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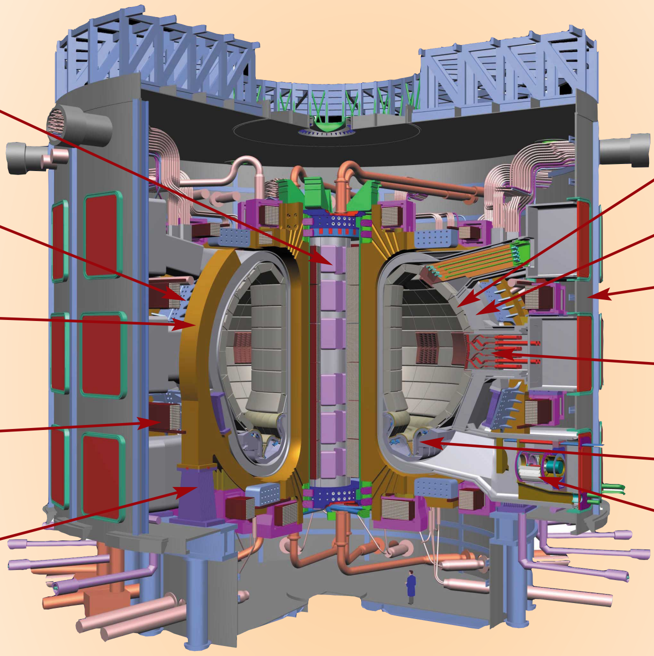
Successful development of fusion energy requires the demonstration of scientific and technological feasibility. ITER will provide the conditions required to integrate high power amplification plasmas, at or near steady state conditions, with essential fusion energy technologies, and demonstrate safe operation of a fusion power system. Its design has focused the attention of the world fusion community on key scientific and technical issues. Through this collaboration, the cost and benefits of this essential step can be shared. The world program is scientifically and technically ready to take this important step.

The ITER collaboration began in the late 1980's. The then Soviet Union, the United States, Japan, and Europe established a collaboration under the auspices of the International Atomic Energy Agency. After Conceptual Design Activities (CDA) between 1988-1990, Engineering Design Activities (EDA) began in 1992 and were completed in mid-2001 with the production of the ITER Final Design Report, which was sufficiently detailed for a reliable cost estimate and to begin talking to licensing authorities of potential project hosts. Negotiations on joint implementation of ITER have since been underway between Canada, Europe, Japan, and the Russian Federation, and were joined by the People's Republic of China, the United States of America and the Republic of Korea during 2003. These Negotiations are drawing up the international agreement for construction, exploitation and decommissioning of ITER, deciding who will pay for what, how the project will be organised and staffed, and comparing sites. The site will be chosen from Cadarache (France), Clarington (Canada), Rokkasho (Japan) and Vandellòs (Spain). Negotiations are now sufficiently promising that the project has entered a phase of "Transitional Arrangements" (ITA) leading to the establishment of the ITER International Fusion Energy Organisation which will build and run ITER. Technical work, conducted by the ITER International team and the Participant Teams of the Negotiators, underpins the Negotiations technically and is preparing for construction by the writing of detailed technical specifications for the most urgent procurements, engaging potential licensing bodies, and putting in place the necessary project infrastructure for such a complicated multi-party construction. Site choice is now expected around the end of 2003, with rapid finalisation of the Agreement immediately thereafter, allowing the project to become established in 2004 and, following licensing, begin construction in 2006, with a view to obtaining the first plasma in 2014. This will be followed by an exploitation phase lasting roughly 20 years.



The ITER tokamak has an elongated plasma and a single null poloidal divertor, which is the main point of contact of the plasma with the material boundary. The plasma is fuelled and heated to reach a high power amplification (Q) burn of deuterium-tritium (DT). The heating systems can be further used to drive the plasma current, extending the nominal inductive burn of 300 s up to steady state. Plasma control is provided by the poloidal field system, and the pumping, fuelling and heating systems, based on feedback from diagnostic sensors.

Main Plasma Parameters and Dimensions	
Total fusion power	*500 MW*
Q — Fusion power/auxiliary heating power	*≥ 10*
Average (14 MeV) neutron wall loading	*0.57 MW/m^2*
Plasma major radius	*6.2 m*
Plasma minor radius	*2.0 m*
Plasma current	*15 MA*
Toroidal field @ 6.2 m radius	*5.3 T*
Plasma Volume	*837 m^3*
Installed auxiliary heating/current drive power	*73 MW*

The major tokamak components are the superconducting toroidal and poloidal field coils which magnetically confine, shape and control the plasma inside the toroidal vacuum vessel. The internal, removable components, including blanket modules, divertor cassettes, and port plugs for the plasma limiter, heating antennae, test blanket modules and diagnostics sensors, absorb most of the radiated heat from the plasma and protect the vessel and magnet coils from excessive nuclear radiation. The divertor exhausts the helium from the fusion reaction and limits the concentration of impurities in the plasma.

The heat deposited in the components is rejected to the environment via the cooling water system. The tokamak is housed in a cryostat, with thermal shields between the hot parts and the magnets and support structures which are at cryogenic temperature. Successive barriers are provided for tritium (and activated dust). These include the vacuum vessel, the cryostat, and active air conditioning systems, with detritiation and filtering capability in the building.

Under normal operation of ITER, the additional radioactive dose to any member of the public will be below 1% of natural background. Under the worst imaginable sequence of events, the additional radioactive dose to any member of the public will be below natural background. Even in hypothetical situations, no member of the public will need to be evacuated for technical reasons.

Since the start of the EDA, $920M (year 2000 values) has been spent on technology R&D, mostly on seven large R&D projects (toroidal field and central solenoid model coils, vessel, blanket and divertor models, and blanket and divertor remote handling), to give confidence in the manufacturing capability to build ITER, and in the safe and reliable operation of components. Direct capital costs for ITER have been calculated at $3800M. Staff and R&D costs during construction add a further $760M. Operation costs will be ~$260M/annum, and decommissioning will cost ~$470M.