf). The decision to construct the facility in the old building imposes certain limitations. Particularly, tritium experiments are absolutely forbidden.

The maximum calculated toroidal field is 5 T on the axis and 9 T on the winding. The operating temperature of the coil is 4.5 to 6 K. At stage 1, the plasma current and the gyrotron frequency (83 GHz) are determined by the magnetic field (3.5 T). The possibility of transition to stage 2 ($B = 5$ T) will be determined after additional engineering studies.

The vacuum, cryogenic, power supply, and TF protection systems were operating at the rated parameters during start-up of the facility. The poloidal field (PF) system, the discharge initiation system, and the plasma diagnostics were operating at lower parameters. The data acquisition system operated only for the above-mentioned systems. During 1989–1990, there are plans to achieve the stage 1 design parameters, to run ohmic heating experiments, to investigate the stressed state of the TF coil, to perform thermophysical measurements on the cryogenic system, to complete the construction, and to achieve the design parameters of the auxiliary heating systems (three injectors and gyrotron set). The waveguide system will be partly replaced by quasi-optical power transmission systems. The model experiments show that the losses for the T-15 geometry may be as high as 15 to 35%. This allows 7.5 MW to be launched in the tokamak chamber.

The general plan for the T-15 scientific program includes fusion plasma physics, recommendations for the ITER/experimental fusion reactor (OTR) program, and engineering studies. The main research areas for the T-15 program are as follows:

1. transport processes in electron and ion components for ohmic heating, electron cyclotron resonance heating (ECRH), and NBI heating
2. investigation of the possibility of preventing or delaying disruptions by local deposition of auxiliary heating power
3. current drive by electron cyclotron resonance waves and by NBI
4. boundary plasma physics, plasma/wall interactions, recommendations concerning wall materials, and methods to reduce Z_{eff}.

Eight scientific programs to meet these goals have been developed. It is necessary to plan and start modernization of the T-15 in the next 2 yr.

The main features of the TSP, a tokamak with a high magnetic field, were discussed by E. A. Azizov, O. I. Buzhinskii, G. G. Gladush, S. V. Mirnov, and V. A. Chuyanov. The main parameters of the TSP are given in Table II, and the expected plasma parameters in Table III.

The TSP TF coil consists of 32 single-turn sections connected in series. Each section includes a heavy zirconium-bronze turn and a steel band. The peak current in the coil is 830 kA, the maximum magnetic field on the conductor surface is 21.2 T, and the maximum temperature on the coil surface is 230°C. The toroidal ripple is 1% on the plasma surface before compression and 0.001% at the axis (in the noncompressed state). The power supply to the TF coil is provided by the toroidal inductive storage, which can store 900 MJ of energy. This inductive storage ensures transmission to a net load of 150 MJ at a maximum power of 1.3×10^{10} W, which in turn ensures compression over the minor radius by a 2.5 times increase in the toroidal field in 10 ms.

To provide fast compression (in 10 ms), the poloidal system (transformer and compression coil) is switched on to separate inductive storages, which ensures transmission of 13 MJ with a power of $\sim 2 \times 10^9$ W. The total stored energy is 90 MJ.

The TSP was assembled in December 1987. Complex adjustment and tests of the facility were performed during 1988. In 1988, two TKD-200 generators reached their rated speeds (3000 turn/min) and were synchronized. The stray magnetic fields in the plasma column area were measured using a special system with hundreds of electron beams. This system allows detection of vertical fields of $\sim 10^{-3}\ B_T$ (B_T is a toroidal field) due to imprecise compensation of stray fields induced by primary and secondary windings of the energy storage.

These investigations show that the assembly technology used allows the available parts of the facility to be mounted at the planned locations for less than 1 week. If we consider that the tokamak cost is <10% of the cost of its servicing set and power supply system, then the TSP can be considered as an operational site for testing and optimizing different tokamaks, which can then be designed, fabricated, and prepared for testing without interfering with ongoing experiments.

In December 1988, complex tests of the total magnetic system were carried out. The TF coil was fed from the TKD-200 generator via a thyristor converter and inductive storage. Routine experiments with plasmas have begun.

The ITER problems were discussed at a special session. K. Tomabechi (chair of the ITER management committee) presented "ITER: Status," which included a description of the ITER objectives, organization structure, and research and development (R&D) necessary to realize the ITER goals. The overall objective of ITER is to demonstrate the scientific and technological feasibility of fusion power. ITER operation will be carried out in two phases: a physics phase devoted mainly to the achievement of the plasma physics objectives and a technology phase devoted to the engineering objectives and the testing program. During the physics phase ITER will first be operated in the pulsed mode, and then the burn pulse will be extended toward a steady state, aiming at as high a Q value as possible. During the technology phase, ITER should be

TABLE II

Main Parameters of the TSP

Parameter	Before Compression	After Compression over the Minor Radius	After Compression over the Major Radius
Plasma volume (m^3)	2.14	0.856	0.13
Plasma major radius (m)	1.06	1.06	0.415
Plasma minor radius (m)	0.32	0.2	0.125
Toroidal field at the axis (T)	2	5	12.8
Plasma current at $q(a) = 2$ (MA)	0.48	0.48	1.23
TF flattop (s)		0.01	0.01
Auxiliary heating power NBI + ion cyclotron resonance heating (MW)	1 + 1		

TABLE III

Expected TSP Plasma Parameters*

Parameter	Before Compression and Auxiliary Heating	Before Compression, After Auxiliary Heating	After Compression over the Minor Radius	After Compression over the Major Radius
$\bar{n}_i$ (m^{-3})	5×10^{19}	5×10^{19}	1.3×10^{20}	8×10^{20}
$\bar{T}_i$ (keV)	0.5	1.2	2.0	7
$\bar{T}_e$ (keV)	0.6	1.1	1.9	6
P_{fus} (MW)	0	0	0	3.5
Calculated energy confinement time (ms)	30	55	125	60

*Auxiliary heating power is 2 MW, initial current is 480 kA, and injected deuteron energy is 40 keV.

operated in a steady-state mode even if it will only allow a Q value of ~5.

To carry out nuclear and high-heat-flux component testing under conditions relevant to a fusion power reactor, it is necessary to have an average neutron wall load of ~1 MW/m^2. ITER should provide an opportunity to operate with a neutron fluence of ~1 $MW \cdot yr/m^2$. However, the design should allow a higher neutron fluence (in the range of 1 to 3 $MW \cdot yr/m^2$).

The areas for physics R&D include the following:

1. power and helium exhaust conditions
2. helium radial distribution in a high-temperature tokamak discharge
3. characterization of low- and high-Z materials for plasma-facing components
4. sweeping of the divertor target load
5. characterization of disruptions
6. rf plasma formation and preheating
7. rf current initiation
8. alpha-particle losses induced by the TF ripple
9. steady-state operation in enhanced confinement regimes (H-mode and "enhanced" L-mode)
10. control of magnetohydrodynamic activity
11. density limit
12. viability of a radiative edge
13. comparison of theoretical transport model with experimental data
14. plasma performance at high elongation
15. electron and ion cyclotron current drives
16. impact of Alfvén wave instability on neutral beam current drive
17. proof of principle of fueling by injection of field-reversed compact torus
18. compatibility of plasma diagnostics with the ITER conditions.

The following are ITER technology R&D requirements:

1. blanket (ceramic breeder, LiPb breeder, H_2O/Li solution breeder, beryllium, structure materials)
2. plasma-facing components
3. magnet
4. fuel cycle
5. heating/current drive
6. maintenance.

ITER physics tasks were considered by Yu. A. Sokolov, including ignition conditions, density limit, beta limit, elongation, and safety factor limit. It is now recognized that the weakest feature of the design is the impurity control and the energy and particle exhaust. According to current model calculations, the total erosion rate of the divertor plates is absolutely intolerable.

J. Gilleland presented "ITER: Scoping Studies and Operational Flexibility." Since typical modern scalings for the energy confinement time depend on the plasma current I and the aspect ratio A, a goal of the study was to find the scope of the device parameter choice in the I-A space, with fixed assumptions about peak fields, safety factor, neutron wall load, current drive power, and reactor cost. It has been shown that flexible scenarios must be developed for ITER.

Nuclear engineering aspects of ITER were discussed by G. E. Shatalov and V. I. Khripunov. The main tasks are to develop a program of nuclear engineering testing; to define more exactly the goals, modes of operation, and conditions of operation for all irradiated components inside the vacuum vessel; to determine the R&D needed to deal with the large number of disruptions expected on the physics phase carbon tiles that will be used as plasma-facing armor on major areas of the first wall. For the blanket and the first wall, the main structure material is solution-annealed Type 316 austenitic steel cooled by water at ≤ 1 MPa and $\leq 100°C$ under normal operation. The basic divertor plate concept consists of a 5- to 10-mm-thick carbon armor brazed onto a water-cooled heat sink of copper or molybdenum alloy.

Three blanket concepts have been selected for more detailed studies during the next year:

1. aqueous lithium salt
2. solid ceramic breeder
3. lithium-lead eutectics.

The different criteria are taken into account for radiation shield designing. The most severe restriction is the need to take into account the total nuclear heating rate during the physics phase when the shield thickness is minimum. During the technological phase, the limiting factor is the local radiation damage of the electric insulation of the magnets. Within the constraints of the reference design, 85 cm is available on the inboard from the TF magnet to the first wall. Typical nuclear response parameters in the TF superconducting coil are insulator doses of $\sim 2 \times 10^9$ rad and nuclear heating rates of 10 to 15 kW. At least eight 1- × 2- × 0.5-m modules with two outboard sectors are necessary to test blanket/first-wall options and reactor-relevant materials.

The engineering aspects of ITER were discussed by R. Toschi and V. A. Glukhikh et al.

The main variation in the reactor assembly is that the reactor is assembled on the general base inside the cylindrical cryostat combined with the biological shield. The magnetic system includes 16 D-shaped TF coils, a multisection central solenoid, and three pairs of concentrated ring coils. The vacuum chamber includes 32 modules welded together. The inner and outer blankets are divided correspondingly into 32 and 48 sections, which are joined to the vacuum chamber and shield by bolts and shear keys. The divertor system is also sectioned along the torus.

The Nb_3Sn alloy with a current density of ~ 6 to 7×10^4 $A \cdot cm^{-2}$ as $T = 4.5$ K at $B = 12$ T is proposed for the TF coil winding. The superconductor coolant selected is HeI forced-flow Nb_3Sn. The Nb_3Sn alloy with a 40-kA critical current in the central solenoid and the ring coils is also proposed for the PF coil winding.

The main insulators are

1. high-module glass impregnated after high-temperature treatment
2. mica-containing materials with nonorganic impregnation
3. Al_2O_3 materials and ceramics.

There are no plans to publish the proceedings of this seminar.

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