Magnetic Fusion Power Plants --
Tritium Systems and Requirements

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Conceptual Design Studies of Magnetic Fusion Power Plants
**Mission:** Perform integrated design studies of the long-term fusion energy embodiments to identify key R&D directions and provide visions for the program.

- Knowledge base of fusion power plants involves subtle combinations of and trade-off among plasma physics, fusion nuclear sciences, and engineering. We simulate conditions that are not encountered in present experiments.
- Commercial fusion energy is toughest standard to judge the usefulness of program elements.

**National ARIES Team** comprises key members from major fusion centers (universities, national laboratories, and industry).
Framework: Assessment Based on Attractiveness & Feasibility

Periodic Input from Energy Industry → Goals and Requirements

Scientific & Technical Achievements

Projections and Design Options

Evaluation Based on Customer Attributes

No: Redesign

Attractiveness

Characterization of Critical Issues

Yes

Feasibility

R&D Needs and Development Plan

Balanced Assessment of Attractiveness & Feasibility
Top-Level Requirements for Fusion Power Plants Were Developed in Consultation with US Industry

- Have an economically competitive life-cycle cost of electricity
- Gain Public acceptance by having excellent safety and environmental characteristics
  - No disturbance of public’s day-to-day activities
  - No local or global atmospheric impact
  - No need for evacuation plan
  - No high-level waste
  - Ease of licensing
- Reliable, available, and stable as an electrical power source
  - Have operational reliability and high availability
  - Closed, on-site fuel cycle
  - High fuel availability
  - Capable of partial load operation
  - Available in a range of unit sizes
Framework: Assessment Based on Attractiveness & Feasibility

- Periodic Input from Energy Industry
- Goals and Requirements
- Scientific & Technical Achievements

- Projections and Design Options
  - Evaluation Based on Customer Attributes: Attractiveness
  - Characterization of Critical Issues: Feasibility

- Balanced Assessment of Attractiveness & Feasibility

- R&D Needs and Development Plan

- Yes
- No: Redesign
Detailed analyses are necessary to understand trade-offs.

Plasma analysis

Engineering Design

System Analysis and Trade-offs

Self-consistent point design for a fusion power plant
Detailed analyses are necessary to understand trade-offs

Plasma analysis
- High accuracy equilibria;
- Large ideal MHD database over profiles, shape and aspect ratio;
- RWM stable with wall/rotation or wall/feedback control;
- NTM stable with LHCD;
- Bootstrap current consistency using advanced bootstrap models;
- External current drive;
- Vertically stable and controllable with modest power (reactive);
- Rough kinetic profile consistency with RS/ITB experiments, as well GLF23 transport code;
- Modest core radiation with radiative SOL/divertor;
- Accessible fueling;
- No ripple losses;
- 0-D consistent startup;

Engineering Design
- Superconducting magnet design
- First wall/blanket, and shield, Divertor;
- Current-drive systems (Launchers, transmission lines, sources),...
  - Configuration
  - Neutronics & Shielding
  - Thermo-fluid & thermo mechanical design
  - MHD effects
  - Tritium Breeding & management
  - Erosion
  - Off-normal events
  - Inventory

- Waste Disposal
- Safety Analysis
- Maintenance
Nature of Power Plant Studies has evolved in time.

- **Concept Exploration (< 1990)**
  - Limited physics/engineering trade-offs due to limited plasma physics understanding.
  - The only credible vision was a large, expensive pulsed tokamak with many engineering challenges (e.g., thermal energy storage).

- **Concept Definition (~ 1990-2005)**
  - Finding credible embodiments (Credible in a “global” sense).
  - Better physics understanding allowed optimization of steady-state plasma operation and physics/engineering trade-offs.

- **Concept Feasibility and Optimization (> 2010)**
  - Detailed analysis of subsystems to resolve feasibility issues.
  - Trade-offs among extrapolation and attractiveness.
ARIES research has examined many concepts

- ARIES-I first-stability tokamak (1990)
- ARIES-III D$_3$He-fueled tokamak (1991)
- ARIES-II and -IV second-stability tokamaks (1992)
- Pulsar pulsed-plasma tokamak (1993)
- SPPS stellarator (1994)
- Starlite study (1995) (goals & technical requirements for power plants & Demo)
- ARIES-RS reversed-shear tokamak (1996)
- ARIES-ST spherical torus (1999)
- Fusion neutron source study (2000)
- ARIES-AT advanced technology and advanced tokamak (2002)
- ARIES-CS Compact Stellarator Study (2008)
- ARIES Pathways
- ARIES-ACT: Detailed studies of in-vessel components and off-normal events (Current Study)
Continuity of ARIES Research Has Led to the Progressive Refinement of Plasma Optimization

ARIES-I (first-stability steady-state):
• Trade-off of $\beta$ with bootstrap
• High-field magnets to compensate for low $\beta$

ARIES-II/IV (2nd Stability):
• High $\beta$ but with too much bootstrap
• Self-consistent plasma only marginally better

ARIES-RS (reverse shear):
• Improvement in $\beta$ and current-drive power
• Approaching COE insensitive of power density

ARIES-AT (Advanced technology):
• High efficiency blanket reduces fusion power and power handling systems, e.g. high $\beta$ is used to reduce toroidal field

Need high $\beta$ equilibrium with high bootstrap
Need high $\beta$ equilibrium with aligned bootstrap
More detailed plasma physics

Improved plasma performance
A range of plasma options is available for tokamak power plants

<table>
<thead>
<tr>
<th></th>
<th>1st Stability, Nb₃Sn Tech.</th>
<th>High-Field Option</th>
<th>Reverse Shear Option</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>ARIES-I’</td>
<td>ARIES-I</td>
<td>ARIES-RS</td>
</tr>
<tr>
<td>Major radius (m)</td>
<td>8.0</td>
<td>6.75</td>
<td>5.5</td>
</tr>
<tr>
<td>β (βₜ)</td>
<td>2% (2.9)</td>
<td>2% (3.0)</td>
<td>5% (4.8)</td>
</tr>
<tr>
<td>Peak field (T)</td>
<td>16</td>
<td>19</td>
<td>16</td>
</tr>
<tr>
<td>Avg. Wall Load (MW/m²)</td>
<td>1.5</td>
<td>2.5</td>
<td>4</td>
</tr>
<tr>
<td>Current-driver power (MW)</td>
<td>237</td>
<td>202</td>
<td>81</td>
</tr>
<tr>
<td>Recirculating Power Fraction</td>
<td>0.29</td>
<td>0.28</td>
<td>0.17</td>
</tr>
<tr>
<td>Gross Thermal efficiency</td>
<td>0.46</td>
<td>0.49</td>
<td>0.46</td>
</tr>
<tr>
<td>Cost of Electricity (1992c/kWh)</td>
<td>10</td>
<td>8.2</td>
<td>7.5</td>
</tr>
</tbody>
</table>

Approaching COE insensitive of power density

* Estimated at 6.5 c/kWh (2010$) using Gen-IV costing methodology
ARIES-AT (tokamak) Fusion Core

Cutaway of the ARIES-AT Fusion Power Core

Cross Section of ARIES-AT Power Core Configuration
The ARIES-AT utilizes an efficient superconducting magnet design

- On-axis toroidal field: 6 T
- Peak field at TF coil: 11.4 T
- TF Structure: Caps and straps support loads without inter-coil structure;

Superconducting Material

- Either LTC superconductor (Nb$_3$Sn and NbTi) or HTC
- Structural Plates with grooves for winding only the conductor.
Configuration & Maintenance are important aspects of the design.

1. Install 4 TF coils at a times
2. Insert ¼ of inner VV and weld
3. Complete the torus
4. Insert maintenance ports and weld to inner part of VV and each other
5. Install outer walls and dome of the cryostat

vacuum vessel

Inner part: 4 pieces, welded during assembly
Complete vessel, outer part is entirely made of maintenance ports
Modular sector maintenance enables high availability
ARIES-AT Fusion core is segmented to minimize rad-waste and optimize functions.
Fusion core segmented into modules and replaced by articulated booms through ports.
Tritium Systems -- Scale & Requirements
Tritium inventories in a fusion plant are limited by safety & licensing requirements.
Tritium Systems and Flows: Scale

Losses:
- **Burn**: 1GW fusion power requires 55.6kg T/FPY
  - A typical 1GWe plant burns ~124kg/FPY (340g/FPday)
- **Other Losses**:
  - Maximum allowed tritium release to environment ~ 1-2 g/y (Current fission reactor emissions: ~60 mg/y)
  - Radioactive decay: 5.5% of the inventory per year (275 g/y loss for 5 kg inventory)

Production:
- Burned tritium: 124kg/FPY
- Generate start-up inventory for another fusion plant (~5 kg)

The dominant breeding constraint is T burn (TBR \(\approx 1\))
Over-breeding of tritium rapidly leads to large tritium inventories.

Control of TBR within 1-2% in a week time scale is necessary (or need to store of a large quantity of T).

Excess T should be compared to reserves/margins (~20% of inventory).
TBR can be controlled in liquid breeders by adjusting $^6$Li enrichment.
Tritium Systems -- Fueling & Exhaust
Constraints imposed by the Fuel Processing System $T$ inventory

Fuel processing (FP) inventory $= 0.5 \; \tau_{FP} \; S_f$

$$\frac{\tau_{FP}}{f_b} = 2 \frac{\text{FP inventory}}{\text{T Burn Rate}}$$

For 1GWe plant: $\frac{\tau_{FP}}{f_b} = \frac{2 \times 3\text{kg}}{0.34 \text{ kg/d}} = 17.6 \text{ d}$

Estimates: 1 day processing time ($\tau_{FP}$) and 6% burn fraction ($f_b$)
Managing the plasma material interface

- Alpha power and alpha ash has to eventually leave the plasma
  - Particle and energy flux on the material surrounding the plasma

- Modern tokamaks use divertors:
  - Closed flux surfaces containing hot core plasma
  - Open flux surfaces containing cold plasma diverted away from the first wall.
  - Particle flux on the first wall is reduced, heat flux on the first wall is mainly due to radiation (bremsstrahlung, synchrotron, etc.)
  - Alpha ash is pumped out in the divertor region
  - High heat and particle fluxes on the divertor plates.
Burn fraction and exhaust mix are tightly coupled to divertor operation

- ITER and power plants require pellet fueling (high-field side) because of the high density of edge plasma.
  - Current experiments mainly use gas injection. (Pellet fueling is demonstrated).
- The current scenario for reducing heat flux on the divertor plates is by gas injection (and impurities) in the divertor to form a “detached plasma”
  - It is demonstrated in current experiments but scaling to ITER/power plant is an open question (several scaling laws fit the observations).
- ITER operation will verify current estimates.
ITER fuel processing system is ~20 larger than the state-of-the-art and will demonstrate power-plant scale operation.

- (See Scott Willms presentation).

ITER design requires a complete separation of isotopes (separate D, T, H, CH, …)

The scale of fuel processing system can be reduced substantially if D and T are NOT separated.

- Analysis for ARIES-I estimates a factor of 5 reduction in the cryogenic distillation system.
- It may also be possible to recycle most of the exhaust and process only a fraction to further reduce fuel-processing inventory.
Tritium in the Fusion Core
Irradiation leads to a operating temperature window for material.

Additional considerations such as He embrittlement and chemical compatibility may impose further restrictions on operating window.

$\eta_{\text{Carnot}} = 1 - \frac{T_{\text{reject}}}{T_{\text{high}}}$

New structural material should be developed for fusion application

Candidate “low-activation” structural material:
- Fe-9Cr steels: builds upon 9Cr-1Mo industrial experience and materials database
  - 9-12 Cr ODS steel is a higher-temperature option.
- SiC/SiC: High risk, high performance option (early in its development path)
- W alloys: High performance option for PFCs (early in its development path)
Radioactivity levels in fusion power plants are very low and decay rapidly after shutdown.

- SiC composites lead to a very low activation and afterheat.
- All components of ARIES-AT qualify for Class-C disposal under NRC and Fetter Limits. 90% of components qualify for Class-A waste.

After 100 years, only 10,000 Curies of radioactivity remain in the 585 tonne ARIES-RS fusion core.
Waste volume is modest (ARIES-AT)

- 1270 m³ of Waste is generated after 40 full-power year of operation.
  - Coolant is reused in other power plants
  - 29 m³ every 4 years (component replacement), 993 m³ at end of service
- Equivalent to ~ 30 m³ of waste per full-power operation.
  - Effective annual waste can be reduced by increasing plant service life.

- 90% of waste qualifies for Class A disposal
Continuity of ARIES Research Has Led to the Progressive Refinement In Blanket Technologies

<table>
<thead>
<tr>
<th>ARIES-I:</th>
<th>ARIES-RS:</th>
<th>ARIES-ST/ARIES-CS:</th>
<th>ARIES-AT:</th>
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<tbody>
<tr>
<td>• SiC composite with solid breeders</td>
<td>• Li-cooled vanadium</td>
<td>• Dual-cooled ferritic steel with SiC inserts</td>
<td>• LiPb-cooled SiC composite</td>
</tr>
<tr>
<td>• Advanced Rankine cycle</td>
<td>• Advanced Rankine Cycle</td>
<td>• Advanced Brayton Cycle at $\geq 650$ °C</td>
<td>• Advanced Brayton cycle with $\eta = 59%$</td>
</tr>
</tbody>
</table>

- Many issue with solid breeder design (could not utilize SiC capabilities)!
- Liquid breeders?
- Insulating coating Development!
- Increase coolant temperature above structure temperature?
- Attractive Concept!
- SiC composite version?
Solid-breeder blanket concepts have many issues

- Low-performance (dictated by breeder T release temperature window)
- High structural and low Li content (a lot of Be multiplier)
- Cannot control tritium breeding in-situ.
Continuity of ARIES Research Has Led to the Progressive Refinement In Blanket Technologies

ARIES-I:
- SiC composite with solid breeders
- Advanced Rankine cycle

ARIES-RS:
- Li-cooled vanadium structure
- Advanced Rankine Cycle

ARIES-ST/ARIES-CS:
- Dual-cooled ferritic steel with SiC inserts
- Advanced Brayton Cycle at $\geq 650 \, ^\circ C$

ARIES-AT:
- LiPb-cooled SiC composite
- Advanced Brayton cycle with $\eta = 59\%$

- Many issues with solid breeder design (could not utilize SiC capabilities)!
- Liquid breeders?
- Insulating coating Development!
- Increase coolant temperature above structure temperature?
- Attractive Concept!
- SiC composite version?
First wall and partitioning walls are cooled with He.
Most of fusion neutron energy is deposited in PbLi coolant/breeder.
SiC insert separates PbLi from the walls: They reduce a) MHD effects and b) heating of the walls by LiPb
Outlet coolant temperature of ~700°C (Max. steel temperature of ~550°C)
Outboard blanket & first wall

ARIES-AT
high-performance blanket (SiC Composite)

- Simple, low pressure design with SiC structure and LiPb coolant and breeder.
- Innovative design leads to high LiPb outlet temperature (~1,100°C) while keeping SiC structure temperature below 1,000°C leading to a high thermal efficiency of ~ 59%.
- Simple manufacturing technique.
- Very low afterheat.
- Class C waste by a wide margin.
Design leads to a LiPb Outlet Temperature of 1,100°C While Keeping SiC Temperature Below 1,000°C

Two-pass PbLi flow, first pass to cool SiCf/SiC box second pass to superheat PbLi

PbLi Inlet Temp. = 764 °C
Max. SiC/SiC Temp. = 996°C
Max. SiC/PbLi Interf. Temp. = 994 °C

Poloidal distance (m)
Radial distance (m)

SiC/SiC
Pb-17Li

Bottom
Top

Bottom

PbLi Outlet Temp. = 1100 °C
Sophisticated 3D CAD model are used for neutronics calculations (ARIES DCLL blanket sector)
Impact of blanket features of TBR (ARIES DCLL blanket Sector)

1. 1-D Infinite Cylinder: 100% LiPb breeder surrounded with FS shield
2. \( \text{Li}_{17}\text{Pb}_{83} \) surrounding plasma in toroidal geometry
3. \( \text{Li}_{15.7}\text{Pb}_{84.3} \) surrounding plasma
4. \( \text{Li}_{15.7}\text{Pb}_{84.3} \) confined to 80 cm OB blanket and 45 cm IB blanket. Outer FS shield, and W-based divertor added
5. 2 cm assembly gap between blanket modules
6. Materials assigned to 3.8 cm thick IB and OB FW
7. Materials assigned to side, bottom/top, and back walls of blankets
8. IB and OB cooling channels added
9. SiC Flow Channel Inserts added
10. Stabilizing shell added
11. Extended IB Blanket
12. Extended OB Blanket
13. Extended IB and OB Blanket
14. 70% enriched Li with extended IB and OB blankets
Uncertainty in TBR calculations is estimated at 4% for ARIES Dual-cooled blanket.

- Uncertainties in LiPB nuclear data base 3%
- Deficiencies in modeling 1%

A large operational window is available by adjusting $^6$Li enrichment.
Off-normal events and accident scenarios are also analyzed

- Pressurization of blanket modules due internal break of He channels (Dual-cooled blanket)

- Disruption forces and thermal loads
  - The blanket internals were not developed in sufficient detail to do a complete study (focus of current study)

- Quench of TF coils

- Detailed accident analysis (e.g., loss of coolant, loss of flow)*
  - Most of the off-site dose after an accident is due to tritium release from fusion core. Fusion core inventories are:
    - SiC-composite, LiPb coolant (ARIES-AT): ~750 g
    - Dual-cooled blanket (ARIES-CS): ~1 kg

* See B. Merrill’s presentation
Impact of confinement concept and advanced fuels

- We have examined stellarators, spherical tokamaks, and reversed-field pinches:
  - We have NOT uncovered any particular advantages in blanket design and operation compared to those of tokamaks.

- Advanced fuels (DD or D-\(^3\)He) do NOT require breeding. However,
  - All D-based fuel cycles generate tritium and tritium management issue remain
  - First-wall/blanket designs are NOT necessarily simplified as most of the fusion power appear as heat flux on the first wall (as opposed to volumetric heating in the blanket in DT cycle).
In Summary:

- We do not envision any show-stoppers in breeding tritium.
- The main issue is NOT tritium breeding but T management:
  - Minimizing tritium inventory
  - Minimizing tritium loss to the environment
  - On-line Control of tritium breeding ratio.

While satisfying all other constraints:

- ITER will verify tritium fueling and processing technologies
- We are in early stages of development of a breeding blankets:
  - Extensive R&D is required to examine the actual hardware development that ranges from basic material properties characterization to fully integrated subsystems, like the blanket.
Thank You!