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50 years of fusion research

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Abstract
Fusion energy research began in the early 1950s as scientists worked to harness the awesome power of the atom for peaceful purposes. There was early optimism for a quick solution for fusion energy as there had been for fission. However, this was soon tempered by reality as the difficulty of producing and confining fusion fuel at temperatures of 100 million °C in the laboratory was appreciated. Fusion research has followed two main paths—inertial confinement fusion and magnetic confinement fusion. Over the past 50 years, there has been remarkable progress with both approaches, and now each has a solid technical foundation that has led to the construction of major facilities that are aimed at demonstrating fusion energy producing plasmas.

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(Some figures in this article are in colour only in the electronic version)

1. Introduction—fusion energy prior to 1958

The 1950s were a period of rapid progress and high expectations in science and technology. Nuclear weapons were advanced with the first fusion assisted nuclear weapons being tested in 1952. Peaceful uses of nuclear energy in the form of nuclear fission were established with the nuclear powered submarine and demonstration fission power plants. The nuclear submarine, Nautilus, was built in only three years and launched in 1954. The successful launch of Sputnik into earth orbit in October 1957 was a spectacular achievement, and fuelled the scientific competition between the West and the East. There was a general optimism that with sufficient effort science and technology could achieve almost any goal.

In the 1940s, nuclear fusion, the process that powers the sun and other stars, was identified as a possible energy source and small groups carried out early experiments [1]. During the early 1950s larger organized efforts to explore the possibility of using fusion for peaceful purposes began under the shroud of secrecy in Europe, the United States and the Soviet Union. These early efforts were carried out with high expectations for quick results. In retrospect, this was unrealistic since the funding and resource levels were very small compared with the effort to develop fission energy. For example, in the US, only $56M was expended from 1951 to 1958 on fusion research [2], which is minuscule compared with the effort on the Manhattan Project [3].

2. Two main approaches to fusion energy

It was understood very early on that fusion fuel temperatures of several hundred million °C would be needed to initiate fusion reactions. By the mid-1950s, Lawson had established the fundamental conditions needed to achieve net power output from fusion reactions [4]. His elegant power balance analysis showed that the product of fuel density (n) and plasma energy replacement time (τ_E) was a function of only plasma temperature (T), impurity content and fusion power gain (Q). The Lawson diagram ($n\tau_E$ versus T) developed from this analysis, with various refinements through the decades, is the standard for measuring progress towards fusion energy.

From the early 1950s forward, there have been two major approaches to fusion energy, inertial confinement fusion and magnetic confinement fusion. In inertial confinement fusion energy, an intense pulse of radiation or particles implodes a spherical fuel pellet producing the temperatures required for fusion, and the reacting fuel is maintained by its inertia so that the achieved $n\tau_E$ condition is sufficient to produce net energy for that pulse. The process is then repeated many times per second to produce fusion power. At the temperatures required for fusion, the fuel becomes a fully ionized plasma that can be confined by a magnetic field. In the magnetic confinement fusion approach, a magnetic bottle is formed that confines the hot plasma for long periods of time approaching steady-state conditions. Again, the Lawson $n\tau_E$ product must be satisfied to produce net energy for the plasma conditions under consideration.

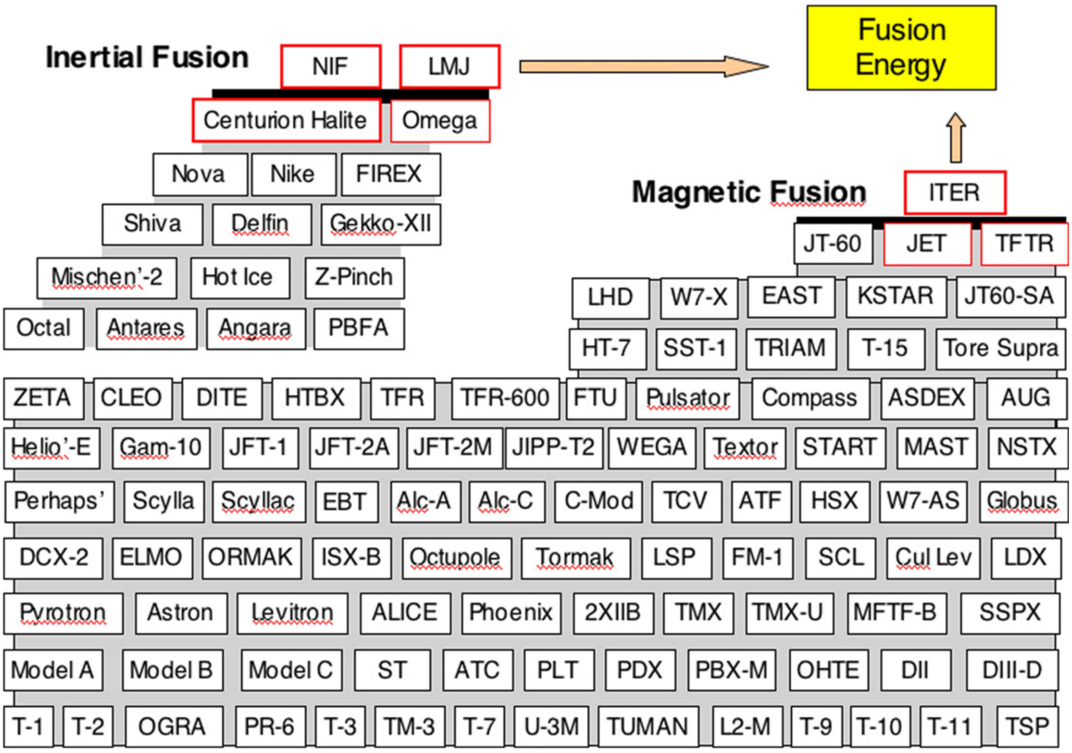

Figure 1. A solid foundation for fusion power has been made by the results from many fusion facilities over the past 50 years. The fusion program is now on the verge of demonstrating the scientific and technological feasibility of fusion power.

The activities of the past five decades have been made possible by the results from hundreds of facilities, many are illustrated in figure 1. These facilities with the accompanying theory, modeling and technology programs have provided the scientific and technological foundation for the construction of facilities capable of producing significant fusion energy.

3. Magnetic fusion

Many innovative proposals were investigated during the 1950s including various magnetic configurations to create and confine the fusion plasma, methods to enhance the fusion cross-section (e.g. muon catalysis) and electrostatic configurations. The majority of the efforts quickly focused on magnetic configurations such as tokamak, toroidal pinch, stellarator, magnetic mirrors and linear pinches. There were also studies of what a fusion power plant might look like. Sakharov [5] and Tamm considered a tokamak based deuterium fuelled system while Spitzer and others [6] analysed a large stellarator configuration using DT fuel. These studies assumed MHD plasma stability and plasma energy losses due to only bremsstrahlung radiation and classical diffusion across the magnetic field. Small laboratory experiments were initiated with a minimal understanding of plasma stability, primitive technologies and few diagnostics. Very quickly it was apparent that a quick solution was not at hand, instead the plasmas all exhibited strong collective instabilities that prevented plasma parameters from exceeding $\sim$100 eV, far from the 10 keV needed for fusion. In the few cases where confinement could be measured, the plasma diffusion was much larger than classical diffusion, and closely resembled the Bohm diffusion [7] observed in the Calutron ion sources used to separate uranium isotopes.

3.1. *Geneva Fusion Conference—1958*

As the difficulty of producing fusion in the laboratory became apparent in the US, Europe and the Soviet Union, discussions were initiated to declassify magnetic fusion research. This was remarkable in the tense atmosphere of the cold war that existed at that time. The first public disclosure of fusion research was made by USSR Atomic Energy Institute Director, Kurchatov, in a report given at Harwell (UK) in 1956 [8]. The 1958 meeting on Peaceful Uses of Atomic Energy at Geneva was planned to be a unveiling of previously secret fusion programmes carried out by all parties. In the US, ambitious plans were made for elaborate exhibits featuring operating experiments. The call went out from Washington for an exhibit that would demonstrate the production of thermonuclear neutrons [9]. There was great anxiety in the US where researchers had been struggling with the challenges of instabilities and Bohm diffusion. Some feared the Soviets had already solved that problem and their success would humiliate the Americans. As it turned out the Soviets had been experiencing the same challenges with instabilities. At the 1958 meeting, the status of various small scale experiments was reported and all exhibited strong instabilities (figure 2). Several theoretical papers described requirements for plasma equilibrium, and the use of energy conservation to predict plasma stability. This 'energy principle' was formally described in a number of papers in the late 1950s [11, 12] and at the 1958 Geneva meeting [13–15]. During this meeting and the years following, a strong collegial bond was formed within

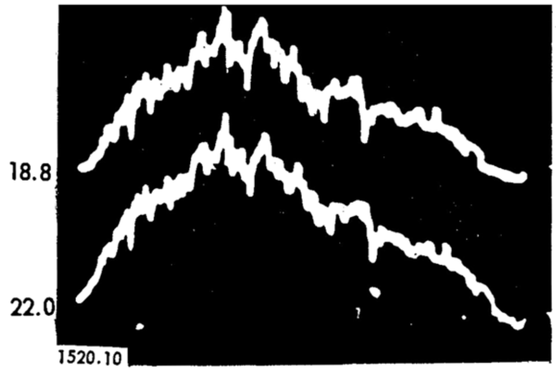

Figure 2. Typical instability from the 1958 meeting. Poloidal magnetic field at 18.8 and 22 cm from the magnetic axis of ZETA. Sweep duration was ≈4 ms [10].

the international fusion research community that provided the basis for an effective and enduring international collaboration to harness the power of fusion.

3.2. *Fusion plasma physics, a new scientific discipline was born in the 1960s*

Instabilities and anomalous transport plagued confinement experiments into the early 1960s. During the 1960s, the foundations of fusion plasma theory were laid as specific experiments were carried out to test and validate the emerging theories of plasma behaviour in a magnetic field. The energy principle developed in the late 1950s became a standard technique for evaluating the macroscopic stability of an ideal plasma in various magnetic configurations. MHD based theory was expanded to include anisotropic pressure plasmas, and led to the idea of a minimum B or magnetic well configuration. This was first demonstrated dramatically by Ioffe [16, 17] in 1961 when following a suggestion by Artsimovich, a hexapole magnetic cusp was superimposed on a MHD unstable mirror plasma. This configuration stabilized the interchange instability increasing the confinement time by a factor of 30 (figure 3). As experiments and measurements became more refined, it was found that nearly all toroidal experimental results could be characterized by Bohm scaling [18] where the energy confinement time scaled as $\tau_B \sim Ba^2/T_e$ and a major theoretical and experimental effort was made to understand the cause of Bohm diffusion. Ohkawa and Kerst [19] put forward the idea of minimum average B stability in a torus using a toroidal multipole field created by current-carrying ring(s) within the plasma. Experiments during the mid- to late 1960s confirmed that interchange instabilities could be stabilized by this technique with confinement times increasing to $>50\tau_B$ in low temperature (5–10 eV) plasmas [20].

A major advance during this period was to include the effect of finite plasma resistivity on MHD stability, that allowed additional modes to be unstable [21]. Linear stability analysis of collisionless plasmas with spatial gradients in idealized geometries revealed a plethora of micro-instabilities with the potential to cause anomalous diffusion similar to Bohm diffusion. The Trieste School [22] with Kadomtsev, Rosenbluth and other leading theorists became a source of new ideas and theoretical models. Experiments on low temperature linear and toroidal plasmas were designed specifically to test

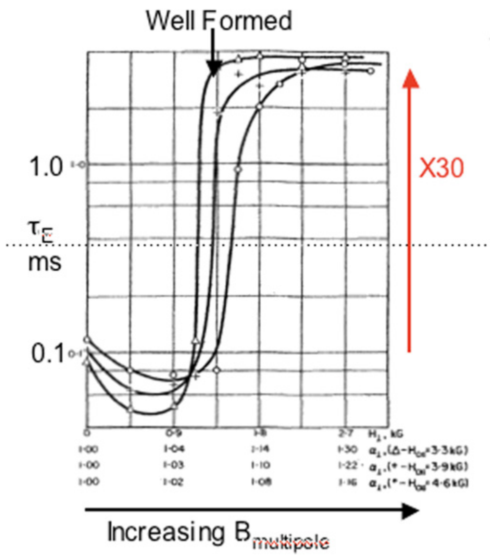

Figure 3. Minimum-B mirror experiment by Ioffe showing increase in confinement when a magnetic well was formed. Adapted from figure 3 in [17], reprinted with permission from the Institute of Physics Publishing.

the properties of micro-instabilities. Landau damping was a key feature of micro-instabilities, and in a classic experiment, Malmberg [23] verified the key predictions of the Landau damping theory. Theory was also used to investigate the propagation of radiofrequency waves first in a cold plasma and then in a warm plasma [24–26]. This work and lab experiments provided the basis for radio frequency heating by ion cyclotron and electron cyclotron waves.

At the 1965 IAEA meeting in Culham, most experiments continued to be limited by Bohm diffusion, but a quiescent period was discovered while analysing the current ramp down phase of ZETA [27]. The quiescent period coincided with the formation of a reversed current layer that had strong magnetic shear, and provided evidence that magnetic shear could stabilize instabilities in a toroidal plasma. The spontaneous generation of reversed fields in toroidal plasmas was shown by Taylor [28] to be a consequence of relaxation under constraints to a minimum energy state.

During this period of time as fusion leaders struggled over what direction to take, some of the US leaders in inertial fusion and magnetic fusion met at Princeton in ∼1967 during a visit by Edward Teller and Lewis Straus, the Chairman of the Atomic Energy Commission during the 1950s (figure 4).

3.3. *T-3 Breaks the Bohm barrier in 1968–1969, tokamaks proliferate*

At the 1968 IAEA meeting at Novosibirsk, the T-3 tokamak reported improved confinement with central electron temperatures reaching ∼1 keV and confinement times $>30\tau_B$ [29] (figure 5(*a*)). The electron temperature was measured using soft x-ray diagnostics and diamagnetic loop measurement of the total plasma stored energy. Questions were raised regarding the validity of these measurements, since they could have been contaminated by non-thermal slide-away electrons that were often present in toroidal discharges.

To resolve these questions, a collaboration was proposed by Artsimovich in which a team from Culham Laboratory

Figure 4. US Fusion Strategy Discussion in 1967 with Tom Stix, Harold Furth, Edward Teller, Lewis Strauss, Marshall Rosenbluth and Mel Gottlieb. Courtesy of Princeton Plasma Physics Laboratory (PPPL).

would transport a Thomson scattering system to T-3 at Kurchatov Laboratory [30]. This was the first major international fusion research collaboration and the transfer of equipment and experiments were carried out in less than a year. At a special fusion conference held in Dubna, USSR in 1969, the results were announced confirming that indeed 1 keV had been achieved (figure 5(*b*)) [31]. This confirmed that the long standing Bohm barrier had been broken with $\tau_E > 30\tau_{\rm Bohm}$ in a hot plasma. Skeptics were converted to advocates overnight, and participants from around the world raced back to their home laboratories to initiate tokamak experiments. The Model C stellarator was converted into the Symmetric Tokamak (ST) in just 6 months. With advanced diagnostics and one of the first computerized data analysis systems, ST results quickly confirmed the improved confinement observed on T-3 and uncovered additional new physics insights [32].

Others around the world were also quickly building a new generation of T-3 scale tokamaks such as ORMAK, Alcator and ATC in the US, CLEO, TFR and FT in Europe and JFT-2 and JFT2a (DIVA) in Japan. In addition, the construction of medium-scale tokamaks (PLT and T-10) with plasma currents in the 1 MA range was initiated in the early 1970s. These devices extended the breadth of tokamak physics and provided test beds for auxiliary heating using ECRH, ICRH and neutral beam injection.

The advances in tokamak performance also encouraged advocates of other configurations—magnetic mirrors, theta pinches and toroidal pinches—to propose and construct medium-size experiments. In the 1970s, the fusion programme strategy was very ambitious using overlapping experimental steps, with new devices beginning construction before experimental results from the preceding generation were available. Fusion laboratories and government organizations around the world began to make long range plans to develop fusion energy based on systems studies of fusion power plants [33, 34] to identify the characteristics of a fusion power plant, and define the technical steps needed to develop fusion energy.

3.4. Mid-1970s oil embargo spurs large increase in fusion research

When the oil embargo hit in 1973, there was a very large increase in funding for Applied Energy R&D as the price of oil surged from \$5 to \$40 per barrel. The international magnetic fusion research programme had momentum and was ready to put forward proposals for fusion facilities. The US fusion budget increases were spectacular rising from \$30M in 1972 to \$450M in 1984 [35], with similar increases in other countries. Several medium-size tokamaks—DIII, DIVA, PDX, Alcator C, ASDEX, JFT-M, were quickly proposed and funded to investigate specific areas—plasma cross-section shaping, divertors and high field/high-density regimes.

Less than a decade after the breakthrough results from T-3, the construction of four large tokamaks (TFTR, JET, JT-60 and T-15) was initiated; these were large bold steps with roughly a 10-fold increase in size and plasma current relative to previous tokamaks, see table 1. Auxiliary heating was proposed to increase by a factor of ∼20 to 30 relative to previous experiments.

The cost of each of the large tokamaks was in the ∼\$500M range. They had the ambitious goal summarized roughly as achieving near fusion plasma conditions thereby demonstrating the possibility of fusion energy. It is important to remember that each went forward before key R&D (high temperature plasma confinement, tests of cross-section shaping, high energy neutral beams and PFC/divertors) had been completed. TFTR featured conservative physics with strong auxiliary heating and began construction in 1976. JET initiated construction in 1977 and took the particularly ambitious step of incorporating significant plasma and toroidal field coil elongation prior to definitive results from the Doublet program. JT-60 used a poloidal divertor and began construction in 1978. All the previous large tokamaks had copper conductor coils, while T-15 was designed with superconducting Nb_3Sn toroidal field coils and began construction in 1979.

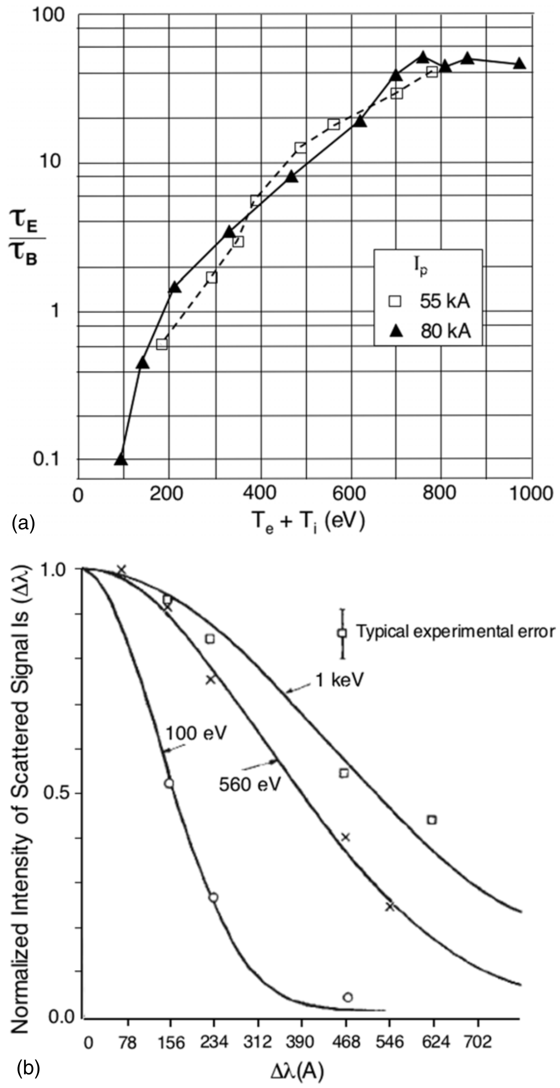

Figure 5. (*a*) Confinement reported at Novosibirsk IAEA 1968 based on x-ray and diamagnetic loops. Reprinted with permission of IAEA. (*b*) T-3 1 keV temperature profile from Thomson scattering. Adapted from figure 3 in Peacock N J *et al*, 1969 *Nature* **224** 488 with permission from Macmillan Publishers Ltd: Nature copyright (1969).

As the large tokamaks were coming off the drawing boards and moving into construction, progress on the medium scale tokamaks continued at a rapid pace. Plasma diagnostics were improved and Thomson scattering became the standard with computerized machine control and data acquisition beginning to be employed on major experiments. ICRF heating was applied successfully to stellarators in the late 1950s and 1960s and to tokamaks such as the ST [36] in the early 1970s. Minority hybrid resonance heating was first demonstrated on TFR [37] in the late 1970s, then extended by work on PLT, TFTR and JET and eventually to the reference scenario for ITER. ECRH heating using gyrotrons was applied at significant levels to the TM-3 tokamak in 1973. Neutral beam heating was applied initially at levels less than ohmic heating on ORMAK, CLEO, ATC and TFR during the mid-1970s and was then steadily increased to levels exceeding ohmic heating by ~1976 with T_i rising to ~2 keV on TFR and ORMAK [38, 39].

Optimism about confinement increased in the late 1970s when PLT using neutral beam injection into low density plasmas achieved ion temperatures, $T_i \sim 5.8$ keV [40] exceeding for the first time in a tokamak, the minimum T_i needed for fusion (figure 6). This reduced concerns that theorists had raised about the trapped ion micro-instability preventing the large tokamaks from reaching their goals. Strong ohmic heating in the high field tokamak Alcator-A achieved $n\tau_E \approx 3 \times 10^{19}\ \mathrm{m^{-3}}$ s [41] with a favourable scaling $\tau_E \sim na^2$. These results and others from around the world maintained the support needed to complete the construction of the four large tokamaks then well under way.

3.5. Strong auxiliary heating reveals new trends in 1980

As the auxiliary heating power began to exceed ohmic heating, controlled experiments revealed the true scaling of the global confinement time. The first hints of trouble were seen in TFR and ISX-B where the confinement time was observed to decrease as auxiliary heating power was increased, a typical example from 1981 is shown in figure 7 [42]. This type of adverse confinement scaling was observed on all tokamaks once the auxiliary heating began to exceed ohmic heating. It was named low mode confinement, and the confinement scaling results were systematized by Goldston in terms of engineering parameters as $\tau_E \sim I_p/P_{aux}^{0.5}$ [43]. This scaling is a weaker form of Bohm scaling and would prevent the large tokamaks under construction from attaining their goals, and would project to unreasonably large fusion reactors.

3.6. H-mode discovered on ASDEX in 1982

As the generation of specialized tokamaks initiated in the mid-1970s came into operation with auxiliary heating, ASDEX operating in a divertor configuration reproduced the L-mode scaling results when the ion grad-B drift was away from the divertor x-point. However, when the ion grad-B drift was towards the x-point, the density (figure 8) and hence confinement doubled relative to the L-mode [44]. This new high confinement mode of operation, or H-mode, was quickly confirmed on other medium-size tokamaks and provided a path for the large tokamaks to achieve their objectives. Detailed physics studies of the H-mode over the next decades helped provide a fundamental understanding of plasma transport in a tokamak. Now thirty years later, the H-mode with a solid experimental base is very robust and has been chosen as the baseline operating mode for ITER [45]. In late 1983, Alcator C, using pellet injection, achieved $n\tau_E \approx 6 \times 10^{19}\ \mathrm{m^{-3}}$ s at $T = 1.5$ keV [46]. The $n\tau_E$ value is comparable to that needed for breakeven but a much higher temperature of ~10 keV would be required for breakeven in a DT plasma. The effectiveness of neutral beam, ECRH, ICRF heating and lower hybrid current drive was demonstrated on a number of medium-size tokamaks during this period.

3.7. Tokamak optimization

As understanding of tokamak plasma behaviour improved, it was becoming clear that plasma performance should and could be improved by optimizing the plasma cross-section shape and edge plasma wall interaction. The potential

Table 1. Large tokamaks initiated in the 1970s.

	R(m)/a(m)/κ	B(T) Design/ achieved	Ops start	I_p (MA) Design/ achieved	Aux heating (MW) Design/achieved
TFTR	2.5/0.9/1	5.2/5.6	1982	2.5/3.0	33/40
JET	2.96/1.2/1.7	3.4/4.0	1983	2.8/7	5/25
JT-60/U	3.4/1.0/1	4.5	1985/89	3/5	20/40
T-15	2.4/0.7/1	3.6	1988	1.8/1	15/2

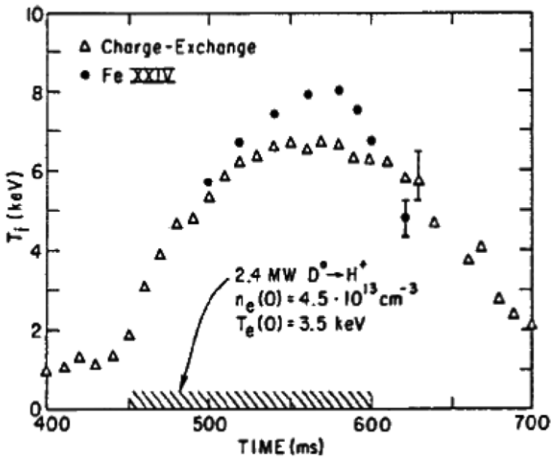

Figure 6. Ion temperature increased above the minimum ignition value of 4.5 keV for the first time. Reprinted with permission from Eubank H P *et al*, 1979 *Phys. Rev. Lett.* **43** 270. Copyright 1978 by the American Physical Society. http://prola.aps.org/abstract/PRL/v43/i4/p270_1

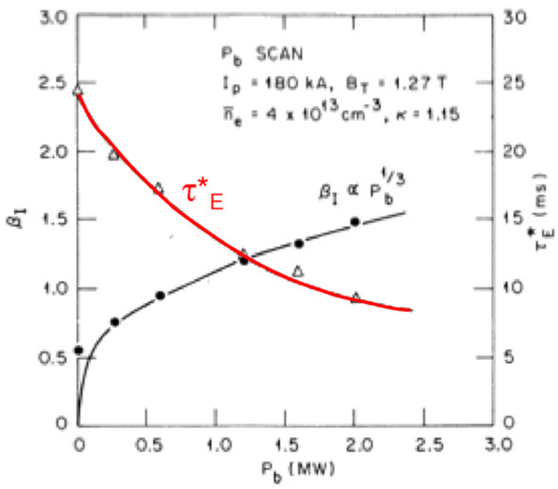

Figure 7. Example of L-mode confinement degradation with increased heating power. Reprinted with permission of the IAEA from [42].

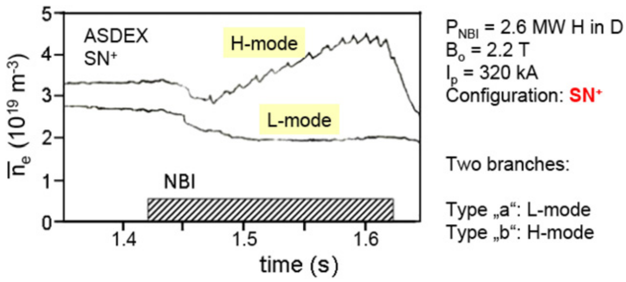

Figure 8. The discovery of the H-mode in ASDEX. Courtesy F. Wagner, Max Planck Institute for Plasma Physics, Garching.

advantages of plasma cross-section shaping were pointed out by Ohkawa [47] at a special evening session at the 1968 IAEA meeting, when he proposed replacing the copper current-carrying rings in a strongly stable toroidal multipole with localized plasma currents. This plasma current multipole (PCM) configuration evolved into the doublet series, and Doublet II was among the first experiments to observe the benefits of cross-section shaping on confinement in ~mid-1970s, and these results were extended on the larger Doublet III in collaboration with JAERI. It was found that even a single localized plasma current in a vertically elongated cross-section could support increased plasma currents and hence achieve higher beta and confinement time. In addition, quantitative predictions based theoretical understanding of plasma beta and experimental measurements of confinement time were developed. The beta limit formulated by Troyon [48] and Sykes [49] was consistent with the experimental data and provided a basis for predicting the beta limit in tokamak configurations. Similarly, the empirical confinement scaling for L-mode and H-mode developed by Goldston [43] and refined by others is embedded in the scaling relations used to design ITER [45]. The beta limit and confinement scaling was the basis for the design of a second generation of flexible optimized tokamaks (DIII-D, ASDEX Upgrade, JT-60U, PBX, Alcator C-Mod) that were built in the mid- to late 1980s to extend the understanding of plasma behaviour in tokamaks, and to explore new techniques for optimizing tokamak plasma performance.

3.8. Large tokamaks extend plasma frontier towards power plant regime

Three large tokamaks (TFTR, JET and JT60) came into operation in the early 1980s (figure 9). Despite the huge step in size and complexity, they were each built in about 6 years as the:

- US decided to build TFTR in 1974, construction approved 1976 with 1st plasma December 1982,
- EU set JET design center at Culham 1973, site decision September 1977 with 1st Plasma June 1983 and
- JA decided to build JT-60 at Naka 1979 with 1st Plasma April 1985.

Within 3 years after achieving first plasma, each of the three large tokamaks had accessed new regimes of plasma behaviour. By 1986, TFTR had achieved reactor temperatures $T_i \sim 17$ keV [50, 51], identified the bootstrap current [52], and extended $n\tau_E$ to record values at low T [53]. JET had

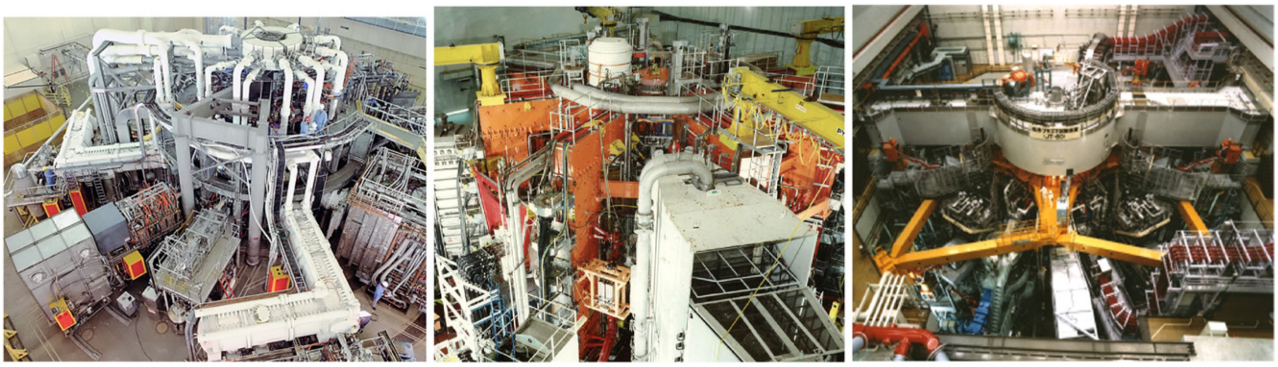

Figure 9. TFTR, JET and JT-60U devices in their test cells. Courtesy Princeton Plasma Physics Laboratory, European Fusion Development Agreement: JET, and Japan Atomic Energy Agency: JT-60, respectively.

extended the H-Mode to large tokamaks with a provisional divertor thereby doubling $n\tau_E$ to values of the Lawson product $n\tau_E$ 300 times larger than those achieved on T-3 [54, 55]. The JT-60 Phase I goals were achieved by 1988 [56, 57] and an ambitious upgrade to JT-60U was initiated in 1989.

Despite budget reductions beginning in the mid-1980s, steady progress towards fusion plasma parameters by the three large tokamaks continued in the late 1980s reaching temperatures $T_i > 35$ keV, $n\tau_E$ approaching breakeven values in high temperature deuterium plasmas, and plasma durations up to 60 s at lower parameters [58–61]. The scientific base of the program was strengthened by results from specialized medium-scale tokamaks, Tore Supra (a long-pulse tokamak with superconducting coils) as well as results from the broader supporting program in stellarators and other configurations. Construction of JT-60U in Japan was initiated in 1989, and completed in 1991.

In addition to the extension of the physics frontier towards fusion plasma conditions, the large tokamaks demonstrated that the complex technology required for magnetic fusion could be built and operated reliably. The physics results were enabled by major advances in plasma technologies—neutral beam injection, RF heating, pellet injection, plasma facing components and plasma diagnostics.

During this period, international fusion research was characterized by healthy competitive collaboration with advancements in machine capability, new detailed plasma diagnostics and supporting theory stimulating discovery of new regimes and their rapid exploration. It was a period of outstanding scientific progress made possible by the large investments made during the late 1970s and early 1980s.

3.9. Fusion temperatures attained, fusion confinement one step away

During the years since the first 1958 Geneva meeting, plasma conditions advanced steadily towards fusion conditions as each new generation of fusion facilities was brought into operation. By the early 1990s, the astronomical temperatures required for fusion (200 million °C) were achieved routinely on several experiments. Since the 1958 Geneva meeting, the plasma temperatures have been increased by 3000 and plasma confinement $n\tau_E$ by 3000, with the familiar figure of merit $n\tau_E T$ being increased by 10^7 as illustrated on the Lawson diagram (figure 10).

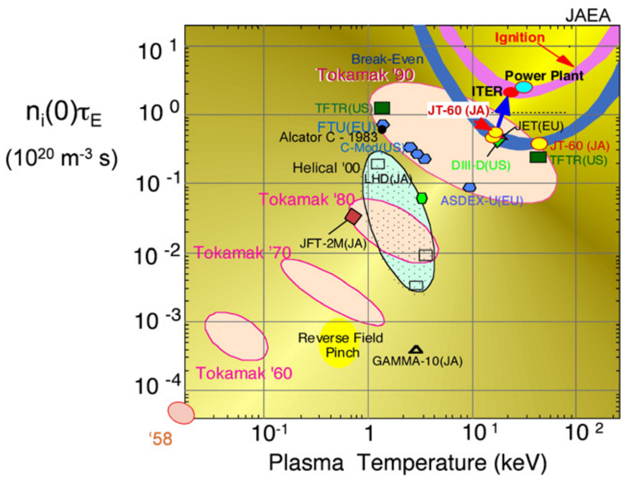

Figure 10. Lawson diagram for magnetic fusion illustrating progress over 50 years. Courtesy of the Japan Atomic Energy Agency: Naka Fusion.

The Lawson confinement parameter $n\tau_E$ has been increased to values near that required for breakeven $Q = 1$ in a DT plasma, and within a factor of 10 of that required for a fusion power plant. The final step to fusion power plant plasma conditions is to be achieved in ITER.

3.10. Significant fusion power (>10 MW) produced in the 1990s

In 1991, JET carried out experiments [62] with 10% tritium (T) added to a deuterium (D) plasma for two pulses, each producing a peak fusion power of ≈1.7 MW and a fusion energy of 2 MJ per pulse. The fusion gain, $Q = P_{\text{fusion}}/P_{\text{heat}}$ was ≈0.15. These experiments demonstrated safe operation of tritium systems with up to ∼0.2 g T, and confirmed that tritium retention in the carbon plasma facing components was ∼30%, similar to earlier results using the trace T produced by DD fusion reactions.

In 1993, TFTR began a three year campaign using 50% T/50% D, the fusion fuel mixture envisioned for the first fusion power plants. Peak fusion powers of 10.7 MW for ∼1 s were achieved with a maximum of 7.5 MJ of fusion energy per pulse. A maximum fusion gain of 0.3 was achieved with fusion power density in the core of the plasma in the range

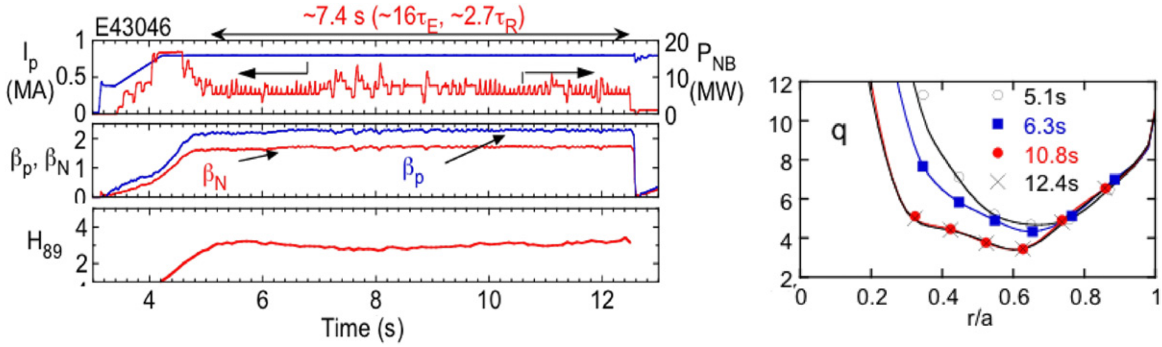

Figure 11. Example of the steady-state AT mode for a tokamak from JT-60U, 75% bootstrap current with 95% non-inductive drive. Additional optimization is required to increase β_N and decrease q. From [75], courtesy of the Japan Atomic Energy Agency JT-60.

of 2.5 MW m^{-3}. These experiments also provided the first evidence for alpha particle collective effects including alpha heating of electrons consistent with the measured alpha particle slowing down spectrum and alpha-driven toroidal Alfvén eigen modes. ICRF heating of a DT plasma using second harmonic tritium, a possible heating scenario for ITER, was also demonstrated. During the three years of operation ∼1000 DT pulses were run producing ∼1 GJ of fusion energy with 1 MCi (100 g T) being processed through the tritium system. Near the end of TFTR operation, the tritium system was operated in a closed loop with the tritium in the plasma exhaust recovered and separated overnight and reused the following day. TFTR operation was completed in 1997, and the facility was decommissioned [63, 64].

During the mid-1990s, deuterium plasma experiments were extended to high performance using the reversed-shear advanced mode envisioned as a possible plasma operating mode for steady-state tokamak based power plants. Improved ion confinement with values near the neoclassical limit was observed in TFTR and DIII-D using deuterium plasmas, with DIII projecting an equivalent DT fusion gain of ≈0.3. JT-60U established a record $n\tau_E T_i$ and equivalent DT fusion gain using the reversed-shear advanced tokamak mode in 1996 [65]. In addition, the deuterium experiments extended the bootstrap current and current drive experiments into the megaamperes range.

In the fall of 1997, JET carried out a series of ∼100 DT pulses reaching fusion power levels of 16 MW for the ∼1 s, and 22 MJ of fusion energy per pulse using longer-duration lower-power pulses. The maximum fusion gains achieved were $Q = P_{fusion}/P_{heat} \approx 0.65$. JET also extended alpha heating experiments and ICRF heating scenarios in DT. A near ITER scale closed cycle tritium plant was tested successfully during this phase. JET made a major contribution to fusion technology by demonstrating remote handling of components inside the vacuum vessel [66].

3.11. Sustainment of fusion plasma conditions, the next challenge in magnetic fusion research

The highest performance plasma conditions described above were transient with high performance lasting $\sim 1\tau_E$. Steady-state plasma operation is a highly desirable characteristic for a fusion power plant [67]. This will require a sustained magnetic configuration such as the stellarator, which is inherently steady state or an advanced tokamak with a very large bootstrap current fraction and the remaining plasma current driven off axis by an efficient external current drive system [68, 69]. The theory of lower hybrid current drive was developed by Fisch [70] in 1977, and was quickly confirmed in the late 1970s and early 1980s by a number of experiments on JFT-2, WT-2, Versator II and PLT. Experiments on a small superconducting coil tokamak, TRIAM-1A in Japan, demonstrated that a tokamak plasma could be sustained by lower hybrid current drive for over 5 h at low parameters [71]. In addition, the heat from high power density plasma exhaust and nuclear heating of first wall components will have to be controlled and distributed to stay within the limits of materials. Progress in this area was demonstrated by using lower hybrid current drive to sustain the plasma in Tore Supra while a thermal energy of ∼1 GJ was extracted from the plasma during a 360s pulse [72], and by the superconducting helical system, LHD, where 1.5 GJ was extracted during a pulse of 54 min [73, 74]. In the future, the steady-state high-power-density plasma will produce self-conditioned PFCs whose behaviour is likely to be different from preconditioned PFCs operated for short plasma pulses.

3.11.1. Advanced tokamaks. Experiments carried out on JT-60U, DIII-D, AUG and C-Mod have explored some aspects of a steady-state advanced tokamak. An example is shown in figure 11 where nearly stationary conditions were maintained for $\sim 16\tau_E$ or 2.7 plasma current redistribution times [75].

The advanced tokamak will also require feedback stabilization of various MHD modes such as NTMs and RWMs. Perhaps most important is the development of robust operating scenarios and plasma control systems that will avoid plasma disruptions in a tokamak. New medium to large scale superconducting tokamaks are being built to address steady-state advanced tokamak issues in non-burning plasmas as shown in figure 12. EAST at Hefei, China began operation in 2006 [76], KSTAR at Daejon, Korea in 2008 [77] and JT-60SA at Naka, Japan is planned for operation in ∼2014 [78].

The plasma aspect ratio is another area for optimization of an advanced tokamak. Various studies show enhanced plasma performance at lower aspect ratio; however, some engineering issues are more difficult. A number of low aspect ratio-tokamaks (START [79], Globus [80], NSTX [81], MAST [82]) have been built to better understand the physics and engineering capabilities at low aspect ratio, and to help determine the optimum aspect ratio to maximize overall fusion

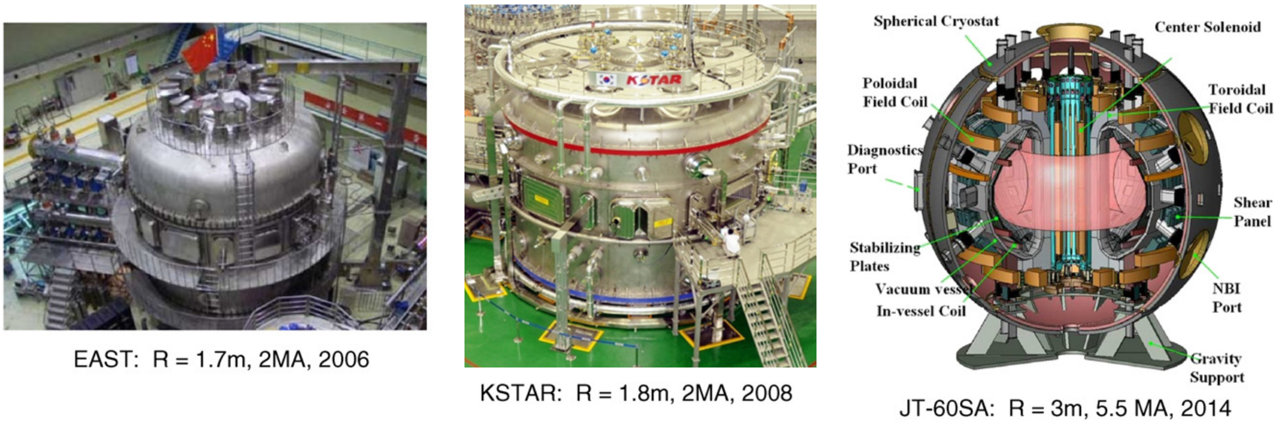

Figure 12. New Asian superconducting tokamaks (> 1 MA). Courtesy of the Chinese Academy of Science Institute for Plasma Physics (ASIPP), the Korean National Fusion Research Institute (NFRI) and the Japan Atomic Energy Agency: JT-60SA.

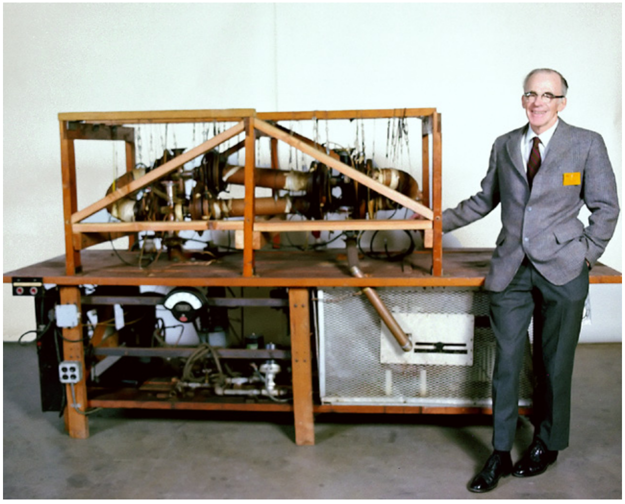

Figure 13. Lyman Spitzer with the Model A stellarator that was exhibited at Geneva 1958. Courtesy of the Princeton Plasma Physics Laboratory.

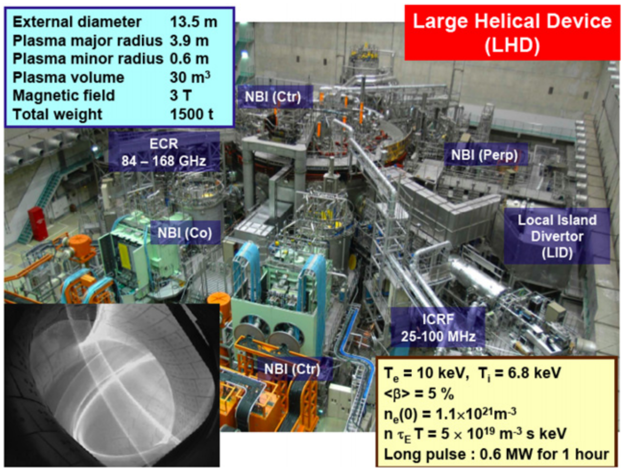

Figure 14. Large Helical Device (LHD). Courtesy of the National Institute for Fusion Science (NIFS), Japan.

system performance. The low aspect ratio approach may provide a new engineering approach with jointed copper toroidal field coils that would allow a different approach to internal maintenance of an activated device.

3.11.2. Stellarator and helical systems. The stellarator was proposed in 1951 with the first configuration being a simple figure 8 to cancel drifts due to toroidal curvature as shown in figure 13 [83]. Over time the design evolved with the magnetic transform being produced by helically wound coils or modular coils. Stellarators of the 1960s were plagued by Bohm diffusion and were left in the shadow of the tokamaks for most of the 1970s and 1980s. However, persistence by the stellarator groups in Germany and Japan gradually and steadily improved stellarator parameters as larger stellarators were built and operated [84]. The construction of the Large Helical Device (LHD) was undertaken in Japan during the 1990s while new medium-scale stellarators (Uragan 3, Modular-W-7AS) were being built. The LHD began operation in Japan in 1998 [85]. This was a remarkable achievement demonstrating that a large complex 3-D vacuum vessel and superconducting helical coil configuration could be built and operated reliably. This device with a plasma volume of 30 m^3 is comparable to the large tokamaks (figure 14). While the peak short pulse plasma performance of LHD is less than the large tokamaks, LHD has higher plasma performance for long pulses ~ 30 min in duration.

The German stellarator group has pursued a different path optimizing stellarators using modular coils. W7-AS was the first stellarator based on modular coils and was partially optimized. It demonstrated a commonality with tokamak physics such as access to the H-mode regime [86]. Success on W7-AS led to the design of W7-X, which is optimized to reduce bootstrap currents and fast particle loss while maintaining good magnetic surfaces, MHD stability and feasible coils. W7-X, a large modular superconducting coil stellarator (figure 15) with a plasma volume comparable to the large tokamaks, is under construction and is expected to begin operation in 2014 [87].

3.12. Fusion technology

The advancement in fusion technology has been equally spectacular during the past 50 years. As an example, reliable large scale (~1.6 GJ) copper conductor magnets at 5 T have

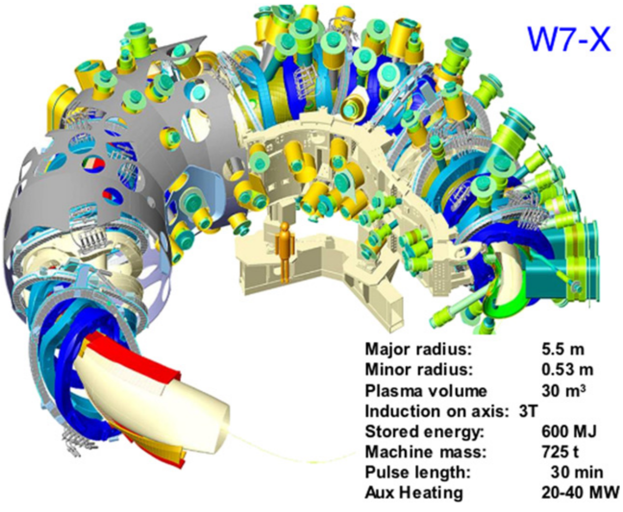

Figure 15. Optimized Stellarator (W7-X) [87]. Courtesy of the Max Planck Institute for Plasma Physics, Greifswald, Germany.

been operational for many years. There is a solid experience base in superconducting (SC) magnet technology such as:

- SC coils for large, complex magnetic mirror experiments in the early 1970s through the mid-1980s,
- First tokamak SC toroidal field coil tokamak T-7 was completed in 1979,
- Large coil project with five different types of SC coils was completed in the late 1980s,
- Large tokamak SC experiments T-15 and Tore Supra were completed in 1988,
- EAST, KSTAR, advanced tokamaks with SC coils began operation in 2006–2008,
- ITER CS Coil Demo (640 MJ) confirmed operation with tokamak dB/dt of ∼1T/s in 2000.
- ITER will demonstrate reactor-scale SC magnets ($\leqslant$ 43 GJ) at $B = 5.3$ T.

There are additional opportunities for progress in this area due to continuing development of higher B and/or higher T superconducting materials and more efficient coil designs.

Significant progress has also been made in the other technologies required for fusion such as large scale vacuum systems, high-power-density plasma facing components, multi-MW auxiliary heating systems, plasma fuelling systems, tritium systems and remote handling of activated components. The safety of large complex fusion systems has been demonstrated with sustained DT operation. The construction and operation of ITER will provide much valuable information on fusion technologies.

3.13. An International Team is forged to take the next step in magnetic fusion towards a new energy source as global concerns about energy and environment increase

As encouraging technical results came in during the 1980s, each of the major international parties in fusion began to plan for the next step that would achieve sustained burning plasma conditions. There were efforts in the 1980s to design single party burning plasma experiments (IT-IGNITOR, FRG-Zephyr, US-CIT) as well as test reactors (US-FED, EU-NET, JA-FER). The idea of ITER, a global collaborative activity, was first discussed by Gorbachev and Reagan at the 1985 Geneva Summit (figure 16) with an agreement at the 1987 Reykjavik Summit for the Soviet Union, EU, JA and US to pursue a joint activity to design a large international fusion facility which later became the International Thermonuclear Experimental Reactor (ITER). Conceptual Design Activity (CDA) of ITER started in 1988 under the auspices of the IAEA with an international team sited in Vienna. An Engineering Design Activity was begun in 1992 and extended until 2001 with a Transitional Activity until 2005. A construction site for ITER in Cadarache, France was agreed on in June 2005, and the ITER Implementing Agreement was signed by EU, JA, RF, ROK, CN, IN and the US on 21 November 2006 in Paris, France. ITER is now officially known by its Latin name 'The Way,' and the project is now well underway with construction scheduled to be completed by 2018 with first DT operation beginning ∼2024 [88]. The overall programmatic goal of ITER is to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes. ITER (figure 17) is expected to achieve sustained burning plasma conditions with $Q \approx 10$ at power levels of ∼500 MW for ∼400 s yielding ∼200 GJ per pulse. In longer pulse operation, ITER is expected to achieve $Q \approx 5$ at power levels of ∼350 MW for ∼2500 s yielding ∼900 GJ per pulse.

4. Inertial confinement fusion

4.1. Inertial confinement fusion—the early days

Radiation implosion of DT 'capsules' to produce fusion energy was demonstrated in the early 1950s in the Greenhouse George Cylinder and MIKE nuclear weapons tests. The challenge for inertially confined fusion, as an energy source, is how to reduce the yield of thermonuclear explosions to values small enough for the energy to be captured and converted into electricity. The invention of the laser in 1960 offered the possibility of a programmable repetitive radiation driver for imploding micro-targets, and several Laboratories in the US, USSR, UK and FR initiated high power laser development programmes. In addition, research on intense particle beam and Z-pinch drivers was carried out in the US and USSR.

4.2. Inertial fusion target design

Idealized calculations in the late 1960s suggested that a 1 kJ radiation pulse would be needed to achieve breakeven using direct drive implosion of micro-targets. A more thorough assessment by Nuckolls *et al* using computer modelling of laser driven implosions [89] indicated higher energy levels would be needed. Laser driven implosion experiments at LLNL and elsewhere from mid-1970s to mid-1980s (Nova), revealed the importance of plasma instabilities and driver uniformity [90]. This required shorter wavelength (351 nm) lasers with laser driver energy in the megajoules range. Classified Centurion-Halite nuclear tests in ∼1986 are reported to have validated the modelling of radiation driven implosions. The Omega Project at Rochester is a 60 beam 40 kJ laser system for testing the physics of direct drive implosions, and has achieved fusion gain of ∼1% using direct drive of a DT capsule in 1996 [91]. The

Figure 16. President Reagan and Prime Minister Gorbachev discussions at the 1985 Geneva Summit. Academician Velihkov, on the right, was instrumental in getting fusion on their agenda.

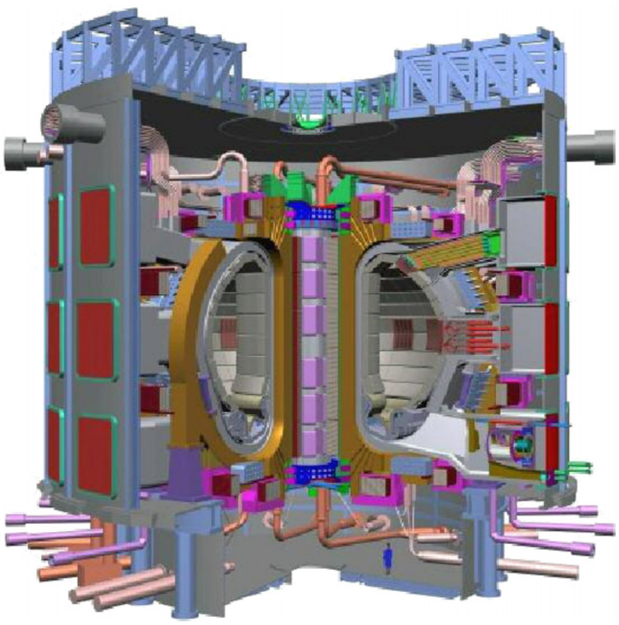

Figure 17. ITER—the way to fusion energy. Courtesy of the ITER International Organization.

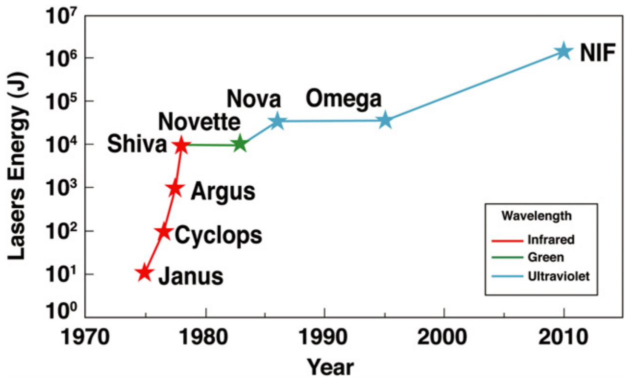

Figure 18. Inertial fusion high power laser development in the United States. Courtesy of Lawrence Livermore National Laboratory (LLNL).

technical issues and status of ICF is described in two National Academy Reviews 1988 [92] and 1992 [93].

Some aspects of ICF target design were declassified in 1994. Analysis of various target designs indicate that laser energies ∼1–2 MJ are needed to achieve a fusion gain of ∼10 with either direct or indirect drive using traditional hot spot ignition. Fast ignition [94], where an ultra-high power pico-second laser acts as a spark plug to ignite a moderately compressed target, is estimated to require laser energies of ∼100 kJ. This concept is in the early stages of development and is being tested in Japan in the FIREX experiments [95].

The analysis of inertial fusion target design has benefited greatly from advanced computing made possible by the extraordinary increase in the capability of ultra-large high-speed computing facilities.

4.3. High power radiation driver development

High power, programmable pulsed glass lasers have made enormous progress with laser energy increasing from 1 J per pulse in 1970 to 100 kJ in 1990, and it was conceivable to build a ∼2 MJ laser system with the properties required to achieve ignition. (figure 18). This increase in laser technology has provided the opportunity to carry out a sequence of implosion experiments with increasing laser driver power and uniformity on Shiva, Nova, Omega, now Omega Upgrade. Additional driver development continues on KrF lasers and high power x-rays produced by Z beams.

4.4. National Ignition Facility (NIF)

During the early 1990s, amidst discussions to eliminate nuclear weapon testing, it was proposed to initiate a Stockpile Stewardship programme that included the testing of laser driven micro-targets. The NIF was initiated in ∼1994 to accomplish a National Security Mission and construction was started in 1997. The NIF construction project (figure 19) was completed in 2009, and the 192 beam laser system has been fired at 1.1 MJ per pulse during commissioning. The ignition campaign is scheduled to begin in 2010 with significant results expected within a few years. NIF has a goal of $Q \approx 10$–20 with fusion energy yields per pulse of 20–40 MJ [96].

In addition to NIF, a second MJ class laser, Laser Mégajoule (LMJ), is being built in France with operation planned for 2012. The system consists of 240 beams with a total energy of 1.8 MJ, and will be used to validate and refine fusion yield calculations relevant to nuclear weapons [97].

Figure 19. One of two NIF laser bays. System has been test fired at 1.1 MJ per pulse. Courtesy of Lawrence Livermore National Laboratory (LLNL).

5. Overall highlights of 50 years of fusion research

Research in magnetic confinement fusion energy over the past 50 years has made tremendous progress with the Lawson parameter ($n\tau_E T$) in magnetic fusion devices increasing by 10 million to within a factor of 10 of that needed for large scale fusion power production. The next major step in magnetic confinement fusion is to be taken by ITER with the production of ∼500 MW of fusion power for ∼400 s. Similarly, inertial confinement fusion has made impressive progress with the increase in laser driver power by 1 million, and the completion of a major facility, NIF, aimed to produce ignition of small DT pellets and 20–40 MJ of energy per pulse. In addition to specific achievements, the overall highlights can be summarized as:

- A strong scientific basis has been established for proceeding to the next stage, fusion energy production, in the development of magnetic and inertial fusion.
- Diagnostics and plasma technology (auxiliary plasma heating, current drive, pellet injection and plasma facing components) have made enormous progress and have facilitated a deeper understanding of the physics, thereby enabling progress.
- There are several promising paths to both magnetic and inertial fusion energy and, each is working on optimization and sustainment (or increased repetition rate).
- Temperatures (>100 million °C) needed for fusion have been achieved in many facilities.
- Confinement needed for fusion is being approached in the laboratory—one step away.
- Complex fusion systems have been operated reliably at large scale.
- Fusion systems using fusion fuel (DT) have operated safely.
- The international fusion programme is on the threshold of energy producing plasmas in both magnetic and inertial fusion.

6. The next 50 years of fusion research

The need for a non-CO_2 emitting power source with a secure fuel supply is much stronger today than it was 50 years ago. The stage is now set for the international fusion programme to begin planning for the step to a fusion demonstration facility (Demo). The major research issues which need to be addressed both in physics and technology in the coming decades include: validated predictive capability for burning plasmas, plasma facing components capable of high-power-density long-pulse operation, breeding blanket design and development, and material technology for high neutron fluence. Fusion could move much faster towards a fusion Demo if the resources required to develop a new source of energy were applied. The time is right for a major increase in effort for fusion energy development so that the promise of fusion energy can be realized in time to address the global issues of secure energy supplies and climate change within the next 50 years.

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