CREATING A STAR – THE GLOBAL ITER PARTNERSHIP

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Abstract

ITER is an unprecedented global partnership to demonstrate the scientific and technological feasibility of generating, carbon-free, and virtually unlimited energy through the fusion of hydrogen isotopes. Now under construction in southern France, the ITER fusion reactor is designed to achieve and sustain self-heated, or "burning," plasma that can produce ten times more power than required for plasma heating. This overview provides an introduction to fusion, summarizes the history of the ITER project, describes key subsystems and elements of the ITER reactor, and includes objectives and goals of the ITER research plan.

Introduction

Bolts of lightning arcing down across the horizon of the African savanna were as mysterious to early humans as the Sun during the day and the veil of stars across the blackness of the night sky. The fact that these phenomena are closely related as an energetic state of matter— now known as plasma— would take many more millennia before being scientifically understood. Nevertheless, human recognition that a bolt of lightning striking a lone tree produced a useful fire signaled the dawn of civilization's endless pursuit of energy resources.

Wood served as an energetic fuel of human society for at least 10 millennia and then, in the late nineteenth century, human understanding of the nature of fuels began changing rapidly. Coal, the petrified remains of biodegraded organic matter, fed the powerhouses that drove a global industrial revolution. By the early twentieth century, petroleum derivatives began joining coal as a more transportable and efficient fuel, mobilizing the accouterments of war and empowering the nations that mastered oil and gas extraction as world leaders.

In 1905, Albert Einstein revolutionized the world of physics with publication of the mass-energy equivalence principle, $E=mc^2$, and by the early 1950s the first nuclear power plants began making their debut. Nuclear energy based on uranium fission—generating low-cost electricity (i.e., 5–10 cents per kilowatt-hour) over long plant life-cycles (i.e., 40–60 years) at unprecedented power levels (i.e., 2–3 gigawatts per plant) —now represents a major carbon-free energy source in the modern world. However, the evolution of energetic fuels does not necessarily have to end at the close of the 20th century with the industrialization of uranium fission. Hydrogen fusion holds the potential to be the next major development in nuclear energy.

Uranium is a “heavy element” (atomic number: 92) that releases energy when the nucleus is split to form smaller nuclei. The energy is released primarily in the form of neutrons, gamma rays and energetic fragments. In contrast, hydrogen is a “light element” (atomic number: 1), that releases energy when the nuclei of hydrogen isotopes are fused together to form helium (atomic number: 2). A technological transition from heavy-element fission to light-element fusion would lead to not only far fewer radioactive isotope byproducts, but also increased global availability of natural fuel. Inputs would come in the form of “heavy” hydrogen− termed deuterium because it has one additional neutron−which is found in seawater, and lithium, which can be extracted from the Earth’s crust and oceans. These two elements are the feedstock for a fusion energy fuel cycle.

Achieving controlled hydrogen fusion in order to generate carbon-free electricity is one of the “grand challenges for engineering in the twenty-first century” according the U.S. National Academy of Engineering.[1] As with many historic advances, the emergence of several disruptive technologies can often converge and precede a practical solution. In the case of fusion, the close of the last century brought with it the availability of high performance computing that enabled great strides forward in the modeling of complex large-scale flows, turbulent small-scale flows and energetic particle dynamics, all of which bear acutely on fusion processes. In combination with highly sensitive diagnostic instruments, theoretical models can be formulated, validated and integrated far more effectively than in the past. Lastly, high-efficiency, low-temperature superconductors, such as niobium-tin (Nb_3Sn), have replaced copper in the electromagnetic coils used for confinement and shaping of fusion plasmas. As such new technologies are applied to the engineering of twenty-first century reactor configurations, fusion need no longer be “thirty years away”.

In the sociopolitical context, hydrogen fusion offers the benefits of nuclear energy without the constraints associated with fission. The primary by-product is inert helium, while the tritium that is also produced by neutron bombardment of lithium remains contained in a closed fuel cycle. Therefore, fusion can be a very safe, carbon-free source of electricity. Fusion also requires high temperatures; any disturbance tends to degrade confinement, cool the plasma and reduce the reaction rate, so there is no risk of a “runaway reactor” or “meltdown.” Nor is there any highly radioactive fuel that keeps emitting heat after shutdown. Thus, fusion reactors would be inherently safer to operate than fission reactors. While the highly energetic flux of 14.1 MeV fast neutrons will activate plant structure and materials, the radioactive lifetime can be engineered through low-activation materials selection, thus reducing radiotoxic lifetimes to readily managed durations (e.g., a century as opposed to millennia).[2] Finally, hydrogen is a plentiful element making up an estimated 90% of all atoms and three quarters of the mass of the visible universe; it is the third most abundant element on the Earth’s surface (behind oxygen and silicon). A hydrogen fusion-based energy source would be globally accessible without the restrictions of geopolitically-controlled natural resources.

The ITER Partnership

ITER— Latin for “the way” or “the journey” and originally an acronym for the International Thermonuclear Experimental Reactor— can trace its origins to the wake of the 1970s global energy crisis. In October 1973, the Organization of Petroleum Exporting Countries announced a trade embargo on oil that jolted the world at large into sudden recognition of energy as a fundamental factor of national production and security. This was followed by a second “oil shock” in 1979 associated with decreased oil production as a consequence of the Iranian

Revolution. Prior to that decade, energy resources had rarely been considered as a constraining factor on national economies.[3]

From INTOR to ITER[4]

In 1978, the International Atomic Energy Agency (IAEA) invited member governments involved in fusion research to consider the timeline for fusion energy development and potential advantages of international cooperation on fusion science and engineering. In response, Evgeny Velikhov of the former Soviet Union proposed an international cooperative project to design, construct and operate an experimental reactor based on the tokamak concept. The International Fusion Research Council, an advisory body to the IAEA, recommended a Specialist Committee be formed, including representatives from the U.S., Europe, Russia and Japan, to evaluate the prospects. At an organizational meeting of the committee in Vienna, Austria, the plan for a series of workshops was agreed upon and the effort was christened the International Group Working on a Tokamak Reactor (INTOR).

During the decade from November 1978 to March 1988, many sessions of the INTOR Workshops were conducted, with contributions from as many as 150 individuals from the participating parties. By 1985, however, plasma physicists and fusion engineers had become concerned that the INTOR Workshops might amount to little more than a "paper study." This changed during the 1985 Geneva Summit Meeting, when USSR Head-of-State Mikhail Gorbachev proposed to U.S. President Ronald Reagan that a joint effort be undertaken to advance the INTOR concept into a final design, followed by construction and operations phases. Within a year, the ITER Project was conceived and there was an intergovernmental agreement to proceed. Meanwhile, the INTOR Workshops continued and culminated in publication of a final report in 1988.[5]

The ITER Project officially commenced in 1988 and was initially hosted at the Max Planck Institute for Plasma Physics near Munich, Germany for the early Conceptual Design Activity (1988–91). An ITER Engineering Design Activity (EDA) period followed (1992–1998) where the U.S., Europe, Russia and Japan formalized their national participation, and many of the scientists who had participated in the INTOR Workshops continued as members of the new ITER global team.

The U.S. did not extend participation beyond the 1998 end of the EDA. Subsequently, a series of US-sponsored workshops over 2001–02 culminated in a 2-week study session in Snowmass, Colorado in Summer 2002 for the purpose of seeking expert assessments of the scientific and technological readiness for studying burning plasmas and of three approaches to that study, including the ITER version that emerged from the EDA. The ITER approach was selected by the U.S. government based on collective judgment of participating experts, including the Fusion Energy Sciences Advisory Committee to the U.S. Department of Energy, as well as a study completed by the U.S. National Academies/National Research Council.[6] Consequently, in 2003 the U.S. rejoined ITER negotiations. China and South Korea also joined in 2003, and India completed the current partnership by joining in 2005.

Provisions of the Joint Implementation Agreement (JIA)[7]

Under the auspices of the IAEA, a joint implementation agreement was formally signed among EURATOM (the European Atomic Energy Community), the Republic of India, Japan, the People's Republic of China, the Republic of Korea, the Russian Federation and the United States of America in November 2006. According to the provisions, an International Fusion Energy Organization (IO) was established for the purpose of demonstrating *"the scientific and* *technological feasibility of fusion energy for peaceful purposes, an essential feature of which would be achieving sustained fusion power generation."* Following years of study and negotiation among the parties, the JIA also resolved that the IO and tokamak research and development (R&D) laboratory be situated at St. Paul-lez-Durance in the south of France, adjacent to the Commissariat à l'énergie atomique (CEA), the French Atomic Energy Commission.

The IO is governed by a Council composed of up to four representatives from each of the seven partners. Each government partner has also designated a "Domestic Agency" (DA), in order to provide contributions to the IO through an established legal entity. Contributions are defined in two forms: (a) in-kind goods and services, consisting of specific components, equipment, materials, and R&D, as assigned to each partner for delivery in accordance with IO technical specifications, and (b) in-cash contributions to cover the IO annual operating expense and components to be supplied directly by the IO. Annexes to the JIA provide further detail on specific technologies to be contributed, etc.

The JIA was approved with an initial duration of thirty-five years and included a provision for a Special Committee to be formed eight years prior to expiration, in order to advise on extension in light of progress achieved.

Project Life Cycle

Based on agreement among the international parties, the project life cycle is divided into three distinct phases:

Phase 1: Design and Construction
Phase 2: Operations
Phase 3: Decommissioning

Design and Construction Phase

The point-of-departure for this first phase was the configuration resulting from the 1992–2001 EDA period. The final design and construction phase includes:

- Establishment and operation of the IO as an institutional entity;
- Resolution of details on functional and design specifications;
- System engineering and analysis;
- Completion of procurement arrangements between the IO and DAs;
- Site survey, regulatory approvals, excavation and development;
- Building and road planning, civil engineering and construction;
- Prototype development, testing and evaluation for plant subsystems and elements;
- Manufacturing and delivery of plant subsystems and elements to construction site;
- Subsystems and elements integration and assembly of plant.

Operations Phase

The primary mission of ITER is experimental research operations; details for the operations phase are therefore discussed in a later section on the Research Plan. The operations phase commences following completion of construction and gradually progresses from commissioning of systems to full safety-qualified nuclear R&D operations as the nuclei fuel mix changes from hydrogen and helium to deuterium and tritium. During the operations phase, the IO is responsible for establishing a fund to provide for decommissioning of ITER facilities.

Decommissioning Phase

The decommissioning fund and facilities are to be transferred to France, as the host state, following completion of research operations. France will remain bound to Article 20 of the JIA, which constrains any further uses of ITER facilities and equipment to peaceful purposes, and ensures conformity with principles of non-proliferation.

Fusion R&D Laboratory Complex, St. Paul-lez-Durance, France

The ITER complex resides on a 180-hectare tract of land, including an elevated 42-hectare platform standing 315 meters above sea level, located approximately 60 kilometers northeast of the Port of Marseille in St. Paul-lez-Durance, France. The site was selected for its proximity to the existing Cadarache CEA facility, which has been a French scientific research center for nuclear energy since 1959.

Two years of site preparations were completed in June 2009, and by May 2010 an architect engineering contract was awarded for design and construction of buildings, infrastructure and power supplies to the European consortium ENGAGE (Assystem, France; Atkins, UK; Empresados Agrupados, Spain; and Iosis, France). Site development commenced with the building of roads, excavation for the first foundations of the tokamak complex, and construction of a power substation.

The ITER Site Master Plan includes thirty-nine buildings and a wide range of supporting infrastructure necessary to support research, operations and maintenance of the ITER tokamak facility.

Status of the ITER Project (2015)

On November 9, 2012 the French Ministry of Ecology, Sustainable Development and Energy issued a decree authorizing construction of the ITER facility; this document granted the IO a license to construct a nuclear facility. During the period 2013–15, the worksite was transformed from a sparse construction platform into a busy hub of industrial activity with foundations being poured, steel exoskeletons erected, and sky cranes rising above the future tokamak pit. The Poloidal Field Coils fabrication building and a Cryostat Workshop, where the approximately 30 x 30 meter stainless steel structure will be assembled by India, were completed, while the Assembly Hall, Tokamak and Diagnostics Buildings are rising up from the ground. Figure 1 provides an aerial view of the ITER construction site taken in August 2015 and Figure 2 provides an artistic rendition of the ITER complex upon completion.

Figure 1

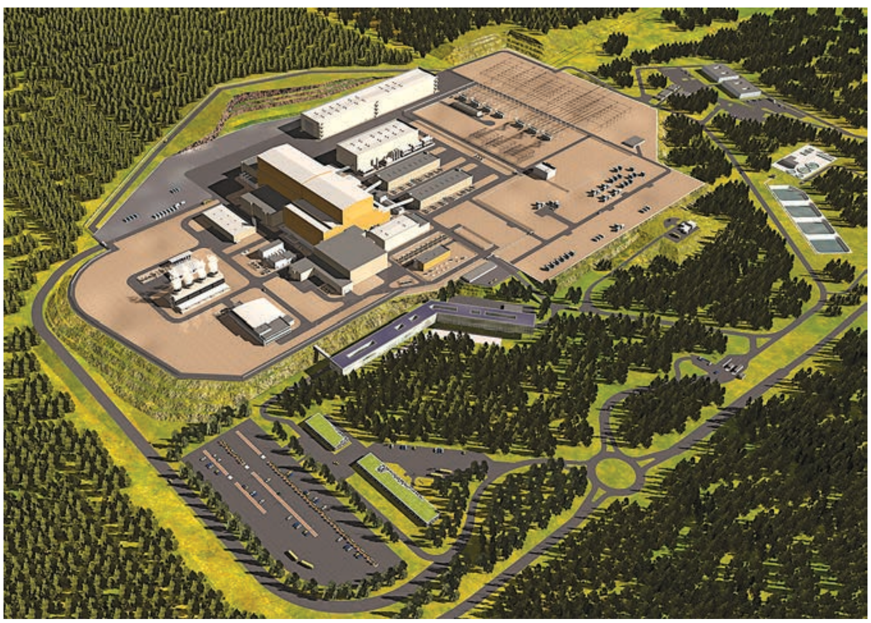

Figure 2

In parallel with the ramp-up in site construction activity, DAs from around the world began fabrication of early-lead components. Toroidal field (TF) cable-in-conduit superconductor entered production in China, Europe, Japan, Korea, Russia and the U.S., and shipments to coil winding facilities began. Figure 3 shows a winding of TF conductor into a D-shaped coil at the European winding facility in La Spezia, Italy. Over 80,000 kilometers of superconducting strand (Nb_3Sn) will eventually be used to wind the eighteen toroidal field coils that form the electromagnetic field surrounding the torus and confining the plasma. In the U.S., the central solenoid coil fabrication facility was completed and the first of seven modules entered production. Figure 4 is a photo of the new General Atomics Magnet Technologies Center where the 18-meter high, 4-meter diameter, 1,000 ton superconducting magnet will be produced. The central solenoid is figuratively the “heartbeat of ITER,” because the pulse it generates drives the current in the plasma. In shipping, the first U.S. procured and fabricated “highly exceptional loads” (over-sized loads) also began arriving at the ITER site, including 87-ton high-voltage transformers required by the steady-state electrical network and 61,000-gallon drain tanks for the tokamak cooling water system.

Figure 3

Figure 4

By 2015, the ITER Project was on the cusp of transition from design to construction with preparations well underway to begin assembling the tokamak. Construction process documents and work packages included stamps of approval, and a global workforce was engaged across the partnership. The IO was in the process of transforming its focus from design to construction management as the assembly phase approached.

Tokamak Design

The term "tokamak" originated at the Kurchatov Institute in Moscow as an acronym for "toroidal chamber with magnetic coils." It was based on a theory of electromagnetic traps proposed by Oleg Lavrentiev while attending the Kharkiv Theoretical Physics School in the Ukraine. Soviet physicists Igor Tamm and Andrei Zhakarov, under the direction of Lev Artsimovich, led early advancement of theory to practice.[8]

An early challenge to sustaining conditions for fusion research was reducing the heat loss by confining the hot (i.e., 150−300 million ˚C) plasma for sufficient time at necessary density. The use of an electromagnetic field shaped as a torus proved to be effective in overcoming this constraint. As a result, the tokamak evolved to become the most practical and well-understood experimental device for fusion R&D. Approximately two hundred tokamaks have been constructed around the world with more than thirty remaining in operation today.[9]

In order to sustain the generation of power and deliver useful energy, a practical fusion process must yield more energy than required to start and maintain the reaction. Attaining the required "Q-value" (ratio of fusion power produced to power required) is therefore dependent on reducing the external heating power by achieving a self-heated, or "burning," plasma state, where the energy released from fusing hydrogen nuclei is sufficient to dominate the heating needed to sustain the reaction. Although power production through controlled fusion reactions was experimentally proven for brief instants during the 1990s[10], generation of fusion power by sustaining burning plasma remains to be demonstrated.

For these reasons, achieving and maintaining substantially self-heated, or "burning," fusion plasma is the mission objective of ITER. Realization of the objective would represent an historic turning point in fusion energy R&D. This most salient aspect was clearly articulated by the U.S. National Research Council in 2004: "*It is widely agreed in the plasma physics community that the next large-scale step in the effort to produce fusion energy is to create a burning plasma – one in which alpha particles from the fusion reactions provide the dominant heating of the plasma necessary to sustain the fusion reaction.*"[11]

ITER is designed to be the first tokamak to produce burning plasma. Since achieving that state demands a large plasma volume, ITER will also be the largest tokamak yet. The rationale for the greatly enlarged scale (i.e., more than eight times the plasma volume of prior tokamaks) was based on the advantage gained by reducing the ratio of toroidal surface area to volume in order to minimize heat loss. An earlier ITER tokamak design, of yet larger toroidal geometry, was reduced in scope due to a trade between cost and scale. The current ITER design is most notable for its flexibility in supporting a wide range of experimental operations. The arrangement of toroidal, poloidal (poloidal field coils and central solenoid), correction, and in-vessel coils permits unprecedented control over plasma current and current profile, plasma shape, and plasma position while cancelling error fields from external sources and ameliorating edge localized modes in the plasma. Multiple heating systems—electron cyclotron, ion cyclotron and neutral beam—provide controlled heat deposition and non-inductive current drive in the plasma. An exhaustive array of diagnostic instruments and associated actuators will provide active control techniques with which to experimentally influence the plasma shape, thermal and density gradients, current pulse and other key parameters. The integrated system represents a powerful tool for fusion R&D. Major physical characteristics and functional performance parameters are summarized in Figure 5.

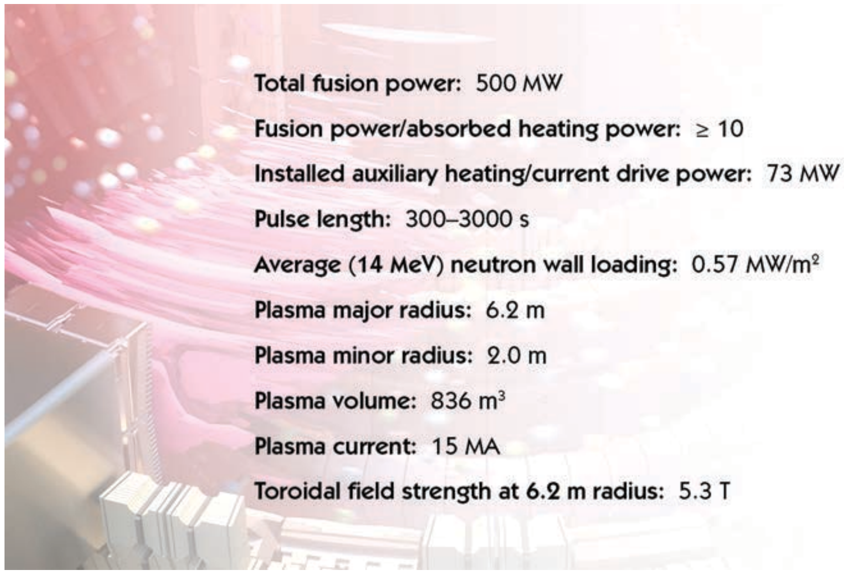

Figure 5

A tokamak configuration will not *a priori* become the optimum approach for generating practical fusion energy. Stellarators, spherical tori, or a variety of exotic architectures could eventually prove more effective, or economical. However, the tokamak was selected for ITER because it is the most mature and best understood configuration to support an extremely wide field of inquiry for probing, manipulating, and finally understanding the complex dynamics of burning hydrogen plasma. The advances in both theoretical and practical understanding that result from ITER experimental research will contribute meaningfully to any future magnetic confinement fusion technology.

Major Elements and Distributed Systems

While the complexity and scale of ITER is unprecedented, the subsystems and elements that comprise the integrated system all fall within the realm of either proven technology or incremental advances that can be attained through focused engineering R&D. In this respect, the design and construction of a fully operable ITER tokamak is most appropriately viewed as an engineering challenge, whereas the achievement of quasi-steady-state burning plasma with a net power gain remains the scientific challenge. The following principal subsystems and elements for meeting these challenges will be summarized in the sections that follow.

a. Power supply
b. Superconducting magnets
c. Vacuum vessel and internal elements
d. Cryostat and thermal shield
e. Fueling pellet injection
f. Plasma heating
g. Tokamak cooling
h. Exhaust processing

i. Biological shield
j. Controls and Instrumentation

Figure 6 provides a computer-aided-design (CAD) drawing of the integrated ITER tokamak.

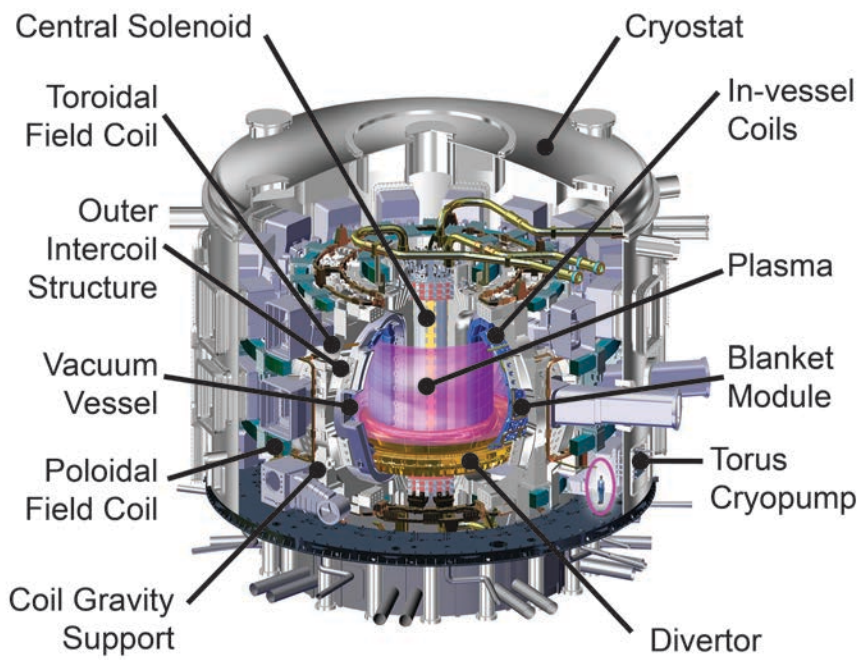

Figure 6

Power Supply

The ITER facility complex is linked by a 1 kilometer extension to the 400 kV Prionnet Substation operated by Electricity of France (EDF)—the world's largest producer of electricity, which supplies approximately 20% of the electrical needs of the European Union and relies on nuclear fission for over 80% of that supply.[12]

The ITER Steady-State Electrical Network (SSEN) is an AC distribution system, consisting of standard commercial grade AC power system components servicing all conventional building loads with an approximately 120 MW power capacity. The SSEN also includes two diesel generators for emergency backup. Since the SSEN provides site power during the construction phase, it is the first major ITER subsystem to be installed and commissioned.

Pulsed Load[13]
Tokamaks operate as pulsed electrical systems with very high cyclical loads. Pulse lengths vary from 300 seconds up to an hour, with a nominal 1800 second repetition period. The total load includes power required for:

- Ramping up and sustaining the pulsed magnetic field and plasma current
- Position and shape control;
- Heating and current drive, and;
- Resistive loss compensation.

The loads nominally occur in four phases:

- Pre-magnetization;
- Plasma initiation;
- Plasma steady-state, and
- De-magnetization.

During a nominal plasma pulse, the AC/DC converters generate extremes in reactive power variation where current and voltage are no longer in phase. The short and steep pulses of active and reactive power negatively affect the power quality of the electric network. While the Prionnet Substation can provide up to 500 MW of active power, it has a limited capacity of 200 Mvar for reactive power. As a result, the ITER plant must provide additional reactive power compensation (RPC). The ITER Pulsed Power Electrical Network (PPEN) is designed to address this, as well as provide harmonic filtering (HF). It will have a capacity of 500 MW and be able to compensate 750 Mvar of reactive power. This is accomplished with three RPC and HF units controlling voltage at the 66 kV bus bar level, each having one thyristor controlled reactor (- 250 Mvar) and six harmonic filters (+ 250 Mvar in total). Figure 7 depicts the Pulsed AC Distribution Network and Figure 8 provides a simplified single line diagram of RPC and HF.

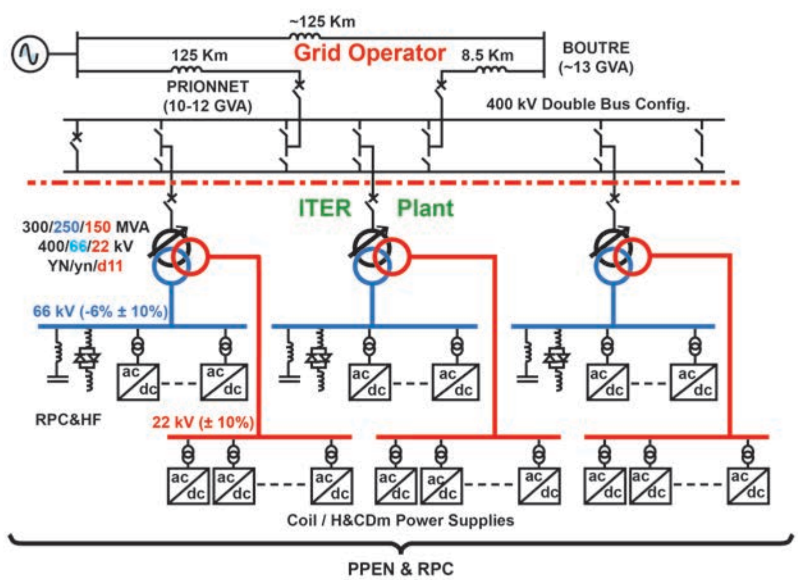

Figure 7

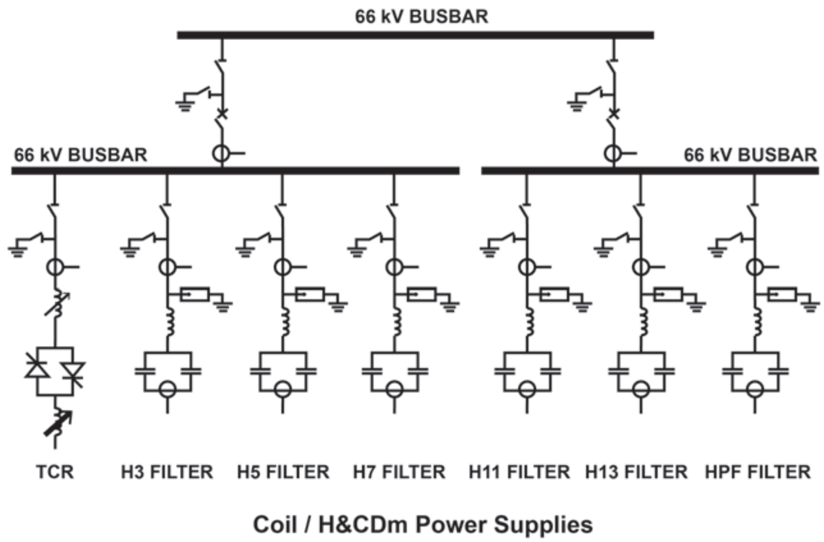

Figure 8

The PPEN is also designed to continue normal operations during major transient disturbances. Magnet quench (the transition of the magnet windings from superconducting to normal conducting) represents the worst-case event, since it will lead to large thermal and electromagnetic stresses on the system. In such a case, the load drops out in 50 milliseconds and the PPEN is designed to withstand this transient fifty times over its life cycle.

Superconducting Magnets

The ITER electromagnetic field is established by a series of toroidal field (TF) coils and further shaped by rings of poloidal field (PF) coils with error fields managed by a series of smaller correction coils (CC). A single large central solenoid (CS) coil is employed to induce and regulate current in the plasma. Figure 9 provides an elevation through a typical cross-section of the torus indicating the locations of TF, PF, CC and CS coils. All magnet structures are designed for 30,000 tokamak pulses at full field and a15 mega-amp nominal plasma current.

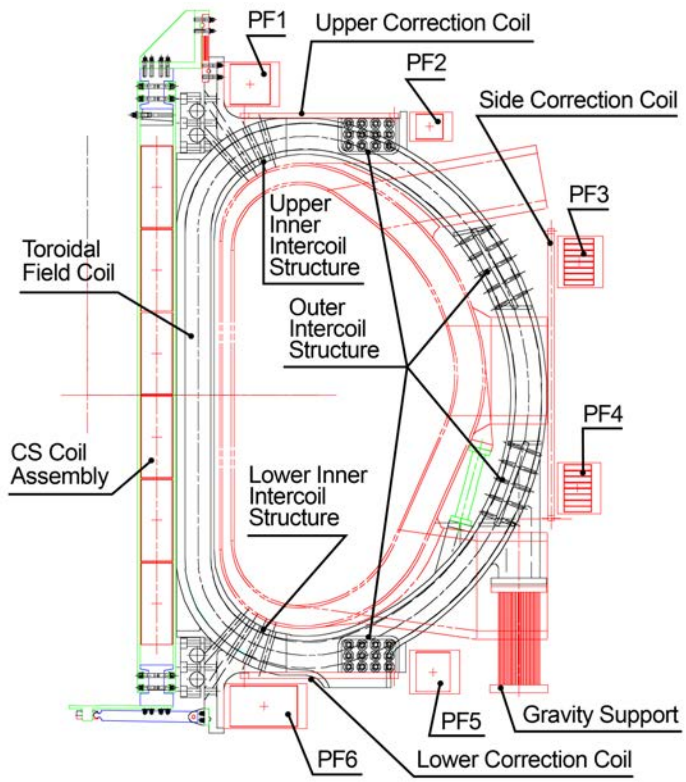

Figure 9

Superconducting magnets lose electrical resistance when cooled down to very low temperatures, thus allowing greater electrical efficiency during the high power operations required in tokamaks. This results in an attractive ratio of power consumption to cost for the long plasma pulses. All of the large ITER magnets are superconducting, and cooling is achieved by circulating supercritical helium in the range of 4 Kelvin (-269 ˚C) through the cores of the cable-in-conduit conductors.

Superconducting cable-in-conduit is formed by first twisting individual superconducting and copper wire strands together into a bundle. Several bundles are then interwoven together around a stainless steel tube that serves as the coolant channel. Finally, this subassembly is inserted inside a stainless steel jacket and compacted. The resulting cable-in-conduit is spooled for shipping to another location where it is de-spooled and run through winding and forming machines that produce magnets in the needed shapes and sizes. Figure 10 indicates typical circular (toroidal field conductor) and rectangular (central solenoid conductor) cross sections for superconducting cable-in-conduit.

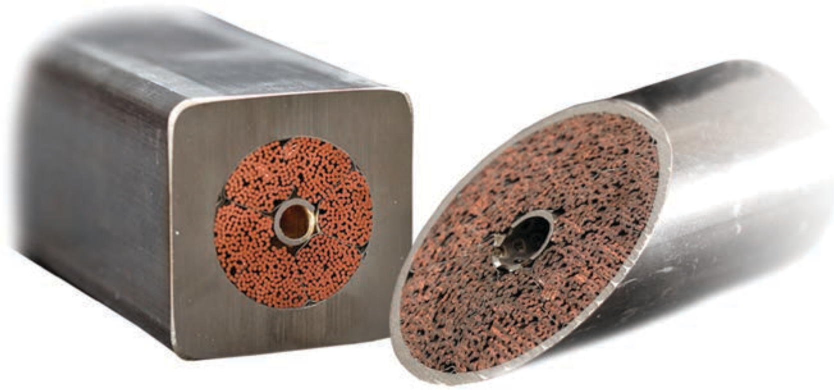

Figure 10

Toroidal Field Coils

The primary ITER electromagnetic field for confining the plasma is formed by eighteen D-shaped TF coils having field strength of 11.8 Tesla and total magnetic energy of 41 Gigajoules. The TF coil current is 68 kiloamps with an 11 second discharge time constant, and each coil has a centering force of 403 meganewtons (enough force to lift a 40,000 tonne object). The coils are wound with 115 km of superconducting niobium-tin (Nb_3Sn) conductor and placed in a stainless steel case. Each coil weighs over 350 tons including the structural coil case.

In order to maintain high operational reliability, the TF conductor is embedded in grooved radial plates mounted inside the structural coil case. Since the coil case experiences cyclical loading from the out-of-plane forces generated by interaction of both the TF coil current and poloidal field coil current, a combination of shear keys and pre-compression rings are used to provide a centripetal preload at assembly. Figure 11 illustrates the TF coil location in the tokamak and highlights a single coil-in-case.

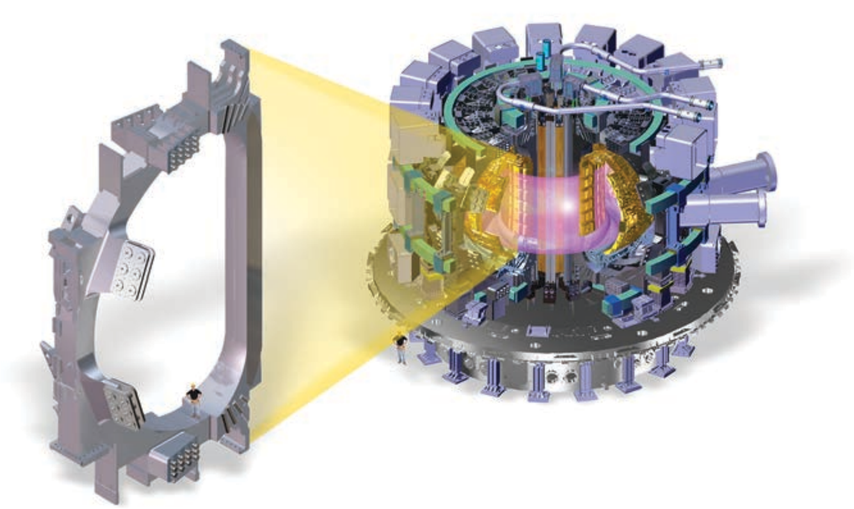

Figure 11

The radial plate design was successfully demonstrated during the TF Model Coil Project conducted during the 1990s EDA period. Solutions were also confirmed for issues involving fatigue life of the conductor jacket and insulation reliability. As a result, the engineering performance and industrial manufacturing feasibility of the conductor and magnets are well-established, and the final production TF coils are now in the fabrication phase for ITER.

Poloidal Field Coils

The six circular and horizontally-positioned PF coils were optimized to provide additional magnetic field control to shape the plasma vertically and radially, and maintain plasma equilibrium. The coils are wound from superconducting niobium-titanium (NbTi) alloy in square jackets, and normally operate at 45 kilo-amps with a 14 second discharge time constant. These coils range from 8 to 24 meters in diameter and are illustrated in Figure 12 along with the three sets of correction coils discussed below.

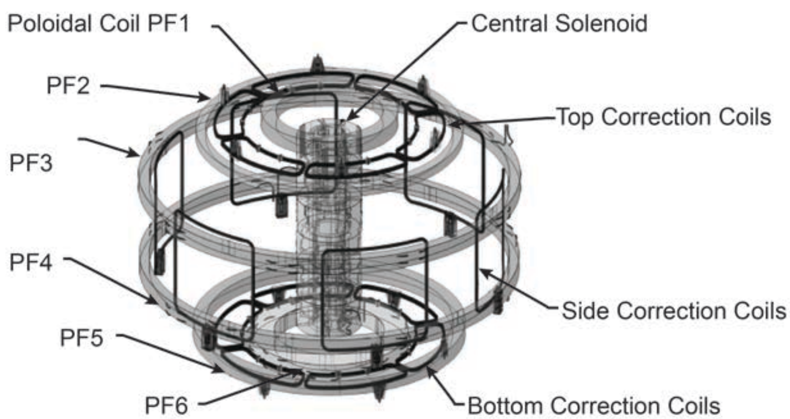

Figure 12

The six PF coils are attached from top to bottom on the exterior of TF coil cases by flexible plates that allow for radial displacements. This positioning presents removal and replacement challenges in the event of failure. All coils include double turn insulation with a metal screen in between that permits detection of an incipient short prior to full failure. This will allow disconnection of coil layers and bypassing with bus bar links. The remaining layers can then be operated in a backup mode at higher current, thereby reducing the risk of coil repair or replacement. While the top two coils could be removed from the cryostat and repaired or re-wound, the remaining four coils would have to be repaired in place. To further reduce risk, the two central coils, which have the greatest access constraints, include metal plate separators with individual ground insulation between the layers.

Correction Coils[14]

The location and geometry of TF, PF and CS magnet coils will not be precise due to small variations in manufacturing and assembly tolerances. This will lead to variances in the axial symmetry of the magnetic field that can in turn cause locked modes in the plasma and consequent disruptions. Such variances are termed “error fields” and the purpose of the CC magnets is to reduce the range of field imperfections by positioning three sets of six coils each at the top, side and bottom of the torus. The bottom set has peak field strength of 4 Tesla, while the side and top sets range from 2.26−2.45 Tesla.

Each coil is rectangular and slightly concave similar to an automotive windshield, and constructed from superconducting niobium-titanium (NbTi) cable-in-conduit coil enclosed in a

20-millimeter thick stainless steel casing. The casing is rigidly connected to the cases of the TF coil set.

Central Solenoid

The CS acts as a transformer inducing the majority of the magnetic flux change needed to initiate the plasma, generate the plasma current, and maintain the current during burn time. The CS is made of six independent coil packs that are composed of superconducting niobium-tin (Nb_3Sn) alloy with each coil pack weighing 110 tons. Approximately 42 kilometers of cable-in-conduit conductor will be used to fabricate the CS. Once integrated it will have peak field strength of 13.1 Tesla, stored energy capacity of 5.5 Gigajoules, and nominally operate at 14 kilovolts and 45 kiloamps. Figure 13 depicts the stack of six CS modules within the support structure.

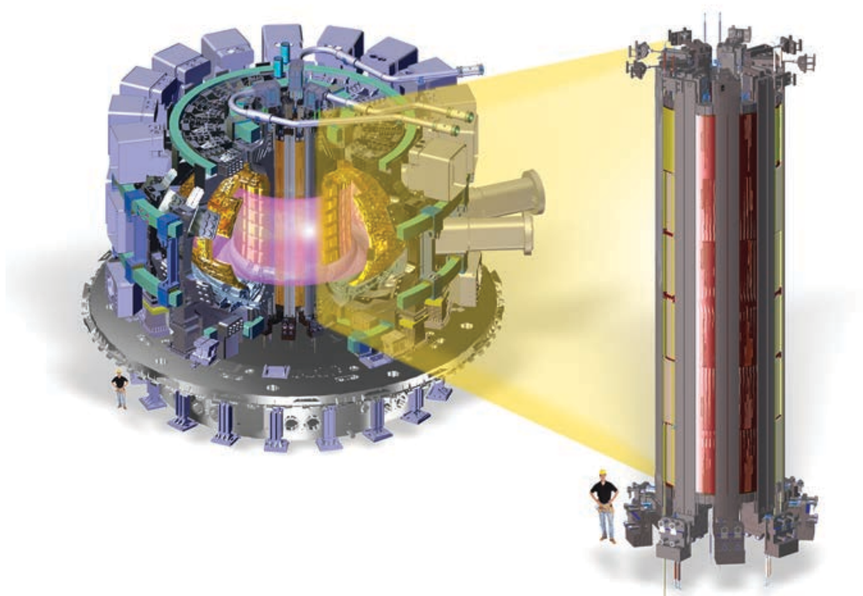

Figure 13

Each of the six coils consists of fourteen turns radially and forty turns high. Only seven lengths of conductor are used to minimize the use of joints and reduce risk of failure. A superconducting bus bar runs vertically across the outer perimeter to connect each coil pack. The stack is supported from the bottom by the TF coils through a pre-loaded structure consisting of nine internal and eighteen external tie-plates. This provides axial pressure on the stack and prevents separation of the modules during operation. Each tie-plate is forged as a single steel member so that when the structure is assembled it can withstand 30 meganewtons of force – equivalent to the force produced by two space shuttles at lift off.

Vacuum Vessel and Internal Elements

The vacuum vessel serves as the plasma chamber and first containment barrier. Inside the vacuum vessel are internal, replaceable components, including blanket modules, divertor cassettes, and port plugs such as the limiter, heating antennae, test blanket modules, and diagnostics modules. These components absorb the radiated heat as well as most of the neutrons from the plasma, and protect the vessel and magnet coils from excessive nuclear radiation and heating.

The heat deposited in the internal components and in the vessel is transferred to the environment by means of a cooling water system. It is comprised of individual heat transfer systems. Some elements of these heat transfer systems are also employed to bake and consequently clean the plasma-facing surfaces inside the vessel by releasing trapped impurities. The system is also designed to prevent the possibility of releases of tritium and activated corrosion products to the environment.

The torus-shaped vacuum vessel is located inside the bore of the TF coils and provides the low gas pressure conditions needed to initiate and maintain fusion reaction plasma. In this torus-shaped chamber, the charged plasma particles follow the magnetic field surfaces, thereby avoiding contact with the vessel walls. The magnet system together with the vacuum vessel and internals are supported by gravity supports, one beneath each TF coil.

The ITER vacuum vessel will be twice as large and sixteen times as heavy as any previous tokamak chamber, with an internal (minor) diameter of 6 meters. It will measure a little over 19 meters (major diameter) across by 11 meters high, and weigh in excess of 5,000 tons.[15]

The vacuum vessel, illustrated in Figure 14, will have double-steel walls, with the interspace filled with cooling water. The plasma-facing surfaces of the vessel will support the in-vessel coils and a continuous layer of blanket modules that will capture the escaping fast neutrons generated by the fusion reactions.

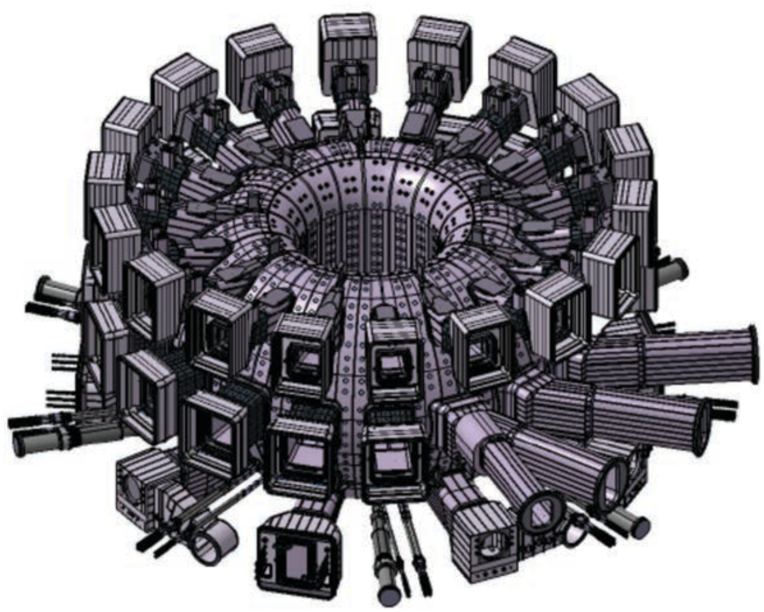

Figure 14

Forty-four ports will provide access to the vacuum vessel for remote handling operations, diagnostic systems, heating, and vacuum systems (eighteen upper ports, seventeen equatorial ports, and nine lower ports).

Because the vacuum vessel is the primary containment barrier against release of tritium and activated dust, it is being constructed according to RCC-MR, the French nuclear code. Due to the nuclear material confinement under high water pressure, it must also meet the essential French safety requirements for nuclear pressure equipment (ESPN).[16]

Blanket System

The blanket covers the interior surfaces of the vacuum vessel, providing shielding to the vessel and the superconducting magnets from the heat and neutron fluxes of the fusion reaction. The neutrons are slowed down in the blanket, where their kinetic energy is transformed into heat energy and collected by the cooling water system.

For purposes of maintenance on the interior of the vacuum vessel, the blanket wall is modular. It consists of 440 individual blanket modules, each measuring 1 x 1.5 meters and weighing up to 4.6 tons. Each module has a detachable first wall, which directly faces the plasma to absorb the plasma radiation and charged particle heat load while protecting the vessel from any plasma impingements, plus a semi-permanent blanket shield dedicated to the neutron shielding. The blanket system is illustrated in Figure 15.

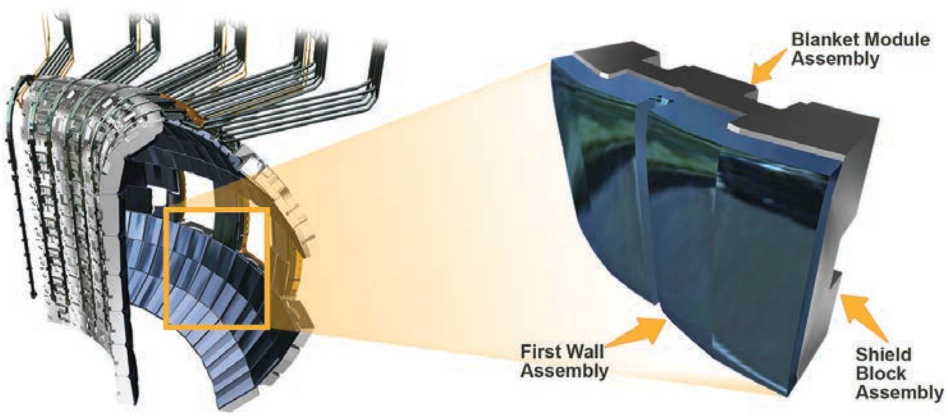

Figure 15

The ITER blanket is one of the most critical and technically challenging components in the ITER system. Together with the divertor, the blanket directly faces the plasma. Because of its unique physical properties, beryllium, a hazardous material requiring special handling, has been chosen to cover the plasma-facing surface "first wall" of the blanket. The rest of the blanket shield layers will be made of high-strength copper backed by stainless steel with internal water-cooling channels.

In-Vessel Coils[17]

There are two sets of the magnetic coils inside the vacuum vessel—the vertical stability coils and the edge-localized mode (ELM) control coils. The vertical stability coils consist of continuous windings above and below the mid-plane that provide fast vertical position control of the plasma. The ELM coils consist of nine sets of three window-framed coils that produce a resonant magnetic perturbation, which limits the energy in ELM events or suppresses the ELMs altogether. Unmitigated ELMs would cause substantial erosion of the plasma-facing components, especially the divertor. Figure 16 shows the configuration of vertical stabilization and mode suppression coils.

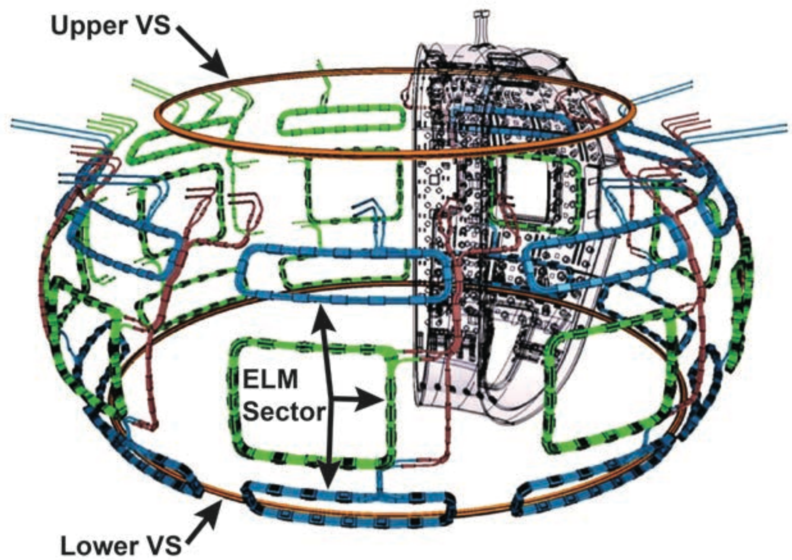

Figure 16

Divertor

Located at the very bottom of the vacuum vessel, the divertor collects and neutralizes the charged particles leaving the plasma— including the inert helium "ash"— and directs these neutral gas particles via the vacuum pumping system to the exhaust processing system. The divertor consists of fifty-four remotely-handled removable cassettes, each holding three plasma-facing component assemblies, or targets, as illustrated in Figure 17. These are the inner and the outer vertical targets, and the dome. The targets are situated at the intersection of magnetic field lines where the high-energy plasma particles strike the components and their kinetic energy is transformed into heat. The heat flux received by these components is extremely intense and requires active water-cooling. The choice of the surface material for the divertor is an important one.[18] Only a few materials are able to withstand temperatures of up to 3,000 °C for the projected 20-year lifetime of the ITER machine, including carbon fiber reinforced carbon (CFC) and tungsten; the current choice is tungsten, since it has less propensity to adsorb the tritium fuel.

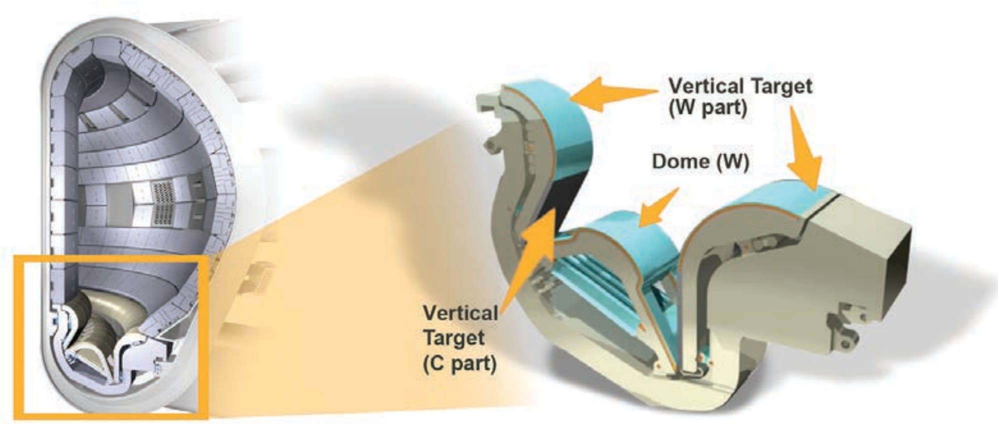

Figure 17

Cryostat and Thermal Shield

The cryostat is a large, stainless steel structure surrounding the tokamak that provides the insulating vacuum for the superconducting magnets, much like a giant thermos bottle. It consists of a single wall cylindrical construction, reinforced by horizontal and vertical ribs. The cryostat is 29.3 meters tall and 28.6 meters in diameter.[19]

The cryostat has many openings, some as large as 4 meters in diameter, which provide access to the vacuum vessel for cooling systems, magnet feeders, auxiliary heating, diagnostics, and the removal of blanket and divertor components. Large, leak-tight bellows are used between the cryostat and the vacuum vessel to allow for differential thermal contraction and expansion in the structures. Each of these openings has sealed closures to allow total evacuation of the cryostat before commencing operation. The cryostat is completely surrounded by a 2-meter-thick concrete biological (bio) shield.

The thermal shield is a set of stainless steel panels, cooled with supercritical helium at 80 K, that provide a thermal radiation barrier between the magnet set and any warmer surfaces (i.e. vacuum vessel and cryostat).[20]

Fueling
The deuterium and tritium fuels used in ITER will be processed in a closed cycle, as illustrated schematically in Figure 18.

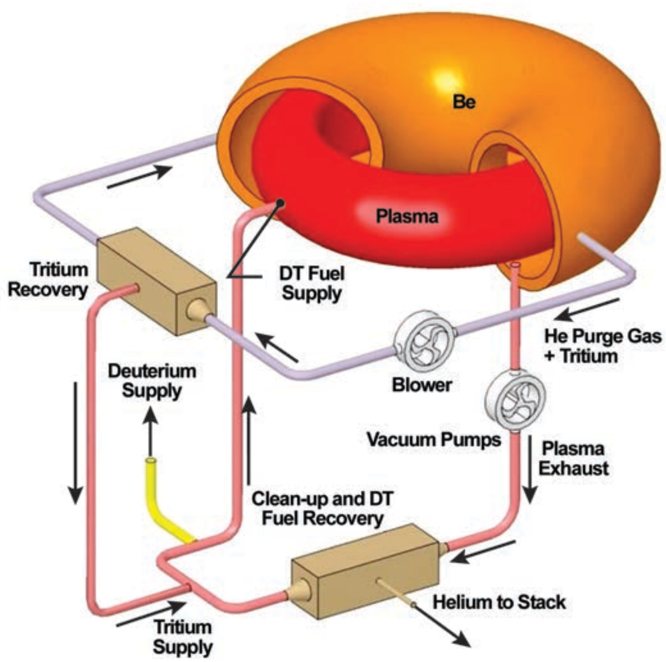

Figure 18

As a first step to starting the fusion reaction, all gases must be evacuated from the vacuum vessel. A vacuum roughing system begins the draw-down, followed by the main pumping system that consists of a set of six torus exhaust cryo-pumps. The cryo-pump panels will be cooled with supercritical helium in order to condense the deuterium and tritium and other gas streams.

Low-density gaseous fuel is then introduced into the vacuum vessel by a gas injection system. Once the fuel is in the vacuum chamber, microwaves are used to pre-ionize the fuel, then an electrical current is applied via the central solenoid coil system that completes the electrical breakdown of the gas, initiates the toroidal current, and forms a magnetically confined plasma.

A second fueling system, a pellet injector, will also be used at ITER. The pellet injector operates like a high efficiency icemaker for frozen fuel pellets. An extruder punches out several millimeter-sized deuterium-tritium ice pellets that are propelled by a gas gun at approximately 300 meters per second — fast and cold enough to penetrate deep into the plasma core where they vaporize and deposit fresh fuel. The frozen pellets are injected through a set of guide tubes located in the inner wall of the vacuum vessel and another guide set of tubes for outer wall injection. Prototype pellet injectors, illustrated in Figure 19, are being developed at Oak Ridge National Laboratory.

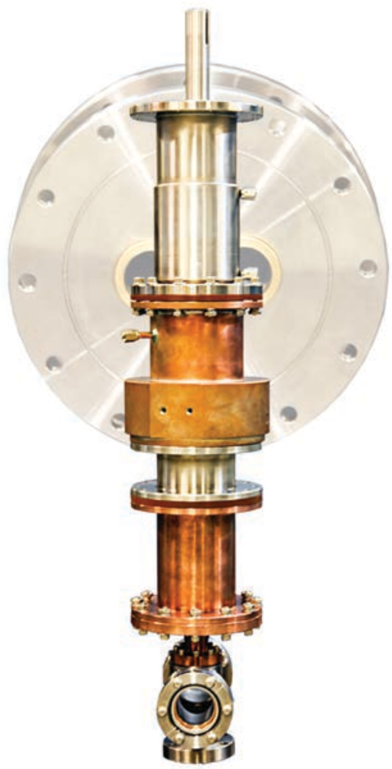

Figure 19

Pellet injection is the principal tool used to control plasma density and is also efficient at controlling Edge Localized Modes, or ELMs. Injecting small frozen deuterium pellets in the edge plasma, has been shown to be effective in ELM mitigation.[21]

Less than 1 g of fusion fuel is present in the vacuum vessel at any moment in time. The divertor, located at the bottom of the vacuum vessel, permits recycling of any fuel that is not consumed. Unburned fuel flows to the divertor, is pumped out and separated from helium produced during the fusion reaction, mixed with fresh tritium and deuterium, and is re-injected into the vacuum chamber.

Plasma Heating

Due to the absence of any equivalent to the sun's gravity, to achieve sufficient pressure, the plasma temperature inside the ITER tokamak must reach greater than 150 million °C, or ten times the temperature at the core of the Sun, in order for the gas in the vacuum chamber to reach the plasma state and for efficient fusion reactions to occur. The hot plasma must then be sustained at these extreme temperatures in a controlled way in order to extract net energy.

ITER will rely on three sources of external heating which will work in concert to provide the required input plasma heating power of 50 MW: Neutral beam injection and two sources of high-frequency electromagnetic waves.[22] The characteristics of these systems are listed in Figure 20.

Heating System Characteristic	Neutral Beam (NB)	Electron Cyclotron (EC)	Ion Cyclotron (IC)
Energy or frequency	1 MeV	170 GHz	40-55 MHz
Power injected-per unit equatorial port (MW)	16.5	20	20
Number of units for the first phase	2	1	1
Total Power (MW) for the first phase	33	20	20

Figure 20

System Cooling
ITER will be equipped with a cooling water system to manage the heat generated during operation of the tokamak. The internal surfaces of the vacuum vessel (first wall blanket and divertor) must be cooled to less than 600 °C only a few meters from the 150-million-degree plasma.

Pressurized water will be used to remove heat from the vacuum vessel and its components, and to cool auxiliary systems such as radio frequency heating and current drive systems, the chilled water system, the cryogenic system, and the coil power supply and distribution system. The cooling water system incorporates multiple closed heat transfer loops plus an open-loop heat rejection system (HRS). Heat generated by escaping fast neutrons slowing down in the vacuum vessel components during the deuterium-tritium reaction will be transferred through the primary cooling water system to the intermediate component cooling water system, and to the HRS, which will reject the heat to the environment. The cooling water system must reject over 1 GW of thermal energy.

Biological shield[23]
Close attention to radiation dosage rates in the tokamak building is critical to ensure occupational safety. The cryostat is surrounded by a two-meter thick concrete bio shield as the primary means of attenuating radiation loads in plant areas that require human activity. The cryostat includes numerous ports for diagnostics, heating, plasma exhaust, etc. that represent strong sources of streaming neutrons and photons. Full three-dimensional nucleonics modeling is therefore necessary to address the geometric details associated with such a large and complex structure.

Figure 21 shows a dose rate map for the central plane and on a plane rotated at 20˚ from the central plane. The central plane bisects the center of an upper diagnostics port and an equatorial port, while the rotated plane bisects two similar ports as well as a divertor port for plasma exhaust. This approach effectively characterizes the areas of peak flux.

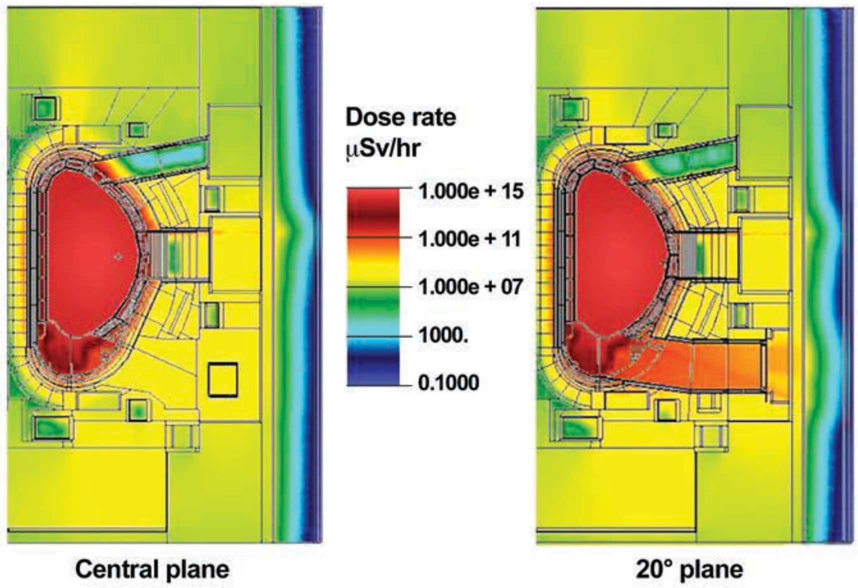

Figure 21

The dose rate varies due to use of shielding plugs of varying thickness at the diagnostic ports, as well as gaps between the port walls and shielding plugs. The divertor pumping ports do not employ shielding plugs and therefore strongly affect the dose rate at the bio shield, as indicated in the lower portion of 20˚ plane map. Analysis has shown that the bio shield reduces the total prompt operational dose by six orders of magnitude. The peak values of the prompt dose rates at the back surface of the bio shield were 240 μSv/hr and 94 μSv/hr corresponding to the regions behind the divertor port and the equatorial port, respectively. For comparison purposes, the United States Code of Federal Regulations (10CFR20 and 10CFR835) limits radiation worker total annual exposure to 50 mSv, or 50,000 μSv.

Instrumentation & Controls

The Instrumentation & Control (I&C) system for ITER relies on a vast array of diagnostic sensors, analytical control schema software and actuators designed to operate the tokamak safely and enable the next generation of R&D for burning plasma.

<u>Diagnostic Instruments</u>[24]

The ITER I&C system will employ a wide number of individual measuring systems that have been drawn from the full range of modern plasma diagnostic techniques, including:

- plasma position reflectometry
- neutron flux monitoring and particle analysis
- neutral particle analysis
- visible and infrared imaging
- microfission chambers
- edge imaging x-ray crystal spectroscopy
- pressure gauges
- fiber optic current sensing
- H-Alpha and visible spectrometry
- residual gas analysis
- continuous external Rogowskis
- gamma ray spectrometry
- discrete inductive sensing
- neutron activation systems
- high and low field side reflectometry
- x-ray crystal spectroscopic surveying
- thermocouples and thermography
- bolometry

- charge exchange recombination spectrometry
- collective and edge Thomson scattering
- beam emission spectrometry
- impurity monitoring
- electron cyclotron emission
- motional Stark effect polarimeter
- Langmuir probes
- polarimetry
- neutron and x-ray cameras
- interferometry
- vacuum ultraviolet surveying, edge imaging and spectrometry

Many instruments will be located in ports and constructed as standardized modules that extend from the plasma "first wall" through the vacuum vessel, cryostat, interspace, and bio-shield as illustrated in Figure 22.

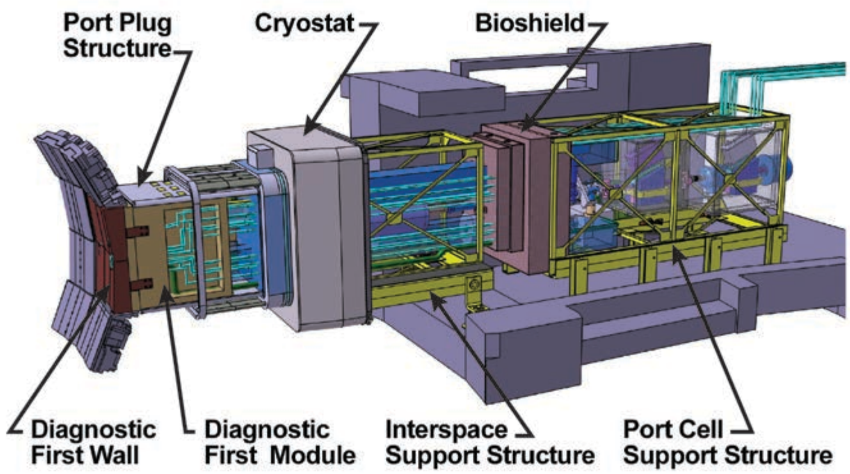

Figure 22

Because of the harsh environment inside the vacuum vessel, these systems will have to cope with a range of phenomena not previously encountered in diagnostic implementation, all while performing with great accuracy and precision. The levels of neutral particle flux, neutron flux and fluence will be respectively about 5, 10, and 10,000 times higher than the harshest conditions experienced in today's magnetic fusion machines, while the pulse duration will be about 100 times longer. To ensure reliable control and measurements, each desired quantity is measured using two or more independent diagnostic techniques.

<u>Control, Data Access and Communication (CODAC)</u>[25]
Unusually high energy level, heat flow, neutron flux and long pulse duration combine to create challenging conditions for control of burning plasma. The plasma control system (PCS) will communicate with at least forty-five unique diagnostic packages and twenty different actuator systems to sense and respond to rapidly evolving conditions occurring in the plasma stream. The various magnetic coils, gyrotron heaters, coolant pumps and associated subsystems are also active elements in the overall control schema.

Plasma dynamics can be viewed as a composite of complex large-scale flows, turbulent small-scale flows and energetic particle interactions. Due to the complexity and uncertainty of responses from the plasma to active control actions, the PCS must be highly robust and reliable in terms of protecting the plant investment during experimental operations. Understanding of reaction dynamics will systematically progress over time through the planned research agenda. The frequency and number of pulse cycles therefore becomes a key factor in the rate of understanding and evolution of steady-state operating principles. Approximately 10,000–12,000 pulses have been projected for the early hydrogen/helium phase of plant commissioning; performance characterization is based on an assumption of dual-shift research operations.

Two separate types of control logic will ultimately be required for ITER: (a) continuous control and (b) exception forecasting, detecting and handling. Continuous control requires the development of algorithms that can reliably produce feedback to regulate and maintain a nominal operating scenario, while exception handling control demands effective responses to off-nominal and fault conditions, both of which are likely during experimental operations.

The fusion research community has had a strong and significant R&D program underway involving smaller scale tokamak reactors deployed around the world. This work provides a sound foundation upon which to build the scientific understanding and technological control that will result from the scaled up ITER R&D opportunity.

Research Plan[26]

The following lists of objectives and goals by phase of operations are extracted from the ITER Research Plan. Research operations are divided into four phases associated with a gradual progression from initial characterization and validation of the performance attributes of the tokamak system to full-up operation in a burning plasma state that allows achievement of mission objectives. The four phases include:

1. Hydrogen and Helium (HH) Phase
2. Deuterium (D) Phase
3. Deuterium-Tritium Phase 1 (DT1)
4. Deuterium-Tritium Phase 2 (DT2)

Hydrogen and Helium Phase: The overall objectives for the hydrogen/helium phase of ITER operation are to:

- Establish routine operation of the tokamak and its subsystems with plasmas;
- Commission all installed heating and diagnostic systems with plasma;
- Commission installed fueling systems with plasma;
- Commission and integrate all installed control systems (including in-vessel coil systems);
- Commission and integrate all safety related systems;
- Demonstrate plasma operation to full technical performance;
- Perform validation of diagnostic data and demonstrate consistency of measurements;
- Characterize aspects of plasma performance critical for subsequent phases of operation;
- Characterize operational boundaries and off-normal events;
- Demonstrate reliable avoidance or mitigation of off-normal events;
- Validate licensing assumptions concerning disruptions;
- Characterize hydrogenic retention and demonstrate techniques to be used later for control of tritium inventory;
- Establish and characterize type-I ELMy H-modes, most likely in helium plasmas, and demonstrate ELM mitigation/ suppression;
- Demonstrate, to the extent possible, plasma performance and scenarios envisaged for D and DT, including plasma operation on tungsten plasma facing components;
- Conduct first exploration of fusion plasma physics at ITER scale and parameters.

Deuterium Phase: The primary goals of the deuterium phase are to:

- Commission the heating and current drive (CD) systems to the level of performance and reliability required for a successful DT program;

- Establish reliable, long-pulse plasma operation on tungsten divertor targets in ohmic, L- and H-mode;
- Develop, demonstrate, and validate H-mode scenarios up to the highest parameters achievable and, if the H-mode scaling turns out to be favorable, to full machine parameters;
- Demonstrate that hydrogenic retention in H-mode is acceptable for DT operation;
- Demonstrate ELM amelioration sufficient for divertor protection over the expected operational range;
- Commission and validate H-mode relevant diagnostics (e.g. neutron detectors, pedestal diagnostics).

<u>Deuterium-Tritium Phase 1</u>: During the DT1 phase ITER should achieve extended burn in inductively driven plasmas with $Q \geq 10$ for a range of operating conditions, and a duration sufficient to achieve stationary conditions on the time scales characteristic of plasma processes. More detailed ITER operational objectives during the DT1 inductive and non-inductive phases are as follows.

Inductive plasmas:

- Develop burn control techniques for DT plasmas, including power and particle exhaust, active MHD control;
- Achieve fusion power of several hundred MW;
- Demonstrate $Q \geq 10$ for several hundred seconds;
- Develop a hybrid mode of operation for longer burn durations or higher fusion performance to the extent possible;
- Pursue a program of burning plasma research based on $Q \geq 10$ operating scenarios.

Non-inductive plasmas: The objective of the DT1 Phase will be to develop – to the maximum extent possible – the basis for non-inductive plasma operation toward the ultimate moderate-Q, steady-state operation. It is likely that this goal will not be fully accomplished until the DT2 Phase:

- Extend current drive studies to DT plasmas – quantify off-axis CD capability;
- Build on the DT-inductive program by establishing a range of target q-profiles with early heating and current ramps;
- Commission feedback control algorithms for H&CD, MHD stability control, fueling and divertor power handling in relevant regimes;
- Explore control algorithms in the presence of strong heating over the current relaxation time, and validate the models for these control algorithms;
- Develop scenarios close to MHD limits and explore stability/control;
- Develop fully non-inductive plasmas and extend performance to $Q = 5$;
- Extend pulse length towards 3000s with $Q \geq 5$;
- Pursue burning plasma physics studies in non-inductive scenarios.

Towards reactor-relevant plasmas and technology: During and subsequent to the development of the operation that achieves these objectives, it is envisioned that experiments will be conducted to explore a wide range of plasma physics issues in the burning plasma state. However, the extended research plan might require additional time for the physics research program in DT2. Key scientific objectives include:

- Improve the understanding of plasma physics phenomena in reactor conditions;
- Validate theory- and simulation-based predictive models of key performance related phenomena;

- Demonstrate the capability for using model-based tools for controlling primary aspects of reactor-grade plasma.

<u>Deuterium-Tritium Phase 2</u>
The present version of the Research Plan is restricted to identifying possible research priorities for DT2. These include:

- The test blanket module development program;
- Full steady-state demonstration with additional heating and current drive tools;
- Extension of ITER regimes towards those required for a reactor, e.g. higher β and higher radiated power fraction;
- Demonstration of compatibility of operation regimes with a DEMO-relevant wall;
- Demonstration of potential DEMO regimes with a reduced number of heating and current drive systems;
- Demonstration of the plasma control required for DEMO using only DEMO-compatible diagnostics.

Safety and Licensing

Safety is a top-priority issue for the project and includes consideration of the safety of the project staff and workers on site, the local population and the environment. French nuclear regulations have been applied throughout the design phase of the project, and will continue to be followed during construction, operation, and decommissioning.[27]

The fusion process itself is inherently safe. In a tokamak fusion device, the quantity of fuel present in the vessel at any one time is sufficient for a few-seconds burn only. It is difficult to reach and maintain the precise conditions necessary for fusion; any significant degradation of these conditions will cause the plasma to cool within seconds and stop the reaction. There is no danger of a run-away reaction, because fusion does not involve a self-perpetuating chain reaction.

When the highly energetic neutrons interact with the walls of the internal components and the plasma chamber, these materials become activated. In-vessel materials can also become contaminated with small amounts of tritium and radioactive dust composed mainly of beryllium and tungsten.

In ITER, confinement of these materials will be based on the principle of defense-in-depth—materials with the highest radioactive content are located in the very center, surrounded by multiple protective layers. Maintenance and refurbishment of the more radioactive elements and components of the tokamak are performed using machines and tools controlled remotely to avoid human exposure to radioactivity. As previously discussed, two-meter-thick protective concrete walls serving as a bio shield completely surround the tokamak.

During the operational lifetime of ITER, remote handling will be used to refurbish components of the vacuum vessel. All waste materials will be treated, packaged, and stored on site in a Hot Cell building to maintain total separation. The half-life of most radioisotopes contained in this waste is less than ten years. The fusion reaction will produce no long-lived waste. Within 100 years, the radioactivity of the materials will have diminished to such a degree that the materials can be recycled for use in future fusion plants.[28]

The confinement of tritium within a closed fuel cycle is one of the most important safety objectives at ITER, because although tritium has a relatively short radioactive half-life of 12.3 years it nonetheless possesses a high radio-toxicity. The total amount of tritium present on site will have a licensed limit of 4 kg. A multiple-layer barrier system has been designed to protect against spread or release of tritium. The first level of the safety confinement barrier is the vacuum vessel itself. Inside this double-steel container, the fusion reaction takes place within a near-vacuum. All pumps, pipes, valves and instruments leading into the vacuum vessel are highly leak-tight.

Surrounding the first confinement system is a second level of security comprising all vessels or systems that surround the vacuum vessel, including buildings as well as advanced detritiation systems for the recovery of tritium from gas and liquids. In ITER, these highly developed detritiation systems will work efficiently to keep the fusion fuels recycled within a closed system and maintain any releases well below regulatory limits. These systems have been designed to remove tritium from liquids and gases for reinjection into the fuel cycle. Remaining effluents will be well below authorized limits. Gaseous and liquid tritium releases to the environment from ITER are predicted to have a dosage rate below 10 µSv per year. This is 1,000 times lower than ITER's General Safety Objective of 10 mSv per year (the regulatory limit in France). Scientists estimate exposure to natural background radiation to be approximately 6,200 µSv per person per year.

To mitigate seismic risks, the ITER tokamak complex is constructed on a foundation of specially reinforced concrete, and will rest upon bearing pads on top of pillars that are designed to reduce the impact of earthquakes. Cadarache, France is classified as an area of moderate seismic activity. The facility will be equipped with seismic sensors around the site to record all seismic activity, however minor.

ITER safety processes are in full compliance with French and international regulations, and the ITER installation is classed as a "basic nuclear installation" by French authorities. A successful Public Enquiry was held in 2011, and on 20 June 2012 the ITER Organization was informed in writing by the French Nuclear Safety Authority (Autorité de Sûreté Nucléaire) that—following an in-depth technical inspection—the operational conditions and the design of ITER as described in the ITER safety files fulfilled expected safety requirements. As part of its responsibilities as a nuclear operator, the ITER Organization will perform regular checks on the installation during construction and operation. French nuclear authorities will also audit and inspect the ITER Organization's application of regulations.

The Coming Era of Burning Hydrogen Plasma

In 2015, the ITER tokamak complex began rising up out of the ground in southern France. Completion of the initial R&D facility is projected to require approximately 10 years. Additional capabilities for deuterium-tritium research operations will follow. The timeline is affected by the rate of annual funding among the seven ITER Partners. Accelerations or delays, both of which are possible, depend on socioeconomic conditions around the world.

The achievement of a capability to sustain burning plasma under short-term, quasi-steady-state conditions is the effective "turning point." At this critical historic juncture, the proven, very large, net energy gain will form a compelling basis for follow-on investments in next-generation containment materials and lithium-to-tritium transmutation techniques necessary to enable long-term, steady-state operations. This technology pull is already underway, but the pressure can be expected to increase in order to seize the commercial opportunity demonstrated by burning hydrogen plasma.

In the global fusion physics community, the step following ITER is envisioned to be state-sponsored, commercial-scale, electric-generating prototypes termed "DEMOs," and the expectation is that the current ITER Partners will each independently pursue national DEMOs. At this stage, levels of government, industry and academic participation will likely vary according to national socioeconomic policies. China, Japan and Korea have initiated planning for DEMOs, and are already investing in upgraded tokamak laboratories and infrastructure (e.g., supercomputers and superconducting cable production).

The power of the stars could achieve practical realization by the mid-twenty-first century. There is little remaining doubt in the informed fusion physics and engineering communities that controlled nuclear fusion for electric generation can be achieved; the only uncertainties are when, at what cost, and by whom? If the paradigms of history are in any way prophetic, then the nations that lead in hydrogen fusion "know how" will likely become the leaders of our world in the future. Energy, a fundamental factor of production, will be available to all due to the ubiquity of fusion fuel; however, the science and technology prowess to "create stars" must be sponsored by national leaders with vision.

ACKNOWLEDGEMENTS

This compilation was based largely on information made available in the public domain by the ITER Organization and US ITER Project Office. In many cases, the technical literature was consulted for details and these references are specifically identified where applicable. The ITER design remains subject to further refinements as the Project continues to evolve. While design details were accurate at the time of writing, some changes are to be expected. This document was prepared by staff in the US ITER Project Office at Oak Ridge National Laboratory and was supported by the U.S. Department of Energy, Office of Science, Fusion Energy Sciences. The document will be maintained by the US ITER Project Office in the public domain at https://www.usiter.org. The site will be updated as necessary to preserve technical accuracy as the ITER Project progresses.

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Figure 18: Fuel Cycle Schematic. Source: ITER Organization

Figure 19: Prototype Pellet Injector. Source: US ITER/ORNL

Figure 20: Characteristics of Heating and Current Drive Systems. Source: US ITER

Figure 21: Dose rates at two planes in ITER. Source: A. M. Ibrahim *et al.*, Fusion Science Tech. **60**, 2 (2011), pp. 676-680.

Figure 22: Diagnostic Instrument Location in Ports. Source: D. Johnson *et al.*, in 20th Topical Conference on High-Temperature Plasma Diagnostics, 2014.

END NOTES

[1] U.S. National Academy of Engineering, *Grand Challenges for Engineering* (National Academies of Science Press, 2008).

[2] D. A. Petti, B. J. Merrill, J. P. Sharpe, L. C. Cadwallader, L. El-Guebaly, and S. Reyes , Nucl Fusion **47**, 7 (2007) S427–S435.

[3] P. Z. Grossman, *U.S. Energy Policy and the Pursuit of Failure* (Cambridge University Press, 2013).

[4] W. M. Stacey, *The Quest for a Fusion Energy Reactor: An insider's account of the INTOR Workshop* (Oxford University Press, 2010).

[5] Executive Summary of the IAEA Workshop, mid-1985 to 1987, International Tokamak Reactor Group, Nucl Fusion **28**, 4 (1988).

[6] U.S. National Research Council Burning Plasma Assessment Committee, *Burning Plasma: Bringing a Star to Earth* (National Academies Press, 2004).

[7] IAEA Information Circular, *Agreement on the Establishment of the ITER International Fusion Energy Organization for the Joint Implementation of the ITER Project* (INFCIRC/702, 2007).

[8] https://en.wikipedia.org/?title=Tokamak, accessed June 19, 2015.

[9] http://www.tokamak.info/, accessed June 22, 2015.

[10] Controlled fusion power was first demonstrated at the 10.7 MW level at the Princeton University Tokamak Fusion Test Reactor in 1994 and replicated in 1997 at the 16 MW level (current world record) at the Joint European Torus operated by Culham Science Center owned and operated by the United Kingdom Atomic Energy Authority outside Oxford, United Kingdom.

[11] *ibid* (6), p. 2.

[12] https://en.wikipedia.org/wiki/Électricité_de_France, accessed June 22, 2015.

[13] A. D. Mankani, I. Benfatto, J. Tao, J. K. Goff, J. Hourtoule, J. Gascon, D. Cardoso-Rodrigues, and B. Gadeau in IEEE/Nuclear and Plasma Physics Society 25th Symposium on Fusion Engineering, 2011, SP#-43.

[14] A. Foussat, P. Libeyre, N. Mitchell, Y. Gribov, C. T. J. Jong, D. Bessette, R. Gallix, P. Bauer, and A. Sahu, IEEE Transactions on Applied Superconductivity, **20**, 3, (2010).

[15] C. Sborchia *et al.* in *25th Symposium on Fusion Engineering*, 2013, pp. 1-8.

[16] H. J. Ahn *et al.*, in *American Society of Mechanical Engineers Pressure Vessels and Piping Conference, Vol 1: Codes and Standards*, 2011, pp. 275-283.

[17] C. Neumeyer *et al*., J. Fusion Science and Technology (2011), 60.

[18] T. Hirai *et al.*, in *Proceedings of the 27th International Symposium on Fusion Nuclear Technology*, 2012, pp. 1798–1801.

[19] B. Doshi *et al.*, in *25th IEEE Symposium on Fusion Engineering*, 2013, pp. 1-6.

[20] C.H. Noh *et al.*, in *Proceedings of the 27th International Symposium on Fusion Nuclear Technology*, 2012.

[21] L. R. Baylor *et al.*,Nucl. Fusion **49** (2009), 085013.

[22] F. Wagner *et al.*, Plasma Phys. Control. Fusion **52** (2010), 124044.

[23] A. M. Ibrahim, M. E. Sawan, S. W. Mosher, T. M. Evans, D. E. Peplow, P. P. Wilson, and J. C. Wagner, Fusion Science Tech. **60**, 2 (2011), pp. 676-680.

[24] D. Johnson *et al.*, in 20th Topical Conference on High-Temperature Plasma Diagnostics, 2014.

[25] D. Humphreys, Physics of Plasmas **22** (2015), 021806.

[26] Extracted from ITER Research Plan, ITER Organization, IDM UID-2FB8AC, Nov. 2009.

[27] C. Alejaldre, J. Elbez-Uzan and L. Rodriguez, in 24th IAEA Fusion Energy Conference, 2012.

[28] S. Rosanvallon, B. C. Na, M. Benchikhoune, J. Uzan-Elbez, O. Gastaldi, N. Taylor , and L. Rodriguez, Fusion Engineering and Design **85,** s 10-12 (2010), pp. 1788-1791.