ITER

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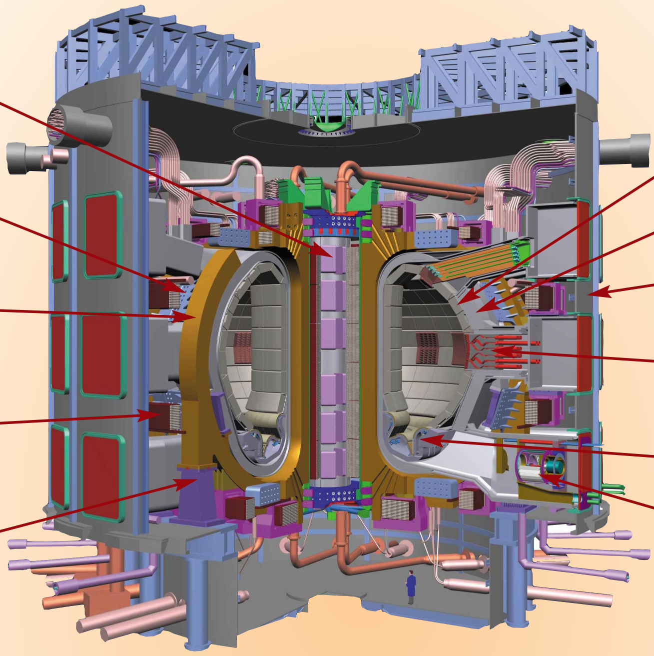
Successful development of fusion energy requires the demonstration of scientific and technological feasibility. ITER will provide the conditions required to integrate high energy gain plasmas at or near steady-state conditions with essential fusion energy technologies, and demonstrate safe operation 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 important 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. Negotiations for joint implementation of ITER, supported by Coordinated Technical Activities (CTA), are now underway between Canada, Europe, Japan, and the Russian Federation, to draw up the agreement for construction, exploitation and decommissioning of ITER, prepare for licensing, and decide on a site. Following the site choice and the commitment by the ITER Participants of suitable funds, expected in 2003, the construction phase (8 years) can start. This will be followed by an exploitation phase lasting roughly 20 years.



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*

ITER 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 energy multiplication (Q) burn of deuterium-tritium (DT). The heating can be further used to drive the plasma current, extending the nominal inductive burn of 300 s up to ~ 3600 s, or longer. Plasma control is provided by the poloidal field system, and the pumping, fuelling and heating systems, based on feedback from diagnostic sensors.

The major components of the tokamak 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. The fullest possible safety assessment of the design was produced in mid-2001. This forms the basis for submissions to obtain regulatory approval for construction for a specific site. Informal contact has already been made with regulatory authorities of potential host countries.

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.