The ITER machine, construction status and the plasma-wall interaction challenge

R. A. Pitts
ITER Organization, Science and Operations Department, Cadarache, France

The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.
What is in this talk

• DT fusion and the tokamak principle
• The ITER mission goal
• Major machine components and systems
  ▪ A challenge for science and industry - very brief fly through here – apologies in advance!
• The plasma-wall interaction challenge
  ▪ Huge area, can only cover a small part
  ▪ Choose to emphasize transient heat loads and long term materials response → a major issue for future reactors which ITER will have to address
Deuterium-tritium fusion

Reactions between heavier H isotopes are much faster than the protonic fusion at work in the stars.

Deuterium-Tritium reaction has highest cross-section at the lowest particle energy and is the best candidate for a terrestrial fusion power plant.

\[ D + T \rightarrow ^4\text{He} + n \]

Potential show stopper in tritium supply \( \rightarrow \) see later.

Energy (keV)

Reactivity (m^3s^-1)
The tokamak principle

Inject gaseous fuel into a toroidal, high vacuum chamber with a strong toroidal magnetic field (ITER will have $B_T = 5.3$ T)

Induce a toroidal electric field through transformer action, avalanche ionization produces plasma current and poloidal magnetic field

Toroidal plasma current provides initial ohmic heating
Importance of size

The “energy confinement time”, \( \tau_E \), depends very strongly on the size of the tokamak and the plasma current:

\[
\tau_E \propto I_p \times R^2
\]

The first tokamaks had:
R = 0.6 m, \( I_p = 40 \text{ kA} \)

ITER will have:
R = 6.3 m, \( I_p = 15 \text{ MA} \)

ITER needs \( \tau_E \sim 3.5 \text{ s} \) to reach its plasma burn condition.
And the temperature

To reach temperatures necessary for fusion the heating provided by the plasma current is insufficient → additional power must be added using large auxiliary heating systems.

ITER will have about 70 MW of additional heating power.
And the confinement mode

- “Type I ELMing H-mode”: robust mode of operation in today’s tokamaks → the baseline for ITER at high fusion energy gain
  - H-mode (Type-I ELM) confinement is ~2× better than L-mode

\[ \tau_E \propto I_p R^2 P^{-2/3} \]

IPB98(y,2) Type-I ELMing H-mode scaling FROM
EXPERIMENT

But …

A problem with the "ELMs" (MHD modes localized at the plasma edge – see later)
Fusion energy gain

\[ Q \equiv \frac{P_{fusion}}{P_{heat}} = \frac{5P_\alpha}{P_{heat}} \]

\( \alpha \)-heating fraction

\[ f_\alpha \equiv \frac{P_\alpha}{P_\alpha + P_{heat}} = \frac{Q}{Q + 5} \]

Scientific breakeven

<table>
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<th>( f_\alpha )</th>
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<td>20</td>
<td>80%</td>
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Burning plasma regime

The JET and TFTR tokamaks have produced DT fusion powers >10MW for \( \sim 1s \) in the 1990’s (Q \( \sim 0.5 \))

ITER is designed to a scale which should yield \( Q \geq 10 \) at a fusion power of \( 400 \) – \( 500 \) MW for \( 300 \) – \( 500 \) s

Alternative scenarios also planned at lower Q but longer pulse (up to 1 hour)
Also a technology tester

ITER should:

- Demonstrate integrated operation of technologies for a fusion power plant
- Test components required for a fusion power plant
- Test concepts for a tritium breeding module
In a fusion reactor will have to be bred via:
$n + Li \rightarrow T + ^4He$

Charged particle $\rightarrow$ used to transfer energy to the fuel and keep the plasma hot
A word on tritium supply

Half-life of T: 21.32 years
Decays ~5.5% / year
Vast majority of production is waste product of CANDU fission reactors
Production is 27 kg T from over 40 years exploitation
A 2GW (500 MWe) DEMO fusion will burn ~110 kg T per full power year

If ITER succeeds (if it achieves the projected $P_{\text{FUS}}$ and/or neutron fluence), it will require ~17 kg T in lifetime.

→ Realistically, DEMOs after ITER have to be tritium breeders
The ITER machine
(a brief tour of some key components)
ITER: main components

- **Toroidal Field Coil**: \( \text{Nb}_3\text{Sn}, 18, \text{wedged} \)
- **Central Solenoid**: \( \text{Nb}_3\text{Sn}, 6 \) modules
- **Poloidal Field Coil**: \( \text{NbTi}, 6 \)
- **Correction Coils**: \( \text{NbTi}, 18 \)
- **Feeders**: \( \text{NbTi}, 31 \)
- **Cryostat**: 24 m high x 28 m dia.
- **Vacuum Vessel**: 9 sectors
- **Blanket**: 440 modules
- **Port Plug**: heating/current drive, test blankets limiters/RH diagnostics
- **Torus**: Cryopumps, 54 cassettes
- **Divertor**:
- **Cryostat Thermal shield**:
- **Machine mass**: 23,350 tonnes (cryostat + VV + magnets)
  - shielding, divertor and manifolds: 7945 t + 1060 t port plugs
  - magnet systems: 10150 t; cryostat: 820 t
ITER construction sharing

A unique feature of ITER is that almost all (~90%) of the machine will be constructed through in-kind procurement from the Parties.

All intellectual property shared by the seven members.

Total intrinsic value = 4585 kIUA (7826 M€ at kIUA 2015 rate)

Real commercial value maybe higher ……..

Overall sharing:
EU: 5/11
Other 6 Members: 1/11 each
Procurement sharing of major components

- **Feeders**: China
- **Toroidal Field Coils**: Europe, Japan
- **Poloidal Field Coils**: Europe, Russia
- **Correction Coils**: China
- **Central Solenoid**: USA
- **Divertor**: China, Europe
- **Blanket**: China, Europe
- **Vacuum Vessel**: Europe
- **Thermal Shield**: India
- **Cryostat**: Korea
Toroidal field conductor procurement

- Facts
  - Procurement underway since 2008
    ~100,000 km / 500 tonnes of Nb\textsubscript{3}Sn strand for TF and CS magnets
  - Pre-ITER world production was
    ~15 t /year $\rightarrow$ 100 t /year for ITER
  - 11 suppliers from 6 Partners
  - 200 km of cable in conduit, 2800 t
  - 70% of conductor unit lengths now accepted by IO
  - Estimated total value ~€600 million
Toroidal field coil radial plates

- The radial plates that hold the conductor of the toroidal field coils are very complex, D-shaped stainless steel structures with grooves machined on both sides along a spiral trajectory.
Vacuum Vessel

Double-walled, stainless steel structure
- 19.4m outer diameter, 11.3 m height, SS 316 L(N)-IG, 5300 t
- Primary tritium containment barrier, bakeable to 200°C
- Must withstand enormous vertical forces during disruptions

Nominal base pressure
\(~5 \times 10^{-8} \text{ mbar}\)
Vacuum vessel procurement

7 sectors of VV
Total cost: 92.06 kIUA (39%)

18 upper ports
Total cost: 20.86 kIUA (9%)

2 sectors of VV
17 Eq. & 9 lower ports
Total cost: 84.06 kIUA (36%)

Inner wall shield/ribs
Total cost: 37.30 kIUA (16%)

234.28 kIUA: 8% of total in kind cost of the ITER machine
ITER Cryostat: vacuum insulation for SC coils

Outer thermal shield

- Diameter: 29.4 m
- Height: ~29 m
- Mass: ~3500 t
- Base pressure <10^{-4} mbar
- SS, 40 – 180 mm thick

IN-DA will manufacture

- Contract signed with L&T Ltd on 17 August 2012
- Will be dispatched in 54 modules to ITER
Cryostat base almost complete

In the Larsen & Toubro factory in India, six 60° sections of the cryostat base are temporarily assembled on the shop floor to verify tolerance.
Principal plasma-facing components (PFC)

Blanket/first wall $\rightarrow$ heat exhaust, impurity management, nuclear shielding

Divertor $\rightarrow$ particle and heat exhaust
Blanket/First Wall

Neutron shielding:
Semi-permanent, full water-cooled massive Shield Block (SB): \( \sim 3.5 \pm 0.5 \) t
Plasma-facing surfaces: separable shaped First Wall Panel (FWP), armoured with Be tiles, water-cooled Rated to 2 MWm\(^{-2}\) and 4.7 MWm\(^{-2}\)

Total number of BMs: **440**
Total mass: \( \sim 1800 \) tonnes

SB:
CN (50%), KO (50%)

FWP:
CN (10%), EU (50%), RF (40%)
The tungsten divertor

Deep, vertical target divertor

54 divertor assemblies

~500 tons total mass

~150 m² W surface

4320 actively cooled heat flux elements

Bakeable to 350°C
The tungsten divertor

Divertor will use state-of-the-art tungsten monoblock technology to handle steady state power loads of $\sim 10 \text{ MWm}^{-2}$

- Total of $\sim 290,000$ monoblocks
Plasma heating systems

- High energy (1 MeV) neutral beams, RF heating tuned to plasma cyclotron frequencies (40-55 MHz ions, 170 GHz electrons)

RF systems follow a common design approach:

- Port plug housing wave launcher
- Transmission lines link port plugs to power tubes outside the torus hall

Neutral beam injectors in NB cell (33 MW)
Test Blanket Modules (TBM)

- Tritium fuel cycle is a **major** challenge for all DT fusion devices → ITER will test concepts
  - 6 modules with different designs, all ITER parties involved

\[
\begin{align*}
\text{n + }^6\text{Li} & \rightarrow \text{T} + ^4\text{He} + 4.8 \text{ MeV} \\
\text{n + }^7\text{Li} & \rightarrow \text{T} + ^4\text{He} + \text{n} - 2.5 \text{ MeV}
\end{align*}
\]
Remote handling

- Extremely challenging to repair and replace complex and heavy components in a nuclear environment
  - Dedicated, state-of-the-art systems for both Blanket, Divertor, Port Plug removal and other in-vessel functions (e.g. inspection, dust aspiration)

On-rail blanket module transporter procured by Japan
Remote handling

- Extremely challenging to repair and replace complex and heavy components in a nuclear environment
  - Dedicated, state-of-the-art systems for both Blanket, Divertor, Port Plug removal and other in-vessel functions (e.g. inspection, dust aspiration)
Plasma control/physics = measurements

- About 60 diagnostic systems (= very well diagnosed)
  - For machine protection, control and physics studies
  - Nuclear nature of device forces tight packing of systems in shared port plugs

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![Diagram of ITER tokamak with diagnostic systems](image-url)

- **UPPER PORT** (12 used)
- **EQUATORIAL PORT** (6 used)
- **DIVERTOR PORT** (6 used)
- **VESSEL WALL** (distributed Systems)
- **DIVERTOR CASSETTES** (16 used)
Progress on the ITER tokamak complex
Arial (balloon) view of the site

- PF winding building
- Assembly Building
- Cryostat Building (IN)
- Tokamak pit and hot cell complex
- 6,000 m² storage area for ITER components

August 2015
Tokamak pit and assembly building

April 2015
Assembly building: raising the roof

Sept. 11 2015
Assembly building: raising the roof
First concrete pour for bioshield

Oct. 21 2015
Challenges in plasma-wall interaction (PWI) (emphasis on plasma transients)
Plasma purity essential

Fusion grade (burning) plasmas have to be **VERY PURE**

Fusion power $\propto (\text{fuel ion density})^2$

Power loss from plasma by radiation $\propto \text{impurity ion density} (n_Z)$ and a strong function of atomic number ($Z$)

$\rightarrow$ avoid contamination by all particles other than fuel (minimize fuel dilution)

This is the challenge of plasma-wall interaction: build a box for a very aggressive, nuclear plasma to sit in, but make sure the box lasts long enough
Plasma purity essential

Fusion grade (burning) plasmas have to be **VERY PURE**

Fusion power $\propto (\text{fuel ion density})^2$

Power loss from plasma by radiation $\propto \text{impurity ion density} (n_z)$ and a strong function of atomic number (Z)

But note that ITER will not produce sufficient neutron fluence for fusion relevant neutron exposure $\rightarrow$ PWI mostly concerned by plasma particle and thermal loads
ITER plasma-facing materials

Blanket first wall

- ~700 m² beryllium
  - Low Z – good plasma compatibility
  - Good oxygen getter
  - Good thermal conductivity

Beryllium

- ~150 m² tungsten
  - Low sputtering yield, high threshold
  - Highest melting point
  - Low fuel retention

Tungsten

Divertor
ITER plasma-facing materials

Blanket first wall

• ~700 m² beryllium

Wall designed to be replaceable → eventually move to tungsten for more reactor relevance

• ~150 m² tungsten

- Low sputtering yield, high threshold
- Highest melting point
- Low fuel retention

Beryllium

Tungsten

Divertor
Still much too high for a reactor → must switch to all high Z (or liquid metal…..)

- Confirms that tritium inventory compatible with projected ITER operation → we also understand this from plasma modelling
The wall matters!

**BUT ......**

- Large reduction (20-30%) in confinement at high plasma current
  - Still not fully understood
  - Would lead to corresponding reduction in ITER fusion energy gain

I. Nunes et al., IAEA FEC 2014
ITER burning plasma power balance

- Worst case for steady state power handling: $Q_{DT} = 10$, H-mode
- $P_{SOL} = P_{IN} + P_\alpha - P_{RAD} \approx 120$ MW → this is exhaust power

In a 2 GW DEMO, the power flowing out of the edge plasma would be 4-5x higher
• Peak divertor power flux densities: \( q_\perp = 10\ \text{MWm}^{-2} \rightarrow \text{requires divertor detachment to make it work (talk tomorrow at YPI)} \)

• Operation at elevated base temperatures (~1200°C) \( \rightarrow \text{close to tungsten recrystallization} \)
ITER burning plasma particle fluence

- Peak divertor particle flux densities: $\Gamma \sim 10^{24} \text{ m}^2\text{s}^{-1}$
- Orders of magnitude step in divertor particle fluence from today to ITER $\Rightarrow$ Influence on materials?
Transients a big issue at the ITER scale

- Plasma stored energy $\propto R^5$
- Energy deposition area $\propto R$
  - Energy transients with rise times $0.5 - 2$ ms
  - Surface temperature rise due to transient:
    $\Delta T_{surf} \propto \frac{E_{trans}}{(A_{wet} t^{1/2})}$
    (from this comes the frequently referred to “Heat flux factor”, e.g. $F_{HF} \sim 50 \text{ MJm}^{-2}\text{s}^{-1/2}$ to melt W)

Transients can easily drive melting of metallic plasma-facing components on ITER
ELM power loads

- High current (3 MA) JET Type I ELMing H-mode
• This plasma is sitting about 2x above the L-H-mode power transition threshold (like ITER)

Something needs to be done to mitigate these instabilities on ITER (even more so for a reactor)

- $t_{\text{rise}} \sim 200 \, \mu s$, $E \sim 0.2 \, \text{MJm}^{-2}$ on divertor target

On ITER $\Delta W_{\text{ELM}} \sim 20 \, \text{MJ at 1-2 Hz, } \tau_{\text{rise}} \sim 350 \, \mu s$!
Tolerable ELMs

- Original ELM energy loss spec for ITER derived from Russian plasma gun experiments on melting of W assuming no misalignments (edges) between divertor elements
  - Fixed at 0.5 MJm$^{-2}$ → translates to $\Delta W_{\text{ELM}} \sim 1.0$ MJ
- Mitigation factors of ~20 required for ITER
- Story might not be so clear cut → the recent experimental finding that $q_{||,\text{peak}} \propto p_{\text{ped}}$ may reduce the required mitigation

A. Zhitlukhin et al., J. Nucl. Mat. 363-365 (2007) 301
In-vessel coils

- ITER will be equipped with a set of 27 “ELM control coils”
  - 3 per vacuum vessel sector

- Physics understanding still incomplete, but best coil set we can design and integrate based on today’s knowledge
Long term ELM effects (1)

- Problem is that even if ELMs can be mitigated, the frequent thermal cycling may anyway lead to damage formation over time.

Damage threshold $\leq 3 \text{ MJm}^{-2}s^{-1/2}$

For $10^6$ square wave pulses at $\sim 500 \mu$s duration, $T_{\text{surf}} = 1200^\circ\text{C}$ ($W_{\text{melt}} \sim 50 \text{ MJm}^2\text{s}^{-1/2}$)

$10^6$ ELMs $\sim 4$ hours ITER plasma time

About 2.5 days ITER operation in target regime
Long term ELM effects (2)

- Tungsten surface can be strongly modified if conditions are not carefully controlled even for sub-melting threshold events

10⁵ ELMs ≡ 24 mins exposure time at \( f_{\text{ELM}} = 70 \text{ Hz} \) on ITER...

Transients: plasma disruptions

- There will many fewer disruptions than ELMs on ITER, but damage can be just as bad

  80 - 320 MJm\(^{-2}\)s\(^{-1/2}\)

  130 - 280 MJm\(^{-2}\)s\(^{-1/2}\)

  up to 770 MJm\(^{-2}\)s\(^{-1/2}\)

  Major disruption
  350 MJ (worst case)

- Loss of thermal, magnetic energy, runaway electrons

- Can potentially melt up to several kg per disruption (large scale shallow melt)

- Runaway electrons: highly localized, deep deposition (e.g. 10 MJ, \(I_{RE} = 5\ MA, <E_{RE}> = 15\ MeV\))

Disruptions also a safety issue → much effort going into design of disruption mitigation systems
Transients: disruptions

- Clear already from JET with $W_{\text{plasma}} < 10$ MJ that Be melting will be an issue for ITER

Accumulated shallow melt, upward VDE

Inner wall deep runaway electron melt

G. F. Matthews et al., PFMC 2015
Transients: disruptions

- For a tungsten divertor, vapour shielding may help
  - But complex 2D, time dependent process

- Factor 5-10 reduction in heat flux with shielding
- Need experimental benchmark (plasma guns)
Effects of W exposure to helium

- He might be an issue for W (independent of ELMs)
  - He insoluble in W but can self-trap: He-W repulsion
  - Clustering of He to form bubbles → material swells

~ 600 - 700 K

~ 900 – 1900 K

Nano-bubbles

‘Fuzz’

M. Miyamoto et al, JNM, 415 (2011)
G. De Temmerman et al, JVSTA 30 4 (2012)
Effects of W exposure to helium

- He might be an issue for W (independent of ELMs)
  - He insoluble in W but can self-trap: He-W repulsion
  - Clustering of He to form bubbles → material swells

Nano-bubbles

~ 600 - 700 K

Large voids

> 2000 K

M. Miyamoto et al, JNM, 415 (2011)
G. De Temmerman et al, JVSTA 30 4 (2012)
Should we be worried for ITER?

- **Conditions for fuzz formation**
  - $T_{\text{surf}} > 700^\circ\text{C}$
  - $E_{\text{ion}} > 20$ eV

- **Would not expect it to form under “reference” burning plasma conditions in ITER**
  - But e.g. ELMs will bring high $E_{\text{ion}}$ for short times

- **If it forms, will it represent a macroscopic erosion pathway?**
  - Still unclear (author’s opinion)
Bubbles might be an issue

- They grow very rapidly \(\rightarrow\) a few sec of plasma exposure
  - Strong decrease of the near-surface thermal conductivity has been observed for tungsten surfaces exposed at 600 K

**Thermal conductivity**

- Long term effect on material thermo-mechanical properties?
- Being actively researched

R.P. Doerner et al, 19\textsuperscript{th} SOL Div ITPA (2014)
G. De Temmerman et al, 20\textsuperscript{th} PSI conf. (2012)
Clear influence on thermal shock

- Simultaneous stationary/transient exposures in Pilot-PSI in pure H and pure He under similar conditions

He, 500°C  H, 400°C  He, 800°C  H, 800°C

\(E_{\text{ion}} = 50 \text{ eV} \)
\(E_{\text{trans}} \sim 0.25-0.3 \text{ MJ.m}^{-2} \)
Fluence: 1-2x10^{27} \text{ m}^{-2}
600 pulses (He)
1000 pulses (H)

- Cracking observed for both H and He at low \(T_{\text{surf}}\)
- Roughening only for H at 800°C
- Cracking under He even at high \(T_{\text{surf}}\) \(\rightarrow\) embrittlement

L. Buzi et al., to be submitted
Summary

• ITER is a conventional tokamak but built on a scale never previously attempted
• ITER is now well into the construction phase
• H-mode operational baseline for $Q_{DT} = 10$ is well researched but ELM control still a challenge
• Plasma-wall interaction is a key issue for ITER and all future reactors
Thank you
Reserve material
Protonic fusion chain

4 protons combine to produce a $^4\text{He}$ nucleus, some $\gamma$ radiation and 2 neutrinos

Gravitational compression produces high temperature plasma ($\sim$15 million°C) in the Sun’s core needed for fusion to occur

Mass defect = $4.8 \times 10^{-29}$ kg $\rightarrow$ 1 kg of H generates $6.4 \times 10^{14}$ J

Sun burns hydrogen at rate of 600 million tons/s!
Not much good on Earth however ... 

On average, hundreds of millions of years required for 2 protons to fuse

If it did not, we would not have been around long enough to work it out ....

Takes about 1 million years for the energy to get out from the core

Not a very useful process for terrestrial fusion ....
Fusion performance

Important figure of merit is the “triple product”: $nT\tau_E$

Ignition criterion:
$nT\tau_E \geq 3 \times 10^{21} \text{ keV s m}^{-3}$

(minimum value of triple product required for plasma self-heating)

DT fuel has lowest $nT\tau_E$ parameter $\rightarrow$ optimum plasma temperature is $\sim 15$ keV (about 175 million°C)
Fusion performance

Important figure of merit is the “triple product”: \( nT\tau_E \)

Ignition criterion:
\[
nT\tau_E \geq 3 \times 10^{21} \text{ keV s m}^{-3}
\]
(minimum value of triple product required for plasma self-heating)

ITER will not ignite but will be the first tokamak which can simultaneously reach \( n, T, \tau_E \) values sufficient for high fusion gain \( \rightarrow \) plasma dominantly heated by \( \alpha \)-particles

A few DT expts. already in tokamaks
The ITER machine

$R \sim 6 \text{ m}$

$h \sim 29 \text{ m}$
ITER history

Idea originates from Geneva superpower summit on 21 Nov. 1985: Gorbachev and Reagan signed an agreement to set up an International Project to develop fusion energy.
ITER history

2006: ITER Joint worksite established in Cadarache

ITER Joint Implementing Agreement signed at a ceremony at the Elysée Palace in Paris hosted by French President and the EU Commission President


7 Parties representing more than half the world population
ITER: the biggest in a long line

ITER builds on nearly 50 years of worldwide tokamak research on more than 150 devices.

From the first, T1 in Moscow, ~1957

To the largest currently operating, JET, which began operation in 1978 and is still going.

And a whole host of superconducting devices (like ITER) operating or planned in Asia/India:

- EAST, China
- KSTAR, S. Korea
- SST-1, India
- JT60-SA, Japan
Although it is reactor scale, ITER is primarily an experiment and aims to demonstrate the scientific and technological feasibility of fusion power.
The tokamak principle

Inject gaseous fuel into a toroidal, high vacuum chamber with a strong toroidal magnetic field (ITER will have $B_T = 5.3$ T)

Induce a toroidal electric field through transformer action, avalanche ionization produces plasma current and poloidal magnetic field

Combination of toroidal and poloidal fields produces helical field for plasma confinement
Superconducting magnet systems

Facts
- 48 superconducting coils (~9800 tons)
- 11.8 T (peak TF field)
- 68 kA (peak current)
- Stored energy – 51 GJ

Central Solenoid (CS) Coils – Stack of 6 (US)

6 PF Coils (EU & RF)

9 Pairs of Correction Coils (CN)

18 TF Coils (EU & JP)

31 Feeders (CN)
The radial plates that hold the conductor of the toroidal field coils are very complex, D-shaped stainless steel structures with grooves machined on both sides along a spiral trajectory.
Plasma heating systems

ITER’s heating systems are not new technology, but are on a totally unprecedented scale.

One of the two negative ion neutral beam injectors
Main inner heat shield

- Provides barrier for thermal loads from warm components to the superconducting coils (4.5K)
- Operates at 80 K (gaseous He in cooling pipes)
- Stainless steel panels are silver coated to reduce emissivity
- Mass: ~1000 t
ITER Cryostat

Transfers loads to tokamak floor

Top lid
Upper cylinder
Lower cylinder
Base section
Installation of neutral beam duct liners

- Tooling likely to be supplied by IO
Tight integration in each port plug

Diagnostics sit in special drawers in each plug (diagnostic shield modules)
Diagnostic first wall

All diagnostics requiring views into the plasma sit behind a stainless steel ‘Diagnostic First Wall’ → mandatory for neutron screening.

Unlike any tokamak before ITER
Assembly strategy built-in from start
Assembly strategy built-in from start
Characteristics of fusion energy

High energy density

Practically inexhaustible fuel

- D is 1/6500 of H
- Li is in minerals and brines and in the ocean

Environmental

- No CO$_2$ emission
- No high-level radioactive wastes

No risk of nuclear accidents

- Only ~1 g of fuel in the reactor at any time
- No chain reactions

No generation of weapons material

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<th>Specific energy (MJ/kg)</th>
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<td>Coal</td>
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<td>Gasoline</td>
<td>50</td>
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<td>Fission (U-235)</td>
<td>85’000’000</td>
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<td>Fusion (D-T)</td>
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<tr>
<td>$E = mc^2$</td>
<td>90’000’000’000</td>
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Availability of fusion fuels

Limited in fact by Li reserves

Deuterium in water

• 156 ppm (1/6400) in naturally occurring H
• Extracted by electrolysis of D₂O obtained via Girdler sulfide (GS) isotopic exchange process

Lithium extracted from brine or minerals

• $^6$Li 7.5% of naturally occurring Lithium
• Terrestrial reserves ~20-30Mt
• Main consumers: glass and ceramic industry, lithium-ion batteries + Li needed for electric cars expected to dominate market (even over fusion)
• 2008 $^6$Li reserves could meet fusion consumption for $\sim 10^3$ years (see [A.M. Bradshaw et al., Fus. Eng. Design] for assumptions) + Requires recycling of Li and separation of isotopes
• Longer term supply has to be based on Li in seawater (0.17 g/t, i.e. 226,000 Mt) $\rightarrow \sim 10^7$ years
Fusion attractive based on power to mass ratio

Take the lithium from the battery of a single laptop computer, add half a bathtub of water, and it can give you 200,000 kWh of electricity.

Enough to power one person in the EU for 30 years, including his share of industrial electricity.
Magnetic confinement in toroidal devices

**Problem:** in a toroidal field alone, particle drifts lead to charge separation

**Solution:** torus solves the end-loss problem

**Problem:** hoop stress from unequal magnetic and kinetic pressures

**Solution:** add poloidal field → particles sample regions of inward and outward drift

**Solution:** add vertical field to counteract hoop stress
Radioactivity levels

Years After Shutdown

Curies/Watt (Thermal Power)

Nuclear Fission Light Water Reactor

Fusion Vanadium Alloys

Fusion Ferritic Steel

Level of Coal Ash

Fusion Silicon Carbide Composites
Rebar for pouring of B2 slab

Tokamak, bioshield and cryostat ~23,000 tons

B2 slab supports the 360,000 ton Tokamak Complex.

June 2013
Similar problems for Be on first wall

- Damage threshold (roughening) at $F_{HF} \leq 6 \text{ MJm}^{-2}\text{s}^{-1/2}$
  @ $T_{surf} = 300^\circ\text{C}$ (cf. Be melt $\sim 28 \text{ MJm}^{-2}\text{s}^{-1/2}$)

- Corresponds to an extremely low $\Delta W_{ELM} \sim 0.25 \text{ MJ}$!
  $\rightarrow$ no “damage” means essentially complete ELM suppression


JUDITH 1 e-beam, 100 pulses, 1 ms

- Melting threshold
- Cracking threshold
- Damage threshold
- No damage
- Roughening
- Cr. n. + melting
- Small cracks
- Crack network

B. Spilker et al., PFMC 2015