The ARIES-ST Spherical Tokamak Power Plant Study

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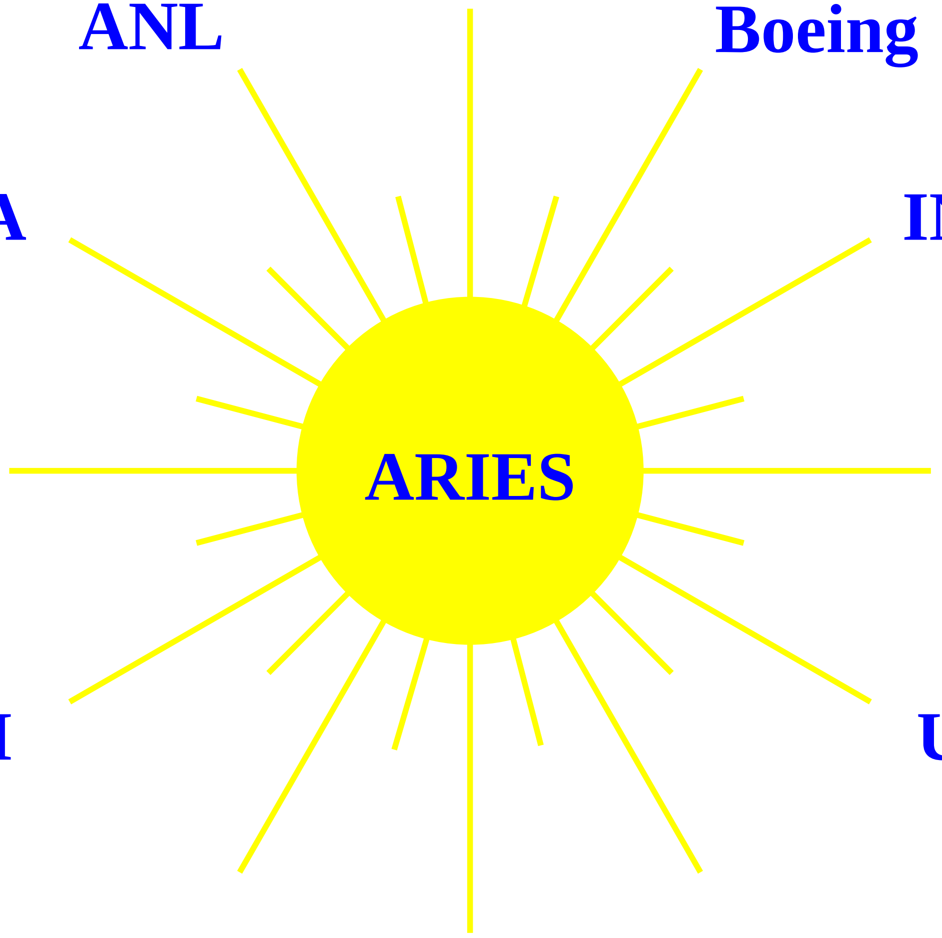

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Collaborations

The ARIES-ST Study

- The ARIES-ST study is a two-year project to investigate the potential of spherical tokamaks as commercial power plants as well as vehicles for fusion development.

- This study was preceded by an examination of critical issues of spherical tokamaks which was done as part of the Starlite study.

- The ARIES-ST study began in Jan. 1997. The effort has been focused on key critical components as well as those areas which are unique to spherical tokamaks.

- The ARIES-ST study will be completed in Fall 1998. This paper represents a progress report as in many areas several options are still under consideration.

Key Physics Issues for Spherical Tokamaks

- Because of low-aspect ratio, the area in the inbarod is limited. A resisitve TF coil system is probably the only option.

- In order to minimize the Joule losses in the TF coils (mainly center-post), MHD equilibria with very high β are required.

- Because there is no room for a central solenoid, these devices have to operate at steady-state. Because of large plasma current inherent to spherical tokamaks, only MHD equilibria with almost perfect bootstrap alignment would lead to a reasonable current-drive power.

- Because of unique magnetic topology, on-axis current drive with RF techniques is difficult. Needed current-drive for profile control as well as start-up are additional challenges.

- Divertors!

MHD Equilibrium and Stability

- We have examined MHD stability of ST discharges for a wide range of aspect ratio, elongation, triangularity, and kink wall location (∼99% bootstrap fraction).

- Typically, β_N exhibit a peak at $A \simeq 1.4$ For a fixed triangularity. β_N ($\beta \sim \beta_N^2(1+\kappa^2)$) increases with increasing κ.

- At least for $A = 1.6$, both ballooning and kink β limits drop significantly with increasing δ. Probably the lowest useful triangularity is $\delta = 0.45$ for $A = 1.6$.

- There is a high leverage to operate at very high elongations (and high δ) in order to achieve a very high β. Because of natural elongation of ST equilibria, it appears that operation at $\kappa \sim 3$ is possible. Detail work in quantifying the feed-back power necessary for vertical stabilization of high-elongation ST plasmas is on-going.

MHD Equilibrium and Stability

- Free-boundary equilibria of low-A equilibria is unique and difficult.
  - ⋆ Strong B variation.
  - ⋆ Natural shape.
  - ⋆ Strong plasma shaping ($\kappa \gtrsim 3$, $\delta \gtrsim 0.5$).
  - ⋆ no PF coils on the inboard side.
  - ⋆ High β_p (~ 2) and low ℓ_i (~ 0.15).
- Because there is no PF coils in the inboard, the inboard plasma boundary cannot be directly controlled.

Current Drive

- It appears that LFFW is the only theoretically plausible RF technique that drives current near the axis on high-β ST plasmas.
  - ⋆ Because $\omega_{pe}/\omega_{ce} \gg 1$, EC and LH waves cannot access the plasma center.
  - ⋆ HFFW does not penetrate to the center because of strong electron and/or ion damping.
  - ⋆ ICRF fast-wave suffer strong electron and alpha/ion damping.
- LFFW current drive has been tried on JET with mixed results. It also requires a large antenna structure for a well-defined spectrum ($\lambda_{\|} \sim 14$ m). It generally has a fairly low current-drive efficiency.
- HFFW can drive the current in mid-plasma efficiently.
- We are currently exploring NBI as an alternative option as well as the current-drive power needed for controlling the profiles.

Several ARIES-ST Options

Aspect ratio	**1.25**	**1.40**	**1.60**	**1.80**
Major toroidal radius (m)	5.06	3.71	2.93	2.97
Plasma minor radius (m)	4.05	2.65	1.83	1.65
Plasma elongation	3.00	3.42	3.20	3.00
Plasma triangularity	0.45	0.55	0.57	0.57
Toroidal β (%)	36.3	44.4	35.3	27.7
Electron density ($10^{20}/\text{m}^3$)	1.29	2.00	3.03	3.16
ITER-89P scaling multiplier	2.89	2.72	2.65	2.65
Plasma current (MA)	41.4	36.3	28.9	25.2
CD power to plasma (MW)	7.9	22	42	39
On-axis toroidal field (T)	1.81	2.04	2.82	3.26
Peak field at TF coil (T)	15.8	13.6	14.0	11.9
TF-coil ohmic losses (MW)	1570	900	636	426
Peak neutron load (MW/m^2)	3.44	5.2	8.6	9.0
P_{TR}/R_T (MW/m)	136	155	182	157
Recirculating power fraction	0.63	0.50	0.43	0.35
Cost of electricity (mill/kWeh):	114.8	90.8	84.2	87.2

Divertors

- The divertor problem is much more difficult than conventional and advanced tokamaks (higher P/R).

- To bring down the heat flux on the divertors to a reasonable level, almost all of the plasma power has to be radiated.
  - ⋆ Radiative mantle.
  - ⋆ Impurity radiation in the divertor channel.

- Impact of impurities on the MHD/current drive is an issue.

- This approach mainly transfers the divertor problem to the first-wall design.

Key Engineering Issues for Spherical Tokamaks

- The small area available for the the inboard legs of the TF coils (center-post) make the center-post challenging.

- Potential advantages of spherical tokamak (high-wall load) make the engineering of first wall and blanket difficult.
  - ⋆ Water-cooled copper coils further narrows the options.
  - ⋆ Highly shaped components (tall and thin) make mechanical design difficult.

- Maintenance of the power core should include provisions for rapid replacement of central column.

- Divertors!

Center-Post Design

- A 20 to 30-cm thick shield is required:
  - ⋆ To ensure low-level waste disposal of center-post material;
  - ⋆ To reduce the nuclear heating in the center-post conductor (and possibly recover this power).

- Leading conductor material is Glidcop AL-15.
  - ⋆ It has adequate strength, ductility, low swelling, and thermal and electrical conductivities;
  - ⋆ It suffers from severe embrittlement (at room temperature);
  - ⋆ Hardening and embrittlement are alleviated by operating at $\gtrsim$ 180°C but then it suffers from severe loss of fracture toughness.

Center-Post Mechanical Design

- Single-turn TF coils
  - ⋆ Higher packing fraction;
  - ⋆ Reduced Shielding requirement (no insulation).

- Sliding electrical joints between Center-post and other TF legs and bus-bars and TF legs.
  - ⋆ Allows relative motion in radial and vertical direction (minimizes axial loads on the center-post).
  - ⋆ Enhances maintainability.
  - ⋆ Several design options have been developed and tested.

- Center-post is physically separate from other components in order to avoid a complex interface.

- Leading conductor material is Glidcop AL-15. However, under irradiation, it suffers from severe embrittlement (at room temperature) or severe loss of fracture toughness (above 250°C).

Center-Post Design-Cooling Options

- Cryo-cooling does not offer major improvement over cooling options at room temperature and above.

- Water-Cooling is the leading option: itemitem⋆ Low-temperature ($T_{inlet} \sim 35°$C) to minimize Joule losses but results in severe embrittlement of conductor.

  ⋆ High-temperature ($T_{inlet} \sim 150 - 180°$C) Avoid embrittlement but lose of fracture toughness and increase Joule losses are key issues.

- Liquid Lithium is properly the best option for very-high temperature operation. In addition to severe engineering difficulties, economic recovery of center-post heating does not offset increased Joule losses.

First Wall and Blanket Options

- Design including solid-breeders require a major improvement in the thermal conductivity of solid breeders to handle high wall loads.

- Self-cooled Li/V option can handle the high-wall load but the use of liquid Li in the vicinity of a water-cooled center-post raises many safety concerns.

- Two blanket designs utilizing ferritic steels with helium coolant and LiPb as the liquid breeder are under consideration. SiC composite fillers are used to achieve a high-coolant outlet temperature and a reasonable power-conversion efficiency.

- Analysis of capability of the blanket options (including Li/V design) is on-going to parameterize the capability of these blanket in terms of neutron and surface heat flux capabilities.

Summary

- Several key challenging issues confront spherical tokamaks as fusion power plants. We have proposed some potential solutions.

- It appears that spherical tokamak power plants do not offer quantum improvements over advanced tokamaks.

- In the up-coming year, we will examine potential of spherical tokamaks as vehicles for fusion development.