THE PHYSICS AND TECHNOLOGY BASIS OF ITER AND ITS MISSION ON THE PATH TO DEMO

SOFE - ICOPS
San Diego, 01 – 05 June 2009

Guenter Janeschitz
Senior Scientific Advisor for Technical Integration (SSATI)
To the PDDG
ITER Organization

Acknowledgements: D. Campbell, M. Glugla, M. Merola N. Mitchell, A. Tanga, A. Tesini
Outline

- Motivation for fusion development (very brief):
  - Climate change, finite Oil and Natural Gas resources
  - Fusion as long-term solution for part of the problem

- Fusion basics – magnetic confinement

- ITER and its mission,
  - Physics Development towards ITER
  - Technology Development for ITER – the ITER design (see also S. Chiocchio)
  - Status of ITER

- Road-map and Technologies needed for DEMO (very brief)

- Conclusion
The climate change problem, the finite oil resources

World oil reserves / resources

Finite oil resources and reserves
Depending on growth oil could run out / be expensive within 2 decades

To replace only half of it means Terra W of energy from other sources (nuclear, coal, renewables, fusion)
Schematic View of a future Fusion Power Reactor

Fusion can be a long term solution not a short term fix

Power generated by hot plasma

(20 keV = 200 Mio °C)

4/5 th of Power transported by 14 MeV Neutrons
Magnetic Confinement of a plasma with 10 to 20 keV

A toroidal magnetic system needs:

- a helical field configuration to compensate drifts
- a magnetic well

Two successful systems:

Stellarator / Tokamak - ITER
A modern Tokamak – Vertical-, Radial-, Divertor Fields

- Poloidal field coils
- Vertical Field
- Magnetic well
- Radial control
- Radial Field
- Vertical control
- Transformator
- Divertor
Methods for the heating of a tokamak plasma

- ICRH
- ECRH
- Transmission line
- Wave guide
- Plasma current
- Ohmic heating
- Neutrals via neutral beam injection
- High frequency heating

Source: Forschungszentrum Jülich
Energy and particle Transport is governed by turbulence

Ion Turbulent energy transport sets in at a critical temperature gradient which depends on the local temperature.

Radial size of turbulent structures can be reduced by ExB shear, by magnetic shear and by zonal flows produced by the turbulence itself.
Confinement Scaling Relation

\[ \tau_{E,\text{th}}^{\text{ELMy}} = 0.0562 \times I^{0.93} B^{0.15} P^{-0.69} n_{e,19}^{0.41} M^{0.19} R^{1.97} \varepsilon^{0.58} \kappa^{0.78} \]

- Extrapolation of global thermal confinement time to ITER-FEAT (R = 6.2 m), using the IPB98(y,2) scaling (t_E=3.9s)
  
  - Only Q= 10 possible, no ignition
  
  - For ignition R~ 8 m and t_E >5.7 sec needed

- ITER represents an extrapolation of a factor ~4 beyond existing database
We start to explain the blue profile types by physic models!!

q profiles for standard and advanced scenarios

Pressure profiles for standard and advanced scenarios

Pressure profiles produce Bootstrap Currents => important for Steady State Operation

H-mode pedestal ballooning unstable => ELMs
Family of Tokamaks defined the ITER Physics Basis

Remote Handling in JET

JET – Internals & Plasma
Progress in Fusion Tripple Product similar to Progress in Microprocessor Development

Fusion Reactor Class in the area of 3 to 6 $10^{21}$
- $Ti > 10$ keV
  - $n_e x \tau_E \sim 6.0 \times 10^{20} \text{ m}^{-3} \text{s}$
  - $n_e \sim 1.0 \times 10^{20} \text{ m}^{-3}$
  - $\tau_E \sim 6.0 \text{ s}$

=>$Energy Confinement time between 3 and 6 sec$
Output of a system code for possible ITERs

Physics and engineering constraints are combined in a "System Code" which is able to calculate a consistent parameter set.

\[ \kappa q = \left(1 + \kappa^2\right) a^2 \pi B \mu_o R I_p \]

Avoid disruptions: \( q > 2.5, k < 1.8 \) to 2.0

Cross sections:
- Critical values \( B_c < 12.5 \) Tesla
- \( \Delta = \approx 1.2 \) m
- \( \delta = \approx 1.0 \) to 1.3 m

\[ \rightarrow \text{ implies } a = f(R_c) \]

Transformer coil cross section

Plasma central axis

R = 8.15 7.53 6.9 6.3 5.67
Rc = 4 3.75 3.5 3.25

\[ Q = \frac{R_{fus}}{P_{aux}} \]

Fusion power depends on confinement and size

Confinement multiplier

<table>
<thead>
<tr>
<th>Q</th>
<th>0.1</th>
<th>0.5</th>
<th>1</th>
<th>10</th>
<th>100</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.5</td>
<td>0.6</td>
<td>0.7</td>
<td>0.8</td>
<td>0.9</td>
<td>1</td>
</tr>
<tr>
<td>0.5</td>
<td>0.6</td>
<td>0.7</td>
<td>0.8</td>
<td>0.9</td>
<td>1</td>
</tr>
<tr>
<td>0.5</td>
<td>0.6</td>
<td>0.7</td>
<td>0.8</td>
<td>0.9</td>
<td>1</td>
</tr>
<tr>
<td>0.5</td>
<td>0.6</td>
<td>0.7</td>
<td>0.8</td>
<td>0.9</td>
<td>1</td>
</tr>
<tr>
<td>0.5</td>
<td>0.6</td>
<td>0.7</td>
<td>0.8</td>
<td>0.9</td>
<td>1</td>
</tr>
</tbody>
</table>

\[ H \rightarrow \text{ Confinement Multiplier} \]

<table>
<thead>
<tr>
<th>HH</th>
<th>0.5</th>
<th>0.6</th>
<th>0.7</th>
<th>0.8</th>
<th>0.9</th>
<th>1</th>
<th>1.1</th>
<th>1.2</th>
<th>1.3</th>
</tr>
</thead>
<tbody>
<tr>
<td>3</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
The ITER Machine

- **V:** 840m$^3$
- **R/a:** 6.2m /2m
- **Vertical elongation:** 1.85
- **Triangularity:** 0.45
- **Density:** $10^{20}$m$^{-3}$
- **Peak Temperature:** 17keV
- **Fusion gain Q = 10**
- **Fusion Power:** ~500MW
- **Ohmic burn 400 sec**
- **Goal Q=5 for 3000 sec**
- **Plasma Current:** 15MA
- **Toroidal field:** 5.4T
- **Toroidal field:** 5.4T
What is ITER?

ITER is a major international collaboration in fusion energy research established in the 80th by Reagan - Gorbachev involving the EU (plus Switzerland), China, India, Japan, the Russian Federation, South Korea and the United States.

Physics Goals:

- ITER is designed to produce a plasma dominated by $\alpha$-particle heating
- produce a significant fusion power amplification factor ($Q \geq 10$) in long-pulse operation
- aim to achieve steady-state operation of a tokamak ($Q = 5$)
- retain the possibility of exploring ‘controlled ignition’ ($Q \geq 30$)

Technology Goals:

- demonstrate integrated operation of technologies for a fusion power plant
- test components required for a fusion power plant
- test concepts for a tritium breeding blanket
What were / are the major Challenges in Physics?

- The solution to the divertor peak heatflux problem (solved / ongoing)
  - The development of the radiative divertor
- The prediction of the ITER Energy Confinement and thus the definition of its size (solved / ongoing)
  - Shortfalls of scaling as a predictive tool => Physics understanding needed!
- The impact of magnetic ripple on energy confinement (solved / ongoing)
  - Impact low at ripple < 0.5% at separatrix, however, not understood!!
- The developing understanding of ELMs and RWM and their stabilisation (solved / ongoing)
  - In vessel coils foreseen in ITER – physics not understood – risk!
- The definition of a credible steady state scenario (solved / ongoing)
  - Development needs new generation of Tokamaks (EAST, KSTAR, JT60SA, ITER)
The Development of a Radiative Divertor

• A large fraction of the $\alpha$- and additional heating power has to be dissipated inside the divertor – in ITER ~100 MW

• Due to fieldline geometry this power is concentrated in a toroidal band of a few cm width => large peak power flux (can be 40 MW/m$^2$)
  
  – Mitigation achieved by angling targets (limited) and by radiating power in the divertor (increase surface for power deposition) => peak power ~10 MW/m$^2$
  
  – However, radiation is limited due to pressure constancy along fieldlines in normal divertor operation
  
  – Loss of momentum lateral to plasma flux essential to increase radiation => neutral plasma interaction transports momentum laterally => needs low temperature divertor => high density
  
  – New physics was developed during 90$^{th}$ and validated on existing tokamaks => 2 D models used for prediction to ITER

Total fieldline angle >1 degree; => Poloidal Angle ~15 to 20 degree
Peak power flux in D-III-D significantly reduced by radiative divertor operation

Power and He exhaust as well as fuel exhaust within achievable engineering parameter
2 important Confinement Physics issues understood

- Core confinement governed by two transport channels with different turbulence behaviour – dominant regime depends on H-mode pedestal temperature
  - Ion Temperatur Gradient Transport (ITG): critical gradient proportional to T
  - Electron Transport – less stiff, critical gradient seems proportional to \(1 / \sqrt{T}\)
  - => A correct scaling depends on all machines being in the same regime - Ion or Electron dominated – was not the case end of 90s !! => error
- Scaling was corrected 99 => bigger ITER machine – i.e. \(R = 6.0\) to \(6.2\) m
- H-mode confinement regime depends on pedestal temperature
  - Scaling performed at medium density => it missed the degradation at high density – understanding emerged end of 90\(^{th}\) => ITER design changed
  - Pedestal pressure ~ constant for given shaping (triangularity, elongation)
  - Change from large ITER to present machine triangularity increased from 0.2 to 0.35 at q95%, elongation 1.8
- Needed a different CS and TF coil design – impact on divertor !!
- The above understanding allowed an optimisation of the ITER machine with less confinement margin => lower cost
Plasma Elongation and Triangularity — produced by currents in the poloidal field coils

Elongation = \( \frac{b}{a} \):
- Ratio of “b” axis to plasma radius “a”
  - Larger Elongation increases plasma cross section and thus allows larger current for the same global edge \( q \)
  - Larger Elongation increases edge magnetic shear – confinement, \( \beta \) !!

Triangularity = \( \frac{c}{a} \):
- Ratio of distance from plasma center to upper and lower X-points (or turning points) “c” to plasma radius “a”
  - Higher triangularity increases edge magnetic shear – confinement, \( \beta \) !!
ITER Operation Space in H-mode predicted by an Integrated Plasma Model


- One-dimensional modelling of the plasma core:
- Two dimensional modelling of the SOL and Divertor
- Physics based empirical model for the Pedestal
- Turbulence fluid model for the core (Multimode – Lehigh – University - Bethman)

- Multimode a bit optimistic!

Operational and objective limits:

\[ Q = 5, \text{ LH transition, low temperature limit on alpha power, auxiliary power, edge density limit} \]
Recent predictions indicate that uncontrolled ELM heat pulse amplitude in ITER will produce energy densities at the divertor target of ~10MJm$^{-2}$ - an order of magnitude above tolerable level for divertor PFCs:

- techniques for ELM suppression or mitigation essential

J Linke et al, 2007
Two principal approaches are currently under development for ITER:

- edge ergodization by RMP coils – see foreseen design below
- pellet pacemaking
Family of Superconducting Tokamaks in the World needed for development of Steady State Scenarios

- The small Tokamaks are starting to operate shortly or have already started to operate.
- They are all in countries (India, China, Korea) who are practically Fusion newcomers.
- The small machines will allow to push fusion research towards steady state operation and will be essential to prepare the ground for the two large projects.
  - Operation - ITER - 2018 and JT60SA - 2014
- The US is also discussing at the moment to construct a large experiment as a satellite to ITER but no concrete designs have yet emerged.
ITER Technology Challenges / Developments

- The development of high field large superconducting magnets
  - Requirements seemed unfeasible 20 years ago
- The development of Divertor High Heatflux Components
  - A large step in development to achieve 20 MW/m
- The development of Remote Maintenance
  - Thought to be impossible by engineers from nuclear industry
- The development of the DT fuel Cycle
  - A key for ITER and DEMO – a challenge up to today
- The development of Heating and Current Drive systems
  - A challenge for all systems envisaged, not fully solved today
- The development of ITER compatible Diagnostics
  - A step from laboratory type systems to a reliability similar to space
The ITER Design and Technology has been underpinned by R&D

**CENTRAL SOLENOID MODEL COIL**
- Radius 3.5 m
- Height 2.8 m
- $B_{\text{max}} = 13$ T
- $W = 640$ MJ
- $0.6$ T/sec

**VACUUM VESSEL SECTOR**
- Double-Wall, Tolerance $\pm 5$ mm
- HIP Joining Tech
- Size: $1.6$ m x $0.93$ m x $0.35$ m

**REMOTE MAINTENANCE OF DIVERTOR CASSETTE**
- Attachment Tolerance $\pm 2$ mm
- Heat Flux $> 15$ MW/m², CFC/W

**REMOTE MAINTENANCE OF BLANKET**
- 4 t Blanket Sector
- Attachment Tolerance $\pm 0.25$ mm

**TOROIDAL FIELD MODEL COIL**
- Height 4 m
- Width 3 m
- $B_{\text{max}} = 7.8$ T
- $I_{\text{max}} = 80$ kA

**DIVERTOR CASSETTE**
Magnet Development: Current-Field-Chart (Lorentz-Force)

Situation today – gives confidence to be able to built the ITER magnets

Situation at the beginning of the 90th

Superconducting coils from earlier experiments
ITER Model Coil, High Temperatur SC Current Leads

Toska Facility

ITER Model Coil

HTS Current Lead

FZK - Germany
CS Model Coil R&D
Closing of the Test Cryostat (JA)
## Overall Features of ITER Magnets
### 4 Main Systems, all superconducting

<table>
<thead>
<tr>
<th>System</th>
<th>Energy GJ</th>
<th>Peak Field</th>
<th>Total MAT</th>
<th>Cond length km</th>
<th>Total weight t</th>
</tr>
</thead>
<tbody>
<tr>
<td>Toroidal Field (TF)</td>
<td>41</td>
<td>11.8</td>
<td>164</td>
<td>82.2</td>
<td>6540</td>
</tr>
<tr>
<td>Central Solenoid</td>
<td>6.4</td>
<td>13.0</td>
<td>147</td>
<td>35.6</td>
<td>974</td>
</tr>
<tr>
<td>Poloidal Field (PF)</td>
<td>4</td>
<td>6.0</td>
<td>58.2</td>
<td>61.4</td>
<td>2163</td>
</tr>
<tr>
<td>Correction Coils (CC)</td>
<td>-</td>
<td>4.2</td>
<td>3.6</td>
<td>8.2</td>
<td>85</td>
</tr>
</tbody>
</table>
ITER Conductors

- ITER coils are wound from **Cable-In-Conduit Conductors (CICC’s)**, relying on superconducting multifilament composite strands mixed with pure Cu strands/cores.
- The strands are assembled in a multistage rope-type cable around an open central cooling spiral.
- The cable and its spiral are inserted inside a stainless steel conduit which provides helium confinement.
The ITER magnet system is made up of:
- 18 Toroidal Field (TF) Coils,
- a 6-module Central Solenoid (CS),
- 6 Poloidal Field (PF) Coils,
- 9 pairs of Correction Coils (CC).
•The vacuum vessel is lined by modular removable components: blanket modules, divertor cassettes, ELM / VS coils and port plugs (heating antennae, diagnostics and test blanket modules) All these removable components are mechanically attached to the VV.
Plasma Facing Components - Challenges

- CFC divertor targets (~50m²):
  - erosion lifetime (ELMs!) and tritium codeposition
  - dust production
- Be first wall (~700m²):
  - dust production and hydrogen production in off-normal events
  - melting during VDEs
- W-clad divertor elements (~100m²):
  - melt layer loss at ELMs and disruptions
  - W dust production - radiological hazard in by-pass event
**Divertor**

**Divertor system main functions:**

- Exhaust the major part of the plasma thermal power (including alpha power)
- Minimize the helium and impurities content in the plasma

**Challenge to develop HHF Componets capable of 20 MW/m**
Plasma-Facing Components

W monoblock
5 MW/m²

Copper Interlayer
CuCrZr Heat Sink

Smooth Tube

CFC monoblock
10 MW/m²
20 MW/m² 10 sec

CFC monoblock
For first divertor set

Copper Interlayer
CuCrZr Heat Sink

Twisted tape: To increase the margins against the Critical Heat Flux

Vertical Targets

W monoblock

XM-19

316L(N)-IG

XM-19
• **W monoblocks:**
  - 10 MW/m² x 1000 cycles

• **CFC monoblock**
  - 10 MW/m² x 1000 cycles
  - 20 MW/m² x 1000 cycles

2 decades of development to achieve these parameters also after neutron irradiation.
Blanket System

Challenge EM Forces

Scope
- 440 blanket modules at ~4 ton each
- ~40 different blanket modules

Blanket system main functions:
- Exhaust the majority of the fusion power
- Reduce the nuclear responses in the vacuum vessel and superconducting coils
**Blanket RH System**

**Payload**: 4 ton - (max. 4.5 ton at limited location)

**Reach**: 1.3 – 3.8m from the rail center - Rail location: R6.2m from machine center
Full scale 180° rail deployment test (1998)

RH system for blanket has shown its ability to perform the job.
Some issues remain.

2 decades of development to provide demonstration of feasibility and reliability.

In-port rail connection 2008-2009
Divertor RH equipment is comprised of two main types of “cassette mover”:

- Cassette Multi-function Mover (CMM)
- Cassette Toroidal Mover (CTM)

Each are to be equipped with a dexterous manipulator arm and RH tooling.
Apart from the cryogenic guard vacuum – exhausts are centralized and controlled
ITER Vacuum Systems

Cryostat vacuum (<10^-4 Pa) 8500 m³
Torus vacuum (~10^-6 Pa) 1400 m³
Neutral Beam vacuums (~10^-7 Pa) 630 m³ (for 4)
Cryogenic Guard Vacuum
Service Vacuum System (Inc diagnostics)
ICRH and ECRH Vacuums
ITER Torus Cryopump Prototype tested in FZK

Successful development of charcoal coating for He pumping in FZK - Germany

4.5 K Panels
Tritium Plant Building Systems Layout

- 7 Floors
  - 2 below grade
- L = 80 m
- W = 25 m
- H = 35 m
- Release point elevation: 60 m
  - Tokamak building height: 57 m
Tritium Plant R&D in FZK

T-plant systems

TEP – tokamak exhaust processing

ISS- Isotope separation

WDS- Water Detriation

Analytical System

Storage system

All Systems validated by R&D in FZK- Germany
### ITER Heating and Current Drive

**P_{aux} for Q=10 nominal scenario: 40-50MW**

<table>
<thead>
<tr>
<th>Heating System</th>
<th>Stage 1</th>
<th>Possible Upgrade</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>NBI (1MeV –ive ion)</td>
<td>33</td>
<td>16.5</td>
<td>Vertically steerable (z at Rtan -0.42m to +0.16m)</td>
</tr>
<tr>
<td>ECH&amp;CD (170GHz)</td>
<td>20</td>
<td>20</td>
<td>Equatorial and upper port launchers steerable</td>
</tr>
<tr>
<td>ICH&amp;CD (40-55MHz)</td>
<td>20</td>
<td></td>
<td>2Ω_{T} (50% power to ions (\Omega_{He3} (70% power to ions, FWCD)</td>
</tr>
<tr>
<td>LHH&amp;CD (5GHz)</td>
<td>20</td>
<td></td>
<td>1.8&lt;n_{par}&lt;2.2</td>
</tr>
<tr>
<td>Total</td>
<td>73</td>
<td>130 (110 simultan)</td>
<td>Upgrade in different RF combinations possible</td>
</tr>
<tr>
<td>ECRH Startup</td>
<td>2</td>
<td></td>
<td>126 or 170GHz</td>
</tr>
<tr>
<td>Diagnostic Beam</td>
<td>&gt;2</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
NBI: 1 MeV beams – 16 MW power deposited in the plasma by each
Development is a challenge – test bed under construction in Padua
first heating and the diagnostic beam installed in 2020 second in 2022
IC H&CD system

Challenge is antenna design and coupling to the plasma - TBD

- Improved and optimized IC system with 2 antennas allows to secure the coupling of 20 MW in all envisaged scenarios (large scrape off, short decay length, 40 to 55 MHz, dipole phasing) with 40 kV max in the circuits
- Efficient resilient matching limit ELM effect: reflected power is kept below 1% of forward power
ECRH - Progress on Gyrotrons

2 MW gyrotrons from EU, 1 MW from JA and RF.
• Diode type gun gyrotrons from EU and RF, triode type from JA (requiring an additional PS for the anode).

Challenge is the development of the sources – good progress
And the development of the lounchers – very good progress
• About 40 large scale diagnostic systems are foreseen:
  • Diagnostics required for protection, control and physics studies
  • Measurements from DC to γ-rays, neutrons, α-particles, plasma species
  • Diagnostic Neutral Beam for active spectroscopy (CXRS, MSE ....)
ITER - Status

• The ITER Team has been established on the Cadarache site for ~2 years - now ~280 team staff on site

• ITER Organization formally established in October 2007

• Design Review carried out to revise Baseline by mid-2008

• On 31 January 2008, the files for DAC (Demande d’Autorisation de Création), including the Preliminary Safety Report, in application of the TSN law, completed and sent to the French Nuclear Authorities => construction license

• The main platform levelling works have finished on the ITER site
### ITER Construction – Updated Schedule

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>ITER Construction</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Tokamak Basic Machine</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Issue TF Coils PAs</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1st TF Coil at Site</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Last TF Coil at Site</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Issue PF Coils PAs</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1st PF Coil at Site</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Last PF Coil at Site</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Issue VV PAs</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1st VV Sector at Site</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Last VV Sector at Site</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Buildings &amp; Site</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Site Leveling</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Tendering process</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Tokamak Complex Excavations</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Tokamak Building Construction</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Tokamak Assembly</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Start Assemble VV</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Tokamak Basic Machine Assembly</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Tokamak Basic Machine Assemble</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Start Install CS</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Start Cryostat Closure</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ex Vessel Assembly</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>In Vessel Assembly</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pump Down &amp; Integrated Commissioning</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

*“Roughly” 2018 First Plasma Minimal internal vessel components*
- First D-T Plasma foreseen at the end of 2026 or beginning of 2027
- Effective use of Shutdown: Phase 2 assembly, W-divertor exchange

Updated Schedule: Experimental Schedule

<table>
<thead>
<tr>
<th>ITER Commissioning and Operations</th>
</tr>
</thead>
<tbody>
<tr>
<td>---</td>
</tr>
<tr>
<td>Start Torus Pump Down</td>
</tr>
<tr>
<td></td>
</tr>
</tbody>
</table>
Main Technology Developments for DEMO

High Temperature Super Conducting Coils (40 to 77 K)

He cooled Breeding Blanket and T-extraction

Low Activation Structural Material for the “In Vessel” Components which can withstand the large neutron fluence (150 dpa end of life)
The Physics and Technology Development of the last two decades was very impressive and made the realisation of ITER possible (was not the case beginning of the 90th).

Based only on the physics and technology advances we could have started to construct ITER ~5 to 8 years ago.

However, to set up an international project like ITER takes time and also to establish the teams to built it.

ITER construction is now well on its way with many procurements being started even if in some cases remaining design issues have to be tackled.

Please see for further information on ITER the talks of S. Chiocchio and G. Johnson as well as talks and posters from many other colleagues from ITER and the DAs.