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A D SAKHAROV'S 90TH BIRTHDAY COMMEMORATION

Tokamaks: from A D Sakharov to the present (the 60-year history of tokamaks)

E A Azizov

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Contents

Abstract. The paper is prepared on the basis of the report presented at the session of the Physical Sciences Division of the Russian Academy of Sciences (RAS) at the Lebedev Physical Institute, RAS on 25 May 2011, devoted to the 90-year jubilee of Academician Andrei D Sakharov — the initiator of controlled nuclear fusion research in the USSR. The 60-year history of plasma research work in toroidal devices with a longitudinal magnetic field suggested by Andrei D Sakharov and Igor E Tamm in 1950 for the confinement of fusion plasma and known at present as tokamaks is described in brief. The recent (2006) agreement among Russia, the EU, the USA, Japan, China, the Republic of Korea, and India on the joint construction of the international thermonuclear experimental reactor (ITER) in France based on the tokamak concept is discussed. Prospects for using the tokamak as a thermonuclear (14 MeV) neutron source are examined.

1. Introduction

Prior to the commencement of controlled thermonuclear fusion (CNF) research, the history of humankind presumably had not encountered a vital technical problem which required more than 20 years for its solution. This 'historical' rule is consistent with the well-known statement made by the Indian physicist Homi J Bhabha at the United Nations First International Conference on Peaceful Uses of Atomic Energy held in Geneva in 1955: "I venture to predict that a method will be found for liberating fusion energy in a controlled manner within the next two decades."

At the 2nd (1958) Geneva Conference, the English physicist P C Thonemann stated that "it is still impossible to answer the question, 'Can electrical power be generated using the light elements by themselves?' I believe that this question will be answered in the next decade. If the answer is yes, a further ten years will be required to answer the next question, 'Is such a power source economically valuable?' "

At the 1st (1961) IAEA Fusion Energy Conference in Salzburg, M N Rosenbluth (USA) delivered the summing-up report about the achievements in plasma theory and stated: "While it is unfortunately true that theorists have not told the experimentalists how to build a thermonuclear machine, it is also true that we have been looking hard for very many years for a fundamental reason why a plasma fusion reactor should be impossible and we have not found any such reason." Next, he added: "If I may make a statement from the heart, I believe the chances are very good that in twenty years or so mankind will have solved the problem of controlled fusion if only he has not lost in the meantime the far more difficult struggle against uncontrolled fusion."

Today, 60 years after the commencement of controlled thermonuclear fusion research, we may conclude that the complexity of the problem was strongly underestimated in the initial stage of the work, especially so when it is considered that the final objective, namely the demonstration of electric power production by a thermonuclear power plant, is still several decades away.

This paper gives a brief review of the development of the concept of magnetic thermal insulation of plasma, which was proposed by Andrei D Sakharov and Igor E Tamm in 1950

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and which underlies the international thermonuclear experimental reactor (ITER) project presently being implemented jointly by the European Union, India, China, Korea, Russia, the USA, and Japan in France.

2. Conception of magnetic thermal insulation of plasma. Magnetic fusion reactor

In 1950, an undistinguished event occurred, whose description by the well-known theoretical physicist and a future Full Member of the RAS Vitaly Dmitrievich Shafranov would open with the humorous couplet:

Listen, guys, to the story of yours,
It all commenced with a soldier who served.

The case in point is sergeant Oleg Aleksandrovich Lavrent'ev, who served in the Army on Sakhalin and wrote a letter to the Central Committee of the Communist Party of the Soviet Union (Bolsheviks) (CC CPSU) on 22 July 1950 to propose:

(i) the use of lithium-6 deuteride instead of liquefied deuterium and tritium in a hydrogen bomb;

(ii) the development of a system with electrostatic confinement of hot plasma for realizing controlled thermonuclear fusion.

It was not long before this letter found itself under review by Candidate of Physicomathematical Sciences Andrei Dmitrievich Sakharov (from 1953 — Doctor of Physicomathematical Sciences, Full Member of the USSR Academy of Sciences), who was working at that time on the development of a hydrogen bomb in the secret town of Arzamas-16 (Sarov). Sakharov later reminisced about that episode:

"In the summer of 1950, a letter was delivered from Beria's secretariat to our organization with a suggestion by a young sergeant Oleg Lavrent'ev, who was serving in the Army on Sakhalin. In its introductory part, the author wrote about the significance of the problem of controlled thermonuclear reaction for the future power engineering....

"...In my review I wrote that the idea of controllable thermonuclear reaction conceived by the author is extremely important.... As regards Lavrent'ev's concrete scheme I wrote that it seemed to me unrealizable, because it did not preclude the contact of hot plasma with grids... While reading the paper, I conceived the first foggy ideas of magnetic thermal insulation....

"Early in August 1950, Igor Evgen'evich [Tamm] returned from Moscow.... He expressed genuine interest in my reflections — subsequently we developed the idea of magnetic thermal insulation entirely in our collaborative work. I E's contribution was especially valuable in all calculations and estimates, as well as in the treatment of the main physical concepts — magnetic drift, magnetic surfaces, and some others [1]."

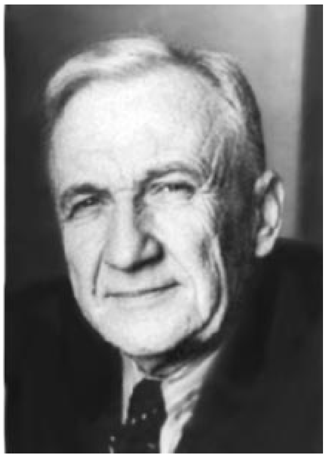
By October 1950, Sakharov and Tamm had completed a preliminary theoretical substantiation of the magnetic thermonuclear reactor (MTR) and made the first estimates of its parameters. In January 1951, I V Kurchatov organized a discussion of the project among the leading physicists involved in the Soviet Atomic project. The meeting supported the continuation of work on the MTR, and in February 1951 I V Kurchatov forwarded to Beria [1] a draft governmental resolution about organization of work on the MTR. On May 5, 1951 the resolution was approved by Stalin. According to the resolution of the USSR Council of Ministers, the task of working on the MTR issue was entrusted to the Laboratory of Measuring Instruments (LIPAN in *Russ. abbr*., presently the National Research Centre 'Kurchatov Institute'). L A Artsimovich was charged with responsibility for the whole project, and M A Leontovich became supervisor of theoretical research. The originators of the suggestion, A D Sakharov and I E Tamm, were invited as permanent consultants.

In Sakharov's report [2] he came up with the idea of confining hot plasma in a toroidal chamber with a strong longitudinal magnetic field. To compensate for the toroidal drift of charged particles, it was suggested to induce, along with the toroidal magnetic field, a poloidal magnetic field, either by passing electric current along a ring conductor placed inside the plasma or by exciting longitudinal current in the plasma itself with the help of a poloidal coil located outside of the vacuum chamber. To maintain the stability of the major discharge radius, Sakharov proposed the employment of a copper casing.

[1] Lavrentiy Beria headed the Special Committee which was founded first under the USSR State Defense Committee in August 1945 "for supervising all the work on the use of the intraatomic energy of uranium."

O A Lavrent'ev
(07.07.1926 – 10.02.2011)

A D Sakharov
(21.05.1921 – 14.12.1989)



I E Tamm
(08.07.1895 – 12.04.1971)

Initiators of controlled thermonuclear fusion research.

Table 1. Parameters of the 'large model' of an MTR.

Major torus radius	R	12 m
Minor plasma column radius	a	2 m
Toroidal magnetic field	B_t	5 T
Longitudinal plasma current	I	0.2 MA
On-axis deuton density	n_0	3×10^{20} m^{-3}
On-axis plasma temperature	T_0	100 keV
Thermonuclear power	P_{DD}	880 MW
Tritium recovery	N_T	100 g (per day)

Sakharov's calculations, which relied on classical transport coefficients and neglected the curvature of the system, resulted in the parameters of the 'large model' of an MTR, collected in Table 1 [2].

I E Tamm [3, 4] proposed the general methods of solving the kinetic equation for toroidal plasmas in the presence of a stabilizing current and showed that the thermal plasma conductivity in a torus may be substantially higher than in a straight cylinder for equal magnitudes of a longitudinal magnetic field and a current-induced magnetic field. At about the same time, G I Budker [5] called attention to the special features of the behavior of particles with a low relative longitudinal velocity, which should be rapidly lost from the plasma.

Unfortunately, this basic work by I E Tamm and G I Budker was not continued and was soon forgotten. Their findings were rediscovered and further elaborated on more than ten years later by D Pfirsch and A Schlüter (1962) [6], V D Shafranov (1965) [7], and A A Galeev and R Z Sagdeev (1967) [8].

3. 1955–1969 experiments

The first toroidal facility with a strong longitudinal magnetic field, based on the ideas of A D Sakharov and I E Tamm and known as TMF (torus with a magnetic field) (Fig. 1) [9], was constructed under I N Golovin's and N A Yavlinskii's supervision at LIPAN in 1955. This facility had the following parameters: $R = 0.8$ m, $a = 0.13$ m, $B_t = 1.5$ T, and $I = 0.26$ MA. The plasma volume, $V = 0.27$ m^3, was approximately 3500 times smaller than that in the MTR project discussed in Section 2. The porcelain discharge chamber was enclosed in a copper casing with slits; a stainless steel helix was accommodated inside the chamber near the wall to weaken the plasma–porcelain contact. The electron temperature was low ($\leqslant 10$ eV), which was due to the high level of radiative energy losses.

Subsequent facilities of this type received the name tokamak (an acronym comprising the initial syllables of the word combination '*to*roidal'naya *ka*mera *mag*nitnaya', where the letter 'g' was replaced with letter 'k' for euphony) [10].

During 1955–1965, eight facilities of this type (TMF, T-1, T-2, T-3, T-5, TM-1, TM-2, and TM-3) were built at the Kurchatov Institute, i.e. on average nearly every year saw the construction of a new facility. This underlay the relatively rapid progress in the revelation of and recovery from the 'childhood diseases' of tokamaks, like excessive inflow of impurities into the plasma and a high level of 'scattered' magnetic fields from external circuits.

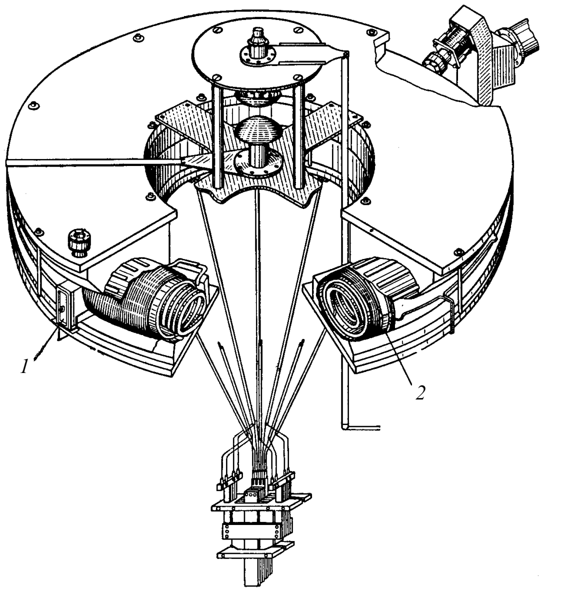


Figure 1. Schematics of the toroidal chamber with coils: *1* — window for photographing, and *2* — longitudinal magnetic field coil.

At the T-1 facility ($R = 0.62$ m, $a = 0.13$ m, $B_t = 1.0$ T, $I = 0.04$ MA) [11], it was shown for the first time that fulfillment of the condition $q_a = 5a^2B_t/RI > 1$, which is known as the Shafranov–Kruskal criterion, where q_a is the so-called stability margin, is necessary for improving the macroscopic plasma stability. Furthermore, it was shown that the plasma in the facility with a metal chamber without baking also loses 80–90% of the energy due to the radiation of impurity atoms.

The T-2 facility, which was close in parameters to the T-1 facility, had a stainless steel bellows vacuum chamber bakeable to 400–450 °C, with a limiter placed inside the chamber [12, 13]. As a result of chamber baking, the fraction of plasma radiation energy losses lowered to $\approx 30\%$. These experiments revealed the last of the childhood diseases of tokamaks. It turned out that the plasma column shifted inside the chamber by far longer distances than would be expected proceeding from the variation of plasma parameters. It was determined that the shifts were caused by the transverse component of the scattered magnetic field, which penetrated inside the chamber due to the nonideality of the conducting casing. After these experiments, all facilities under construction were equipped with special correcting and controlling coils, which cancelled out the scattered magnetic fields and controlled the position of the plasma column.

Comprehensive investigations of plasma equilibrium inside the conducting casing and a comparison of their results with theoretical ones were performed on the T-5 facility ($R = 0.625$ m, $a = 0.2$ m, $B_t = 1.2$ T, $I = 0.045$ MA) in 1961–1964. In 1965, this facility was transferred to the A F Ioffe Physical-Technical Institute (PTI) (Leningrad), where it received the name FT-1.

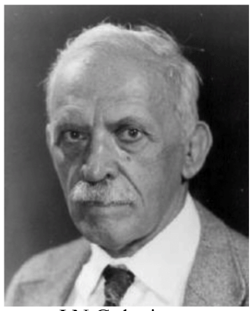
A larger facility, T-3 ($R = 1.0$ m, $a = 0.06$ m, $B_t = 4.0$ T, $I = 0.06$ MA), was built in 1960. Before long it was upgraded and was termed T-3A ($R = 1.0$ m, $a = 0.15$ m, $B_t = 3.8$ T, $I = 0.14$ MA).

In 1962, E P Gorbunov and K A Razumova obtained for the first time a discharge which retained macroscopic stability throughout the current pulse on the TM-2 facility ($R = 0.4$ m,

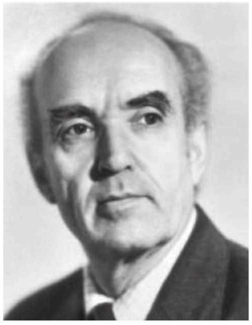

I V Kurchatov

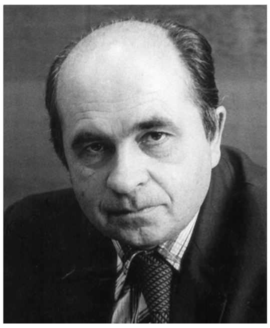

L A Artsimovich


M A Leontovich


I N Golovin


N A Yavlinskii


V D Shafranov


B B Kadomtsev


E P Velikhov

Supervisors of the thermonuclear program at the Kurchatov Institute.

$a = 0.08$ m, $B_t = 2.2$ T, $I = 0.02$ MA) with a rather large stability margin: $q_a \approx 5$. They identified the most dangerous instability in a tokamak — disruption instability [14].

The lowering of radiation energy losses and attainment of stable modes put on the agenda the question about the plasma energy transport lifetime $\tau_E = W/(P_{\mathrm{heat}} - P_{\mathrm{rad}} - \mathrm{d}W/\mathrm{d}t)$, where W is the store of plasma kinetic energy, P_{heat} is the power of heating, and P_{rad} is the radiation loss power. These investigations were carried out primarily at the TM-2 (TM-3 after the 1966 upgrade) and T-3 (T-3A after the 1967 upgrade) facilities. In these experiments, the sum of electron and ion temperatures $\langle T_e + T_i \rangle$ averaged over the section of the plasma column was determined from diamagnetic signal measurements (K A Razumova on TM-2, S V Mirnov on T-3A) and multichannel interferometric electron density measurements (E P Gorbunov). Proceeding from these data, it was possible to obtain for the first time in the history of tokamaks the similarity law for the plasma energy lifetime, which has come to be known as GMS scaling,[2] or Mirnov scaling [15]. Its more recent version assumes the form $\tau_{\mathrm{MI}} = (0.05 - 0.15) f_L a I \kappa^{0.5}$ ($f_L \sim 1$ for the L mode, and $f_L \sim 2$ for the H mode). This scaling law also predicts, correct to $\approx 50\%$, the magnitudes of τ_E in present-day tokamaks. The magnitudes of τ_E obtained in these experiments were also compared with the so-called Bohm time $\tau_B \approx 8a^2(eB/\kappa T_e)$ characteristic of turbulent plasmas. As reported by L A Artsimovich to the Second (1965) IAEA Fusion Energy Conference in Culham [16], the τ_E magnitudes in tokamaks turned out to be three times higher than the Bohm values observed in the majority of experiments at that time. In 1967, V S Strelkov reported to the 2nd Workshop on Plasma Confinement (Princeton, USA) the data on τ_E values in tokamaks, which exceeded τ_B by up to a factor of 10, while measurements on the C stellarator[3] yielded good agreement with the Bohm time [17]. Measurements of the energy spectra of fast neutral atoms at the T-3 facility were indicative of Maxwellian spectrum and yielded values of the ion temperature of several hundred electron-volts for the central regions of the plasma column (M P Petrov).

A year later, the evidences of further experiments on T-3A — magnitudes of τ_E up to 50 times higher than the Bohm time [18] — were presented at the 3rd (1968) IAEA Fusion Energy Conference in Novosibirsk. S V Mirnov, a participant in those experiments, thus described the events that followed [19]: "So important a result called for careful verification. Also there, in Novosibirsk, Director of the Culham Laboratory (England) R S Pease and L A Artsimovich reached the final agreement about the execution of a joint Soviet–English experiment in laser probing at T-3A. In spring of 1969, a group of experimentalists headed by N J Peacock came from Culham Centre for Fusion Energy to T-3A and brought experimental instrumentation. They were joined by D C Robinson, a Culham researcher working on an exchange basis at T-3A, and V V Sannikov from the Soviet side. It was precisely they, Robinson and Sannikov, who managed in July 1969, on transferring the English laser to the giant pulse mode, for the first time 'to force their way through' the plasma noise background and record the scattered laser radiation signal, which paved the way to the success of the experiment" (Fig. 2). The measured radial profiles of the electron temperature showed that the bulk electron temperature at the center of the plasma column amounts to ≈ 1 keV. These results, which were reported to the 2nd International Symposium on Plasma Confinement in Toroidal Systems in Dubna in autumn 1969, were hardly different from the diamagnetic ones [20, 21]. The doubts of the skeptics about

[2] GMS scaling is an abbreviation comprising the first letters of the surnames of all the authors of Ref. [15].

[3] A stellarator is a toroidal currentless magnetic trap, in which the magnetic configuration required for plasma confinement, unlike that in tokamaks, is produced by currents flowing in external conductors.

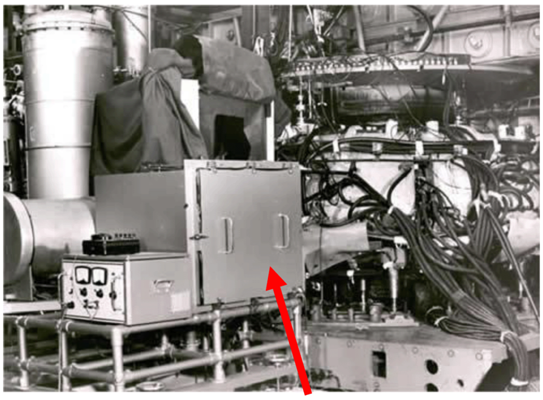


Figure 2. English laboratory equipment (indicated by an arrow) for measuring the electron temperature in T-3A by examining Thomson scattering of laser light.

the correctness of the interpretation of experimental data were thereby dispelled, and the Dubna Symposium proceeded as major triumph of tokamaks.

In parallel with the laser-assisted measurements of the electron temperature, intensity measurements of neutron radiation were made on T-3A in experiments on deuterium. The absolute magnitude of the neutron flux and the character of its temporal variation allowed a conclusion that the physical thermonuclear reaction was obtained for the first time in the T-3A tokamak in 1969 [22].

4. 1970–1990 experiments

After the Dubna Symposium, the world saw the onset of a 'tokamak boom'. While only one tokamak type facility — LT-3 in Canberra (Australia), with rather modest parameters ($R = 0.4$ m, $a = 0.1$ m, $B_t = 1$ T, and $I = 0.033$ MA) — had been constructed outside of the USSR prior to 1969, during the subsequent years tokamaks were built in 29 countries, including the USA, Japan, the majority of European countries, Canada, India, China, South Korea, Iran, Libya, and Egypt. In 1970, the C stellarator at Princeton was transformed to the ST tokamak. In all, over 200 tokamaks have been constructed in the world to date, including 31 in the USSR and Russia, 30 in the USA, 32 in Europe, and 27 in Japan [23].

The T-6 facility ($R = 0.7$ m, $a = 0.25$ m, $B_t = 1.5$ T, $I = 0.27$ MA) was constructed at the Kurchatov Institute in 1970. At this facility, the copper casing was accommodated inside a bellows vacuum chamber made of stainless steel. A gold layer was deposited onto the inner surface of the casing to reduce the impurity particle flux into the plasma. Shortening the gap between the plasma and the conducting casing was shown to improve the magnetohydrodynamic (MHD) plasma stability. Specifically, for $d/a \leqslant 1.2-1.3$ (d is the radius of the casing section), the feasibility of obtaining discharges with no disruption instability was demonstrated for $q_a \approx 1.2-1.3$, though with a shorter plasma energy lifetime. Measurements of the perturbations of the poloidal magnetic field outside the plasma with a high spatial ($\sim 15°$) and temporal (~ 1 μs) resolution revealed for the first time that the disruption instability (major disruption) begins with a buildup of the helical harmonic with $m = 2$, which is replaced with rapidly growing $m = 3$ and $m = 4$ harmonics [24]. The toroidal solenoid in T-6 consisted of 32 coils, which ensured a small ripple of the magnetic field. As it turned out, the plasma current at a low initial gas pressure was carried by runaway electrons with an energy of 10–500 keV, while the bulk plasma temperature remained low: $T_e \sim T_i \approx 10$ eV [25]. Gas preionization or the formation of a local magnetic mirror with an amplitude of about 2% transferred the discharge to the normal state.

In 1971, the T-4 facility ($R = 0.9$ m, $a = 0.16$ m, $B_t = 5$ T, $I = 0.25$ MA), the most powerful at that time, was built at the Kurchatov Institute and replaced the T-3A facility. In T-4, advantage was taken of a carbon limiter for the first time. Due to a higher current, a stronger magnetic field, and the use of a carbon limiter, record values of the electron temperature (≈ 3 keV) and ion temperature (≈ 0.65 keV) were reached at this facility.

In the same 1971, a TUMAN-2 tokamak ($R = 0.40$ m, $a = 0.08$ m, $B_t = 1.2$ T, $I = 0.08$ MA) of circular cross section with a limiter was constructed at the Ioffe PTI (Leningrad). This facility was employed to investigate the heating of plasma through its adiabatic compression by the growing toroidal magnetic field. In 1976, after the reconstruction of this facility, the toroidal magnetic field and the plasma current were raised to 1.5 T and 0.12 MA, respectively. The experiments on adiabatic compression were continued and the facility received the name TUMAN-2A.

In 1972, the first experiments on plasma heating by electron cyclotron resonance (ECR) were carried out at the TM-3 facility ($R = 0.4$ m, $a = 0.08$ m, $B_t = 2.5$ T, and $I = 0.1$ MA) [26].

In the same 1972, a TO-1 facility ($R = 0.6$ m, $a = 0.13$ m, $B_t = 1.5$ T, $I = 0.07$ MA) was put into operation, where use was first made of a feedback system to stabilize the plasma column position relative to the major radius [27]. A TO-2 facility was commissioned in 1976, which was equipped with two toroidal divertors and a system for plasma heating and current generation by Bernstein ion waves.

In 1972, L A Artsimovich and V D Shafranov revealed that the neoclassical ion thermal conductivity in tokamaks with a vertically prolate cross section should be lower than in tokamaks having circular cross section [28]. The influence of cross sectional plasma shape on discharge characteristics was experimentally investigated on the T-8 (1973–1978) and T-9 (1973–1976) tokamaks at the Kurchatov Institute.

At the T-8 facility ($R = 0.28$ m, $a = 0.05$ m, $B_t = 0.9$ T, $I = 0.024$ MA) [29], the plasma shape was set by the combined effect of the conducting casing with the elliptical cross section, the limiter, and the currents in quadrupole coils controlled by the feedback system. The highest plasma ellipticity reached in these experiments was $\kappa_{max} \approx 1.6$. A lengthening of plasma energy confinement time was observed with an increase in the ellipticity, being roughly proportional to κ^2. The feasibility of obtaining stable regimes with $\kappa_{max} \approx 2.0$ in the limiter configuration was demonstrated on the T-9 facility ($R = 0.36$ m, $a = 0.07$ m, $B_t = 1.0$ T, $I = 0.04$ MA) [30]. The T-12 facility, which was constructed on the basis of T-9, was equipped with a double-null poloidal divertor. This facility, as well as its subsequent modifications [T-13, TVD — *T*okamak *V*ytyanutyi s *D*iverterom (Prolate Tokamak with a Divertor)], was employed to investigate the stability of the plasma column with respect to vertical shifts and to develop methods of controlling the column position.

In 1976, the T-6 facility was modernized and renamed to T-11: the number of magnetic coils was lowered to 24 to make possible the tangential injection of fast neutral atomic beams.

A molybdenum liner was mounted on the inner surface of the copper casing. Initially, the system was degassed by baking it at a temperature of 400–450 °C; then the liner was processed with a glow discharge (for the first time in tokamaks) — initially in krypton, and next in helium. After this processing, the effective ion charge $Z_{\rm eff}$ in ohmic discharges in deuterium was approximately unity. Proceeding from the results of studies of thermal plasma insulation in the ohmic heating regimes, a scaling was proposed for the electron thermal conductivity, which is referred to as the Merezhkin–Mukhovatov scaling or the T-11 scaling. In 1976, experiments were performed (for the first time in the USSR) to heat the plasma by neutral particle injection with a power of ≈ 0.6 MW [31]. In 1983, in connection with a start of constructing the T-15 facility, the T-11 facility was transferred to the Branch of the I V Kurchatov Institute of Atomic Energy [presently the Troitsk Institute for Innovation and Fusion Research (TRINITI), Troitsk], where it received the name T-11M on reconstruction. In recent years, lithium technologies aimed at weakening the interaction between the plasma and the chamber walls and the limiter have been pursued at this facility [32].

In 1975, a large tokamak machine, T-10 (Fig. 3), with the following parameters: $R = 1.5$ m, $a = 0.39$ m, $B_{\rm t} = 4$ T, and $I = 600$ kA, was put into operation at the Kurchatov Institute [33]. The T-10 tokamak was equipped with a gyrotron complex providing a power supply up to 2 MW for ECR plasma heating. With ECR heating, it was first possible to obtain plasma in T-10 with a central electron temperature of ≈ 10 keV, which is only two times less than that expected of a thermonuclear reactor. The feasibility of generating current with the help of ECR was first demonstrated and a study was made of several physical effects in the plasma, which determined its confinement.

In the USA, two new facilities were created almost simultaneously with T-10: the Princeton Large Torus (PLT) [34], having nearly the same size as T-10, and Alcator [35], which was smaller in size but had a stronger longitudinal magnetic field, $B \leqslant 10$ T. In 1978, ion heating by neutral atomic beam injection was implemented in the PLT, and an ion temperature $T_{\rm i} \approx 5$ keV was obtained.

In 1976, a TUMAN-3 tokamak ($R = 0.55$ m, $a = 0.23$ m, $B_{\rm t} = 1.0$ T, $I = 0.15$ MA) with the capacity of adiabatic plasma compression and high-frequency heating was commissioned at the PTI.

A TMG facility ($R = 0.4$ m, $a = 0.078$ m, $B_{\rm t} = 3.2$ T, $I = 0.082$ MA) [36], which was developed on the basis of TM-3, was the first tokamak with a graphite first wall. It was revealed that the optimal temperature of the graphite discharge chamber amounted to ≈ 350 °C, when chemical sputtering was insignificant. Under these conditions, the plasma parameters of the TMG facility turned out to be close to those obtained in tokamaks with a metal discharge chamber.

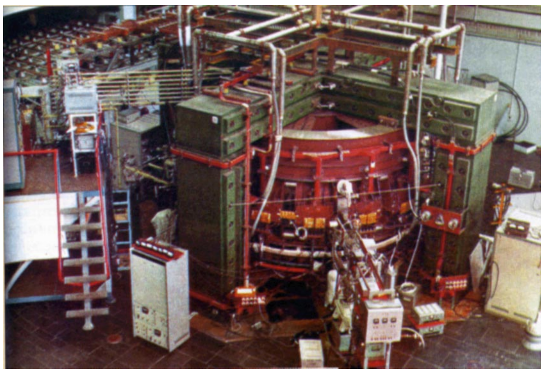


Figure 3. Photo of a T-10 tokamak taken immediately after its assembling (1975).

In 1979, a T-7 facility — the first tokamak with a toroidal magnetic field winding made of NbTi superconductor ($R = 1.2$ m, $a = 0.3$ m, $B_{\rm t} = 3$ T, $I = 0.3$ MA) — was constructed at the Kurchatov Institute. The T-7 facility was equipped with electron-cyclotron and lower hybrid heating means.

In 1982, researchers participating in the Axially Symmetric Divertor EXperiment (ASDEX) (Max-Planck Institute, Garching, Germany) were able to transfer for the first time the discharge from the divertor mode with a sufficiently high power of additional heating to the so-called high mode (H-mode) with improved energy confinement time due to a transport barrier formation at the plasma boundary [37].

The construction of facilities of progressively larger size was being continued in the USA, Europe, Japan, and the USSR. A Tokamak Fusion Test Reactor (TFTR) in Princeton (USA) and a Joint European Torus (JET) in Culham (UK) were commissioned in 1983. These nuclear fusion machines were equipped with a neutron shield which permitted operating with highly intense deuterium–tritium (DT) reactions. The biggest tokamaks are the JET and the Japanese JT-60 tokamak, whose latest modification, JT-60U, was put into operation in 1991. Both facilities have a vertically elongated plasma column cross section and a single-null divertor (Fig. 4).

In 1986, a DIII-D tokamak ($R = 1.66$ m, $a = 0.67$ m, $B_{\rm t} = 2.2$ T, $I = 3$ MA), which could operate both with single-null and double-null divertors, was commissioned in San Diego (USA). The facility was equipped with twenty independently powered poloidal coils which made it possible to optimize the shape of the cross section of the column (ellipticity, triangularity, quadraticity) and stabilize the instability localized at the plasma boundary. The total power of additional heating systems amounted to ≈ 30 MW. It has been possible to achieve at this facility the parameter $\beta = 8\pi \langle p \rangle / B_{\rm t}^2 = 12.5\%$, which is a record high value for ordinary tokamaks. A vast program of physical research in support of the ITER is being pursued at this facility [38], which is presently the largest tokamak in the USA.

The construction of two large facilities with superconducting magnetic coils was completed in 1988: Tore Supra with NbTi coils ($R = 2.25$ m, $a = 0.7$ m, $B_{\rm t} = 4.5$ T, $I = 2$ MA) in Cadarache (France), and T-15 with $\rm Nb_3Sn$ coils ($R = 2.4$ m, $a = 0.7$ m, $B_{\rm t} = 3.6$ T, $I = 1$ MA) at the Kurchatov Institute. At these facilities, the round cross section of the plasma column was bounded by limiters (Fig. 5).

In 1989, the H-mode was obtained in the ohmic heating mode at the TUMAN-3 facility. The transition to the H-mode was initiated by applying an electric potential to a peripheral probe. The transfer to the H-mode could also be initiated by a short gas puff, a fast plasma compression in the minor radius, or a pellet injection. The maximum value of $\tau_{\rm E}$ in the H-mode turned out to be an order of magnitude greater than in the ordinary ohmic mode [39]. The dependences of $\tau_{\rm E}$ on B, I, and $n_{\rm e}$ turned out to be close to those observed at large facilities in the H-mode with additional high-power heating in the absence of instability localized at

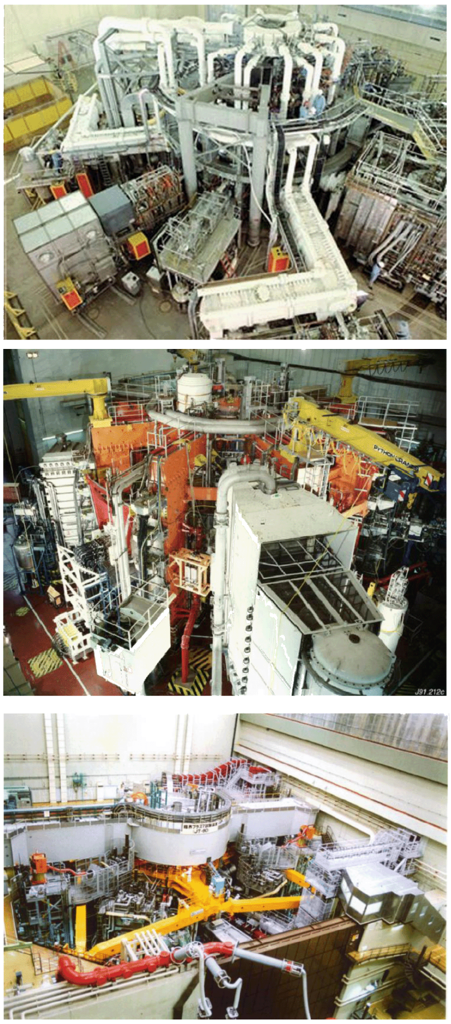


Figure 4. Three largest tokamaks with warm coils: (a) TFTR (1983–2002): $R = 2.4$ m, $a = 0.8$ m, $B_t = 6$ T, $I = 3$ MA, $P_{\rm ICRH} = 11$ MW, $P_{\rm NBI} = 39$ MW; (b) JET (since 1992): $R = 2.96$ m, $a/b = 0.96/2.1$ m, $a = 0.96$ m, $B_t = 4$ T, $I = 6$ MA, $P_{\rm ICRH} = 12$ MW, $P_{\rm NBI} = 24$ MW, $P_{\rm LH} = 7$ MW, and (c) JT-60U (1991–2010): $R = 3.4$ m, $a = 1$ m, $B_t = 4.2$ T, $I = 5$ MA, $P_{\rm ECRH} = 4$ MW, $P_{\rm ICRH} = 10$ MW, $P_{\rm NBI} = (40 + 10)$ MW, $P_{\rm LH} = (8-12)$ MW) [22]. ICRH — Ion Cyclotron Resonance Heating; ECRH — Electron Cyclotron Resonance Heating; NBI — Neutral Beam Injection Heating; LH — Lower Hybrid Heating.

the plasma boundary. The authors attributed these results to the formation of transport barriers at the plasma boundary and in the inner zone in the region where $dq/dr \sim 0$.

5. Progress in experimental research on tokamaks over the last 20 years

The most impressive event was the production of significant thermonuclear power in deuterium–tritium plasma experiments in the TFTR (11 MW, 1994) and JET (16 MW, 1997) tokamaks (Fig. 6) [40]. The maximum value of $Q = P_{\rm fus}/P_{\rm aux}$ attained at the JET facility was ≈ 0.65. These results were obtained in the modes with hot ions, $T_{\rm i} \gg T_{\rm e}$, which are not typical for the nuclear fusion reactor. In the reactor-like H-mode at the JET facility with $T_{\rm i} \approx T_{\rm e}$, a thermonuclear power $P_{\rm fus} = 3-5$ MW was obtained in a long (≈ 5 s) pulse. Similar results were achieved on the JT-60U facility in deuterium discharges: the equivalent value of $Q_{\rm eqv}$ calculated for a DT plasma amounted to $\cong 1.25$ in a short pulse for $T_{\rm i} \gg T_{\rm e}$, and to $\cong 0.5$ in the quasistationary mode [41].

Figure 7 exhibits the values of a factor $M = n_{\rm i}(0)\, T_{\rm i}(0)\, \tau_{\rm E}$ as a function of $T_{\rm i}(0)$, which were obtained in experiments on several tokamaks [42]. The shaded domains of M values in Fig. 7 correspond to the calculated values $Q = 0.1$, 1.0, and ∞ for a DT plasma. When the JET and TFTR data with $T_{\rm i} \gg T_{\rm e}$ are excluded in accordance with the aforesaid, and it is considered that the DT reaction ignition mode at $T_{\rm i}(0) \approx 30$ keV calls for a value of $M \approx 100$, one can see from this figure that the distance (in units of ΔM) from the modes with the best quasistationary discharges at the JET and JT-60U facilities to the mode with DT reaction ignition amounts to 20–30.

Figure 8 depicts the maximal thermonuclear power measured in DT discharges, or the equivalent power calculated from the DD plasma parameters in different tokamaks, $P_{\rm fus}^{\rm max}$, as a function of the calendar date between 1975 and 1995 [19]. One can see that $P_{\rm fus}^{\rm max}$ rose by a factor of 10^8 over the 20-year period. This was achieved by constructing new, larger facilities and equipping them with higher-power additional heating. On obtaining the record-high power pulses at the JET and JT-60U facilities, no further increase occurred in $P_{\rm fus}^{\rm max}$. The new superconducting facilities constructed during the last decade, which are smaller in size than JET and JT-60U, are intended for the realization and investigation of stationary discharges, rather than the attainment of high $P_{\rm fus}^{\rm max}$ values. The further increase in $P_{\rm fus}^{\rm max}$ (by a factor of 30–50 in comparison with the values attained in JET and TFTR) should occur when ITER reaches its design objectives, i.e. about 2027.

Important tasks during the last 20 years have comprised the improvement and analysis of experimental databases in different areas of tokamak's physics and derivation on their basis of empirical scalings employed to calibrate theoretical

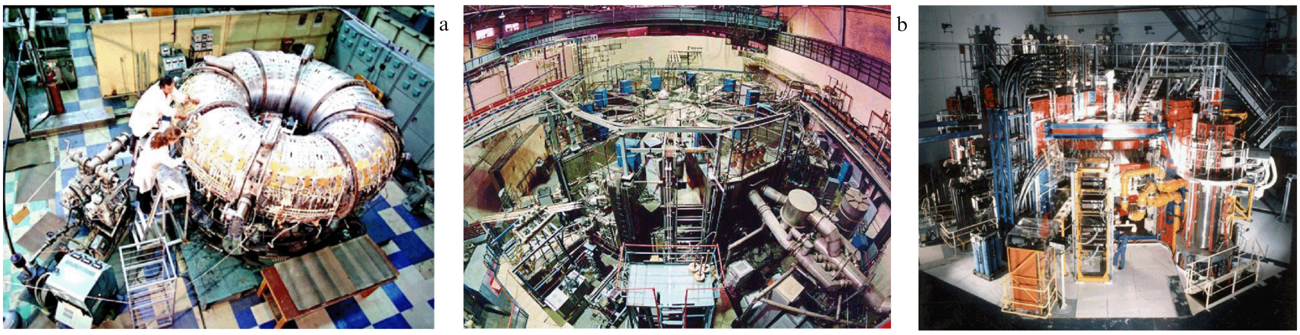


Figure 5. Tokamaks with superconducting coils: (a) assembly of the superconducting coils of T-7, (b) T-15, and (c) Tore Supra [23].

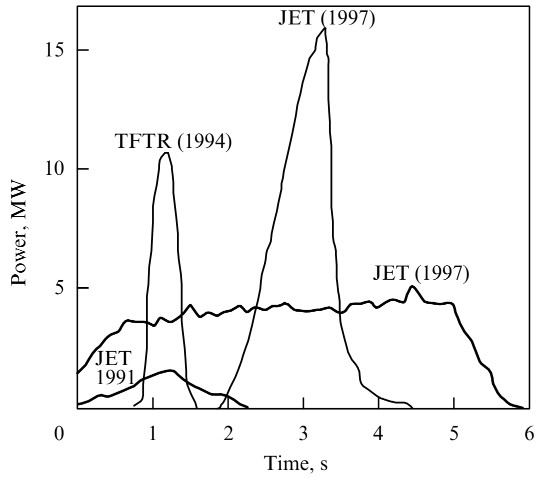


Figure 6. Thermonuclear power produced in DT experiments at the TFTR (Princeton, USA) and JET (Culham, England) facilities [40].

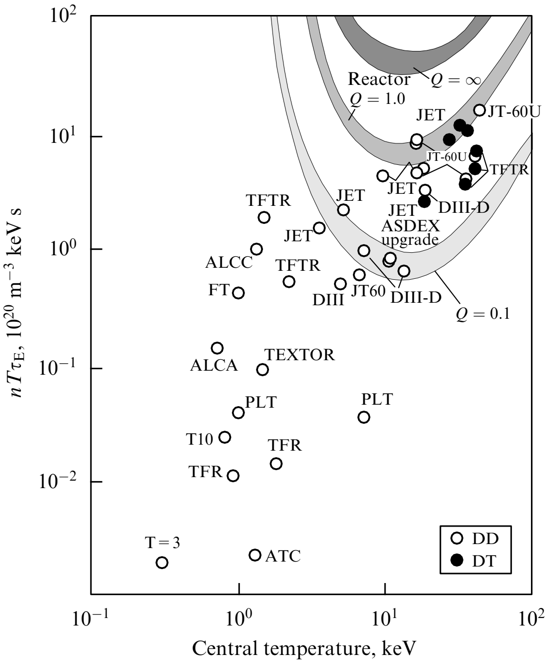


Figure 7. Experimental values of $nT\tau_E$ as a function of central ion temperature. Shown are the zones with the values of $Q \equiv P_{fus}/P_{aux} = 0.1$, 1.0, and ∞ [42].

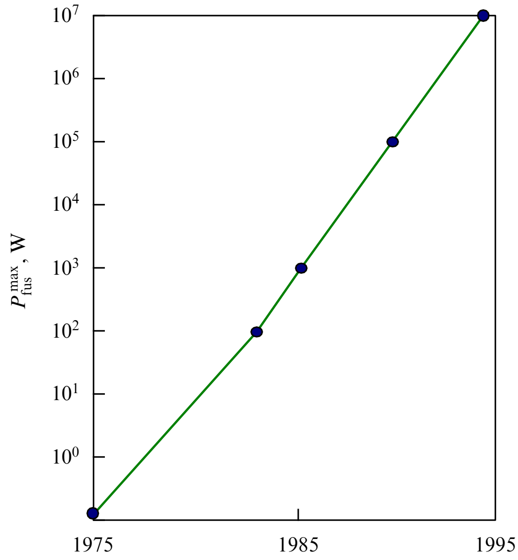


Figure 8. Growth dynamics of fusion power generated in different experimental facilities over a period of 20 years (1975–1995) [19].

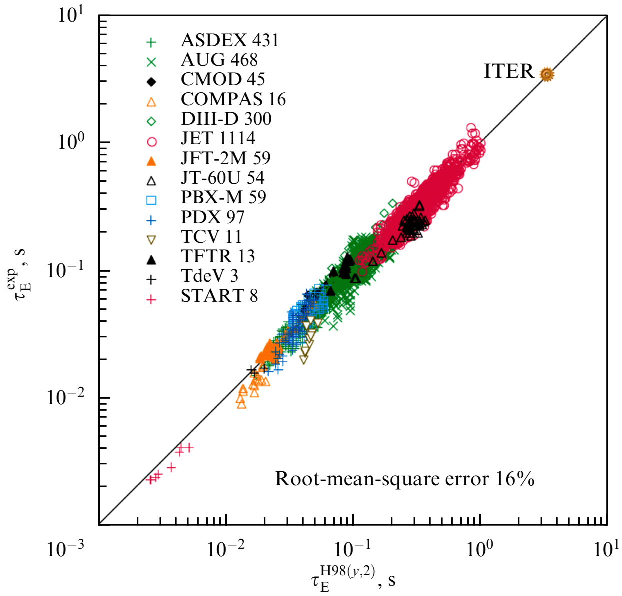


Figure 9. Comparison of the thermal plasma energy confinement time τ_E^{exp} in the H-mode for 14 tokamaks (indicated in the drawing) with empirical scaling $\tau_E^{H98(y,2)}$.

models and predict plasma parameters of future nuclear fusion machines.

By way of example, Fig. 9 demonstrates the plasma energy lifetime τ_E^{exp} for H-modes at 14 different facilities as a function of lifetime predicted by the empirical scaling IPBH98(y,2), which is based on the analysis of data from eight facilities [40]:

$$\tau_E^{H98(y,2)} = 0.0562 I^{0.93} B^{0.15} n^{0.41} P^{-0.69} R^{1.97} \kappa^{0.78} \varepsilon^{0.58} M_i^{0.19},$$

where $\kappa = V/2\pi^2 Ra^2$, $\varepsilon = a/R$, τ_E is measured in seconds, the units of measurement for I — [MA], B — [T], n — [10^{19} m^{-3}], P — [MW], and M_i — [amu].

One can see a relatively good agreement between experimental τ_E^{exp} values and the scaling predictions as τ_E^{exp} is changed approximately 400-fold. Also shown is the value of $\tau_E = 3.4$ s required to obtain $Q \sim 10$ in the ITER in inductive mode for a plasma current of 15 MA and a thermonuclear power of ≈ 500 MW.

The construction of new experimental facilities was being continued. In 1991, an ASDEX Upgrade tokamak ($R = 1.6$ m, $a = 0.5-0.8$ m, $B_t = 3.9$ T, $I = 2$ MA) was built in Garching (Germany), which had a D-shaped cross section and a single-null divertor. An improved H-mode with an internal transport barrier was obtained at this facility for the first time. In 2009, it was possible to demonstrate at this facility the feasibility of producing plasma with high parameters in a chamber with a tungsten wall, with plasma heating by fast neutral atomic beams [43]. Difficulties were encountered in obtaining stable discharges in this chamber due to the high inflow of tungsten atoms with the use of ion-cyclotron plasma heating.

In 1991, a small START tokamak ($R = 0.3$ m, $R/a = 1.25$, $B_t = 0.5$ T, $I = 0.3$ MA) with a rather high-power injection heating (≈ 1 MW) was commissioned in Culham (UK) [44]. A record value of $\beta = 8\pi\langle p\rangle/B_t^2 \approx 40\%$ was attained in this tokamak. This facility belongs to the class of so-called spherical tokamaks. Three spherical tokamaks of larger size were launched in 1999: MAST ($R/a = 1.4$, $R = 0.85$ m, $B_t = 0.4$ T, $I = 1.4$ MA) in England (Culham), NSTX ($R/a = 1.4$, $R = 0.85$ m, $B_t = 0.38$ T, $I = 1.4$ MA) in the USA (Princeton), and Globus-M ($R/a = 1.5$, $R = 0.36$ m, $B_t = 0.38$ T, $I = 0.25$ MA, b/a up to 1.8) at the PTI (St. Petersburg) [45].

The main parameter which distinguishes spherical tokamaks from ordinary ones is a substantially smaller aspect ratio ($R/a = 1.3-1.8$). This underlies the main attractive features of spherical tokamaks: their compactness, a higher limit in β, and a softer disruption instability. However, lowering the aspect ratio encounters additional technical difficulties in comparison with ordinary tokamaks. Among them is the absence of free space for accommodating the neutron shield in the central zone of the facility and, therefore, the impossibility of using superconductors for the toroidal solenoid and the central poloidal circuit in the operation with a DT plasma. Due to a small reserve of volt-seconds in the central solenoid, problems emerge with inductive discharge ignition.

At the present time, the feasibility of a noninductive discharge ignition and stationary current drive in spherical tokamaks are under investigation; in particular, under analysis are the prospects of closing toroidal solenoid coil currents along the vertical axis of the system with the employment of a liquid metal jet or a plasma column produced by a Z-pinch. Investigations of the plasma behavior in spherical tokamaks are being pursued, and the merits and demerits of their employment as fusion reactors or as fusion neutron sources are being discussed.

Three superconducting tokamaks with a D-shaped plasma cross section and a divertor have been constructed during the last decade, which were intended for studying discharges up to 300–1000 s in duration: Experimental Advanced Superconducting Tokamak (EAST) ($R = 1.7$ m, $a = 0.4$ m, $B_t = 3.5$ T, $I = 1$ MA) in Hefei (China) [46]; Korea Superconducting Tokamak Advanced Research (KSTAR) ($R = 1.8$ m, $a = 0.5$ m, $B_t = 3.5$ T, $I = 2$ MA) in Daejeon (South Korea) [47], and SST-1 ($R = 1.1$ m, $a = 0.2$ m, $B_t = 3$ T, $I = 0.22$ MA) in Gandhinagar, India [48]. Under construction in Naka (Japan) is a large superconducting JT-60SA tokamak ($R = 3.16$ m, $a = 1.02$ m, $B_t = 2.7$ T, $I = 5.5$ MA) with a double-null divertor and a ~ 100-s-long plasma current plateau (an EU–Japan collaboration project) [49]. Experiments executed at these facilities will be aimed at obtaining the physical and technological information required to optimize and monitor stationary discharges at ITER and the DEMOnstration Power Plant (DEMO).

Other important results obtained in recent decades are as follows:

(i) Discovery of a 'hybrid' regime with improved energy retention in comparison with that in the standard H-mode for which the ITER pulsed operating mode is designed. The improved hybrid mode, if realized successfully in the ITER, will make it possible to obtain the design parameters for a lower plasma current and sustain them for several thousand seconds.

(ii) Raising the limiting plasma pressure and, consequently, the limiting fusion power in the reactor due to stabilization of the neoclassical tearing instability with the help of a focused microwave radiation beam correcting the profile of the plasma current, and due to suppression of the instability that bears a relation to the finite wall resistance by compensating the scattered magnetic fields and producing a variable magnetic field of a given configuration controlled by a feedback system.

(iii) Discovery of systems with a peripheral transport barrier free from instability bursts at the plasma boundary (ELM [4]-free quiescent H-mode) and demonstration of the suppression of this instability by dint of resonance magnetic field perturbations and its attenuation by injection of hydrogen pellets.

(iv) Significant progress in the development of methods of early warning about disruption instability development in tokamaks and methods for mitigating its consequences.

Over the past 10–15 years, the experiments in Russia have been performed on six tokamaks: T-10 at the Kurchatov Institute, T-11M at TRINITI, and Globus-M, TUMAN-3M, FT-1 (until 2006), and FT-2 at the PTI. The biggest Russian tokamak T-15, which was constructed in 1988, was taken out of service in 1995 because of insufficient financing.

6. Development of tokamak plasma theory

Since 1951, theoretical investigations on controlled nuclear fusion at LIPAN have been supervised by M A Leontovich. The theoretical school he founded became a leader in the theory of high-temperature plasma for years to come.

Russian scientists constructed the theories of equilibrium, transfer processes, magnetohydrodynamic and kinetic plasma instabilities, plasma turbulence, and atomic processes and radiation, and laid the theoretical foundations for the methods of plasma heating and current generation. B B Kadomtsev laid the groundwork for the theory of transport phenomena (diffusion and thermal conduction) in turbulent plasmas. V D Shafranov is the author of papers on the theory of equilibrium and stability of plasma in the tokamak magnetic fields. He derived the equation for plasma equilibrium in two-fluid plasmastatics (the Grad–Shafranov equation), which underlies the theory of plasma equilibrium in axisymmetric magnetic configurations, and deduced the stability criterion for a plasma current column in a magnetic field, which is known as the Kruskal–Shafranov criterion.

In 1967, A A Galeev and R Z Sagdeev [8] constructed the so-call neoclassical transport theory which takes into account the presence of a special group of plasma particles trapped between the portions of force lines with a magnetic field enhanced owing to its toroidicity. They showed that the particles trapped in a rarefied high-temperature plasma play a decisive part in collisional processes of a heat and particle transport.

The results of theoretical investigations were published in collected articles entitled *Voprosy Teorii Plazmy* (*Reviews of Plasma Physics*). Beginning from 1963, 24 volumes in all have been published up to the present. The first 18 volumes were published in Russian and English, and the latest 5 volumes only in English. Recently, a decision was taken to republish volumes 19–24 in Russian at the National Research Centre 'Kurchatov Institute' and to publish the future volumes in

[4] Edge localized mode (ELM) — *Translator's comment*.

Russian with subsequent translation into English under Academician E P Velikhov's supervision.

At present, elaboration of the theory and numerical simulations are being carried out in all key areas of tokamak physics, including research in support of the ITER program. These areas comprise:

— the initial stage of discharge;

— confinement and transport processes;

— stability (MHD, turbulence);

— disruption instability, development of methods for its suppression and minimization of its detrimental consequences;

— the physics of near-wall plasma;

— controlled discharge physics (integrated scenarios, multiparametric plasma control, the physics of high-energy particles, etc.);

— methods of plasma heating and plasma current drive;

— integrated discharge simulation.

7. T-20 and INTOR projects

By 1975, a conceptional design of a T-20 reactor-scale facility ($R=5.0$ m, $a=2$ m, $B_t=3.5$ T, $I=6$ MA, $n=0.5\times10^{20}$ m^{-3}, $T=10$ keV, $P_{fus}=0.5$ GW, $q=2.3$) was prepared. Its commissioning was planned for 1985. It was conceived that T-20 would be sited in the town of Sosnovyi Bor, near the Leningrad atomic power plant, in a new center for testing experimental thermonuclear reactors (the State Nuclear Fusion Reactor Test Center, SNFRTC) [10]. An amount equivalent to US \$2 billions was to be allocated for setting up the T-20 and construction of the SNFRTC. However, more recently these plans were revised, and instead of T-20 and the SNFRTC it was decided to build about ten less ambitious projects like T-15 and TSP tokamaks, as well as a long open trap (LOT) and a rippled open trap (ROT).

In 1979, on the initiative of Soviet scientists, the IAEA established an international workgroup with the aim of exploring the feasibility of a nuclear fusion reactor based on a tokamak system. The workgroup was to determine the program, technical objectives, and facility parameters, and to appraise the existing scientific and technical basis for making a fusion reactor on an international ground, which was to demonstrate the technical feasibility of energy production from thermonuclear fusion and be the most reasonable step after the building of the T-15 (USSR), JT-60 (Japan), JET (Europe), and TFTR (USA) facilities. Workgroup participants from the USSR, the USA, Japan, and the European Community prepared a report on the scientific and technical basis and arrived at the conclusion that this basis is sufficient for designing and constructing an INternational TOkamak Reactor (INTOR) in a decade. The main characteristics of this reactor are as follows: $R=5.2$ m, $a=1.3$ m, $b/a=1.6$, $B_t\leqslant5.5$ T, $I\leqslant6$ MA, $\langle n_e\rangle=1.4\times10^{14}$ cm^{-3}, and $T=10$ keV. Unlike T-20, the INTOR was supposed to have a divertor and a vertically elongated plasma cross section.

The INTOR project was never implemented, but the results of the almost 10-year-long work of the project participants have played an important role in the development of the ITER project.

8. ITER and DEMO

In November 1985, in the name of the USSR E P Velikhov came up with a proposal to make a new-generation tokamak with the participation of the USSR, Europe, the USA, and Japan. In 1986, an agreement was reached in Geneva about the collaborative design of a facility which was to demonstrate the scientific and technological feasibility of harnessing thermonuclear reactions for peaceful purposes. During a three-year period, from 1988 to 1990, the combined effort of Soviet, American, Japanese, and European scientists and engineers resulted in the development of the conceptual project of a fusion reactor, which received the name ITER (Fig. 10). The project was aimed at attaining a self-sustained nuclear fusion reaction ($Q=\infty$) in a DT plasma with $P_{fus}\approx1$ GW in the inductive mode, and obtaining $Q\approx7$ at $P_{fus}\approx0.75$ GW in the stationary mode [50].

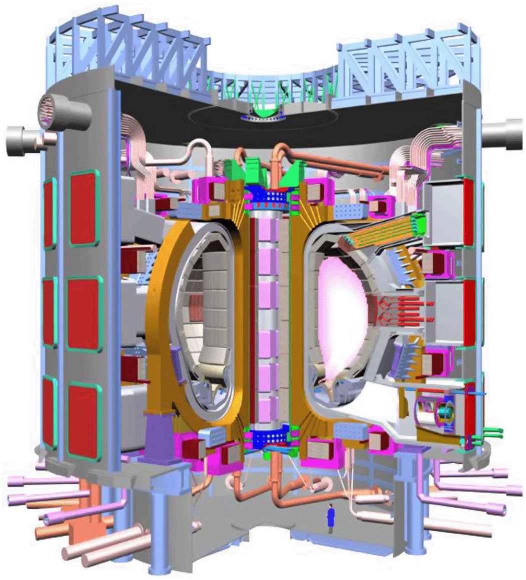


Figure 10. Poloidal section view of ITER. To estimate the ITER dimensions, a human silhouette is depicted at the bottom of the drawing.

In July 1992, the EC, Russia, the USA, and Japan signed a quadrilateral agreement on the development of an engineering design of ITER. The engineering design was completed in 1998. In the course of work on the design, several facility parameters were changed in comparison with those accepted in the conceptual project. In particular, the double-null divertor was replaced with a single-null one, the plasma volume and the thermonuclear power in the inductive mode were increased by factors of 2 and 1.4, respectively, while the magnitudes of magnetic field and plasma current were hardly changed.

In January 1999, the USA withdrew from the ITER project due to a decision by Congress. The US Congress justified this decision by the high cost of the project [10]. The EC, Russia, and Japan continued their work on the project to reduce its cost. In 2001, a second, smaller version of the technical design that was approximately two times less expensive was completed. In 2003, the USA resumed its participation in the project. China and South Korea also joined the project, and India did so in 2005. In May 2006, the consortium participants signed in Brussels an agreement about the beginning of practical implementation of the project in 2007. The first stage of construction should be completed by 2018. The first plasma production is planned for late 2019. Commencement of full-scale DT-plasma experiments is planned for 2027.

Table 2. Comparison of the design parameters of ITER [51] and one of the DEMO versions [52].

Parameter	ITER		DEMO
	Inductive mode	Stationary mode ($\geqslant 3 \times 10^3$ s)	Stationary mode ($\geqslant 10^6$ s)
Plasma current I, MA	15	9	15
Magnetic field B_t on plasma axis, T	5.3	5.18	6.8
Maximum field B_{max} on superconductor, T	11.8	11.8	14.6
Minor plasma radius a, m	2	1.85	2.1
Major plasma radius R, m	6.2	6.35	6.5
Ion temperature $T_i(0)$ on plasma axis, keV	23	25	45
MHD stability margin q_{95} at radius $r = 0.95a$	3.0	5.2	5.3
Ratio of $\langle n_e \rangle$ to Greenwald limit, $\langle n_e \rangle / n_G$	0.85	0.75	1.0
Confinement improvement factor $H_{H98(y,2)}$	1.0	1.4	1.3
$\beta_N = \beta(100aB/I)$	1.8	3.0	3.9
Bootstrap current fraction f_{BS}	0.15	0.5	0.79
Noninductive current fraction f_{NI}	0.21	1	1
Fusion power P_{fus}, MW	400	350	3000
Plasma heating power $P_{heat} = P_\alpha + P_{aux}$, MW	120	140	654
Thermal plasma energy W_{th}, MJ	320	290	1215
Fraction of radiative energy loss $f_{rad} = P_{rad}/P_{heat}$	0.5	0.57	0.86
$Q = P_{fus}/P_{aux}$	10	5	54
Disruption frequency $f_{disruption}$	~ 0.1 (per pulse)	~ 0.1 (per pulse)	$\leqslant 1$ (per year)

The final ITER version is intended for DT plasma production with $P_{fus} = 400-500$ MW and $Q \geqslant 10$ in inductive mode with a pulse duration of about 500 s. The feasibility of 'controllable burning' of the DT plasma, i.e. modes with $Q > 30$, should also be explored. Among important ITER tasks remain experiments with noninductive current drive in the quasistationary mode with a pulse duration of ≈ 3000 s for $Q \geqslant 5$, which are of immediate interest in designing the first experimental fusion power plant — DEMO. Another important ITER task is the execution of nuclear technological tests required for designing DEMO. ITER should demonstrate the combined operation of all technological systems required for DEMO and test the tritium recovery modules. The inductive and stationary mode parameters of ITER and of one of DEMO's versions with $P_{fus} = 3$ GW are collected in Table 2 [51, 52].

The surprising thing is that the plasma volume (950 m^3) and the longitudinal magnetic field intensity (5 T) in Sakharov's project (see Table 1) are practically the same as the corresponding parameters of the ITER project (831 m^3 and 5.3 T). At the same time, other parameters of these projects are markedly different. For instance, the magnitude of longitudinal current in Sakharov's project is 75 times lower than in the ITER project. Today we know that the magnitude of current in Sakharov's project could be raised to ≈ 2.8 MA without violating the hydromagnetic plasma stability for a safety margin factor $q_a = 5B_t a^2/(IR) \approx 3$ accepted for the ITER project. To avoid disruption instability with respect to the limiting density, the average plasma density should satisfy the empirical Greenwald scaling n $[10^{20}\ m^{-3}] < I\,[MA]/(\pi a^2)$ $[m^2]$, from which it follows that the plasma density must be approximately one order of magnitude lower than in Sakharov's project, even for the corrected current $I = 2.8$ MA. Accordingly, the fusion power release at the same temperature would be lower by about two orders of magnitude. Therefore, a self-sustained DD fusion reaction is out of the question in this system; however, this system would hold considerable interest with the use of a DT mixture, although not as great as ITER because of the nonoptimal shape of the plasma cross section and the absence of a divertor. This commentary serves to illustrate the exercise of knowledge gained by the nuclear fusion community on the thorny path pointed out by A D Sakharov and I E Tamm 60 years ago.

9. New stage of tokamak research in Russia

Losing the leading place in the physics and technology of tokamaks significantly lowers Russia's capabilities in mastering fusion power production. The plan of making an up-to-date divertor tokamak by upgrading the T-15 complex (Table 3), adopted in the framework of a Federal dedicated program, will serve to restore this place. Putting into operation the new tokamak in 2015 not only will permit carrying out topical studies in support of the ITER program, but will also be a major step towards the development of a fusion neutron source for hybrid systems.

Table 3. Main parameters of the modernized T-15 tokamak.

Aspect ratio	2.2
Plasma current I_P, MA	2.0
Major torus radius R, m	1.48
Elongation of plasma cross section, κ	1.9
Plasma triangularity δ	0.3 – 0.5
Plasma configuration	SN
Discharge duration, s	5 – 10
Toroidal field B_t on plasma axis, T	2.0
Flux content in solenoid $\Delta\Psi_{CS}$, Wb	6
Neutral injection power, MW	9
Microwave heating power, MW	6
Ion-cyclotron heating power, MW	6
Lower hybrid heating power, MW	4

The experimental program of the new tokamak will cover a wide range of research, including the solution to the following problems.

• Physical and technological substantiation of the demonstration thermonuclear neutron source (TNS).

• Attainment of high β_N values as a way to reducing the cost of fusion reactors and simultaneously ensuring a high density and a high temperature.

• Control of current and pressure profiles as a way to increase β_N and the confinement time τ_E.

• Implementation of improved confinement modes with inner and outer transport barriers.

• Feasibility study of modes with high β and n_e values in a stationary discharge with an all-noninductive current drive.

• Divertor optimization and investigation of the peripheral plasma effect on the global plasma discharge characteristics.

• Exerting real-time control over the stability, equilibrium, heating, and confinement of high-temperature plasma.

• Exploration of plasma interactions with various materials, including graphite, tungsten, and lithium.

• Employment of the new tokamak as a test site for trying out systems like stationary neutral injectors, and stationary high-frequency, microwave, and lower hybrid plasma heating devices, as well as for testing first-wall and divertor materials and technologies, etc.

The design of the new tokamak based on the upgraded version of the T-15 complex has already been made, and its construction will begin in 2012 (Fig. 11).

It is well known that Andrei D Sakharov and I E Tamm considered the MTR as a high-power neutron source for the production of artificial fissionable material. This idea revived in the 1970s in the form of so-called hybrid reactors, whose conception was elaborated under the supervision of E P Velikhov, I N Golovin, and V V Orlov. For various reasons, however, the work on hybrid reactors was suspended in the USSR and the USA.

At the present time, the interest in hybrid systems, including those based on tokamaks, has been rekindled. In accordance with the work plans on CNF elaborated and adopted in Russia, apart from participation in the ITER project, an industrial tokamak-based fusion neutron source should be designed and built with the objective of fuel production and transmutation of highly active nuclear reaction products for scientific and technological purposes. The new tokamak at the Kurchatov Institute may be regarded as a hydrogen prototype of the neutron source.

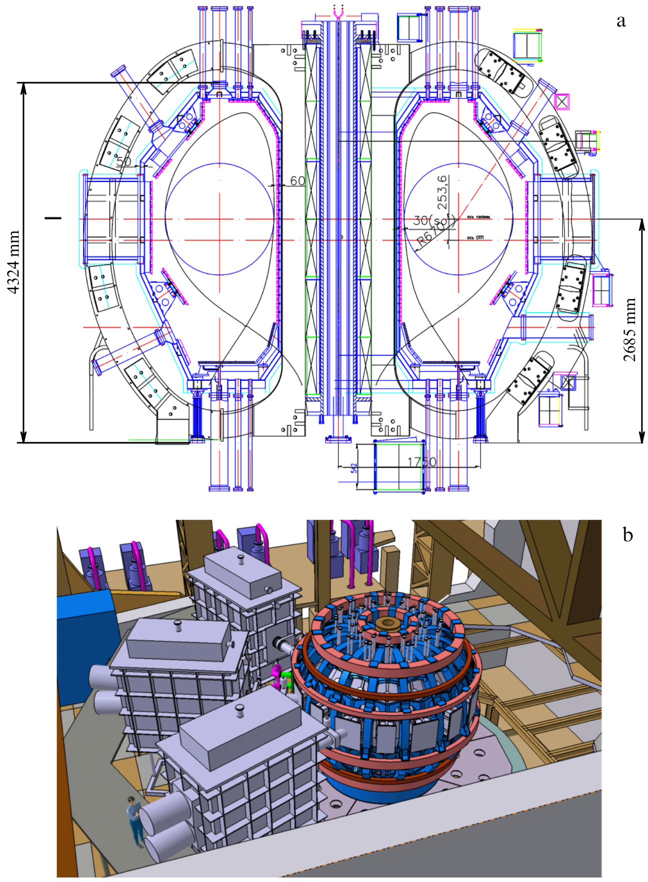


Figure 11. Cross section (a) and accommodation of modernized T-15 tokamak in experiment room (b).

Therefore, proceeding from the great body of findings made in the course of experimental and theoretical tokamak research and related technological developments, Russian physicists and engineers come back to Sakharov's original ideas.

10. Conclusions

Investigations into the magnetic thermal insulation of plasma initiated by A D Sakharov and I E Tamm 60 years ago [5] have reached the stage which allows designing in 1990–2010 and making a start on the construction of the tokamak-based International Thermonuclear Experimental Reactor (ITER). ITER is supposed to reach the design objectives in the inductive mode with $P_{fus} = 0.4 - 0.5$ GW and $Q \geqslant 10$ in 2027. Next, this should be followed by the attainment and study of long-duration modes ($\geqslant 3000$ s) with $Q \geqslant 5$, which hold great interest to extrapolations towards DEMO.

[5] The Early history of research into a controlled nuclear fusion (CNF) was presented in reviews [53, 54, 56 – 61] and a scientific literary composition [55].

The Russian Federation is among the seven member countries of the international agreement on the construction of ITER. It participates in designing and making the components and units of the facility, in developing and producing diagnostic systems and control systems, in developing discharge scenarios, and in simulating the physical processes in the ITER plasma.

The construction at the NRC 'Kurchatov Institute' of a modern tokamak equipped with a divertor and based on the modernized T-15 complex will enable carrying out extensive research in support of ITER, developing hydrogen prototypes of a fusion neutron source, and preparing personnel in the area of controlled thermonuclear fusion.

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