Status of Engineering Effort on ARIES-CS Power Core

Presented by

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With contributions from L. El-Guebaly, S. Malang, B. Merrill, X. Wang and the ARIES Team

ARIES Meeting

UCSD

January 23, 2006

Outline

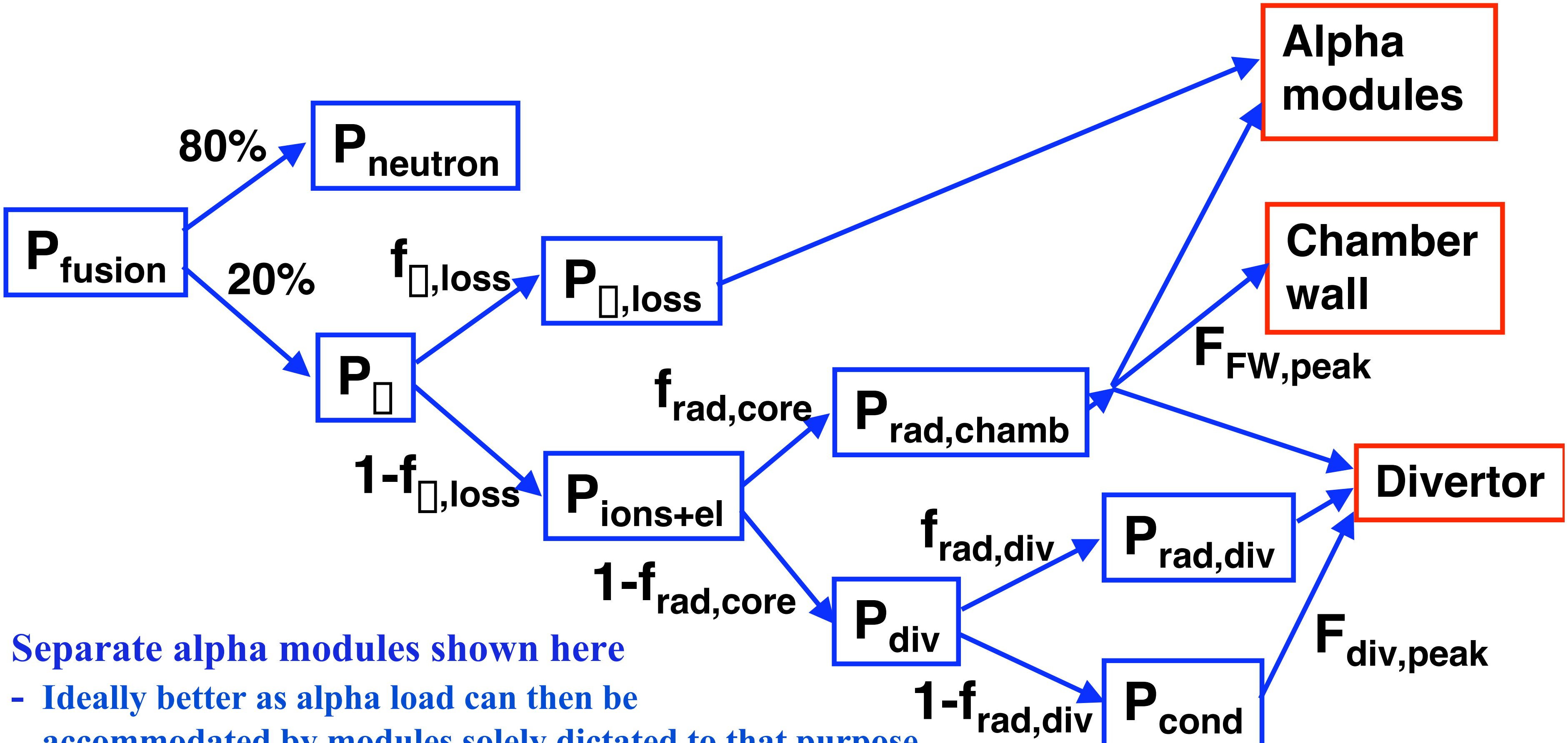
This presentation:

- **Power flows and heat loads on power core components**
- **Alpha particle threat and possible accommodation**
- **Thermal-hydraulic optimization of dual coolant blanket coupled to Brayton cycle for updated heat loads and including friction thermal power and coolant pumping power to estimate net efficiency (including updated divertor parameters)**
- **Orbital tool for remote pipe cutting and rewelding**
- **Cryostat design**
- **Progress on divertor study**
- **Work documentation**

Other presentations:

- **Maintenance and layout update for shield-only and transition regions (Xueren)**
- **Revised radial builds for breeding and shielding only modules (Laila)**
- **Coil design and analysis (Xueren, Leslie)**
- **Plan for divertor thermal-hydraulic experiment (Said)**

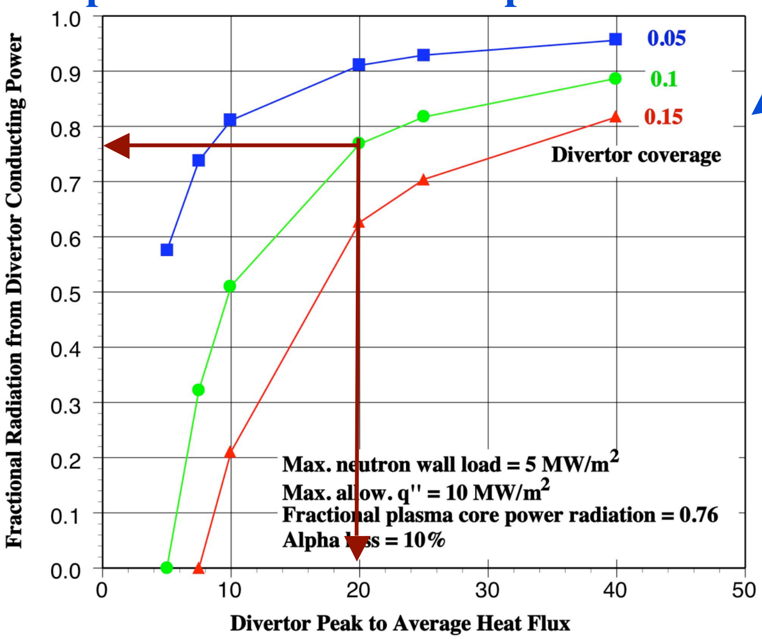
Power Flow Diagram for Estimating Maximum Heat Fluxes on PFC's for ARIES-CS (with input from J. Lyon)



- Separate alpha modules shown here
  - Ideally better as alpha load can then be accommodated by modules solely dictated to that purpose.
  - However, extra coverage for alpha modules (~5%??) has adverse impact on breeding.
  - Can we assume that all alpha power goes to divertor?
  - More efficient coverage (e.g. 10% for combined divertor and alpha module functions) but can divertor modules accommodate extra load requirement?
  - Final design decision depends on divertor and alpha loss physics to get a better idea on location and peaking factors (KEY AREA WHERE WE NEED AN ANSWER SOON)

Illustration of Requirements on Different Parameters for Separate and Combined Divertor and Alpha Modules

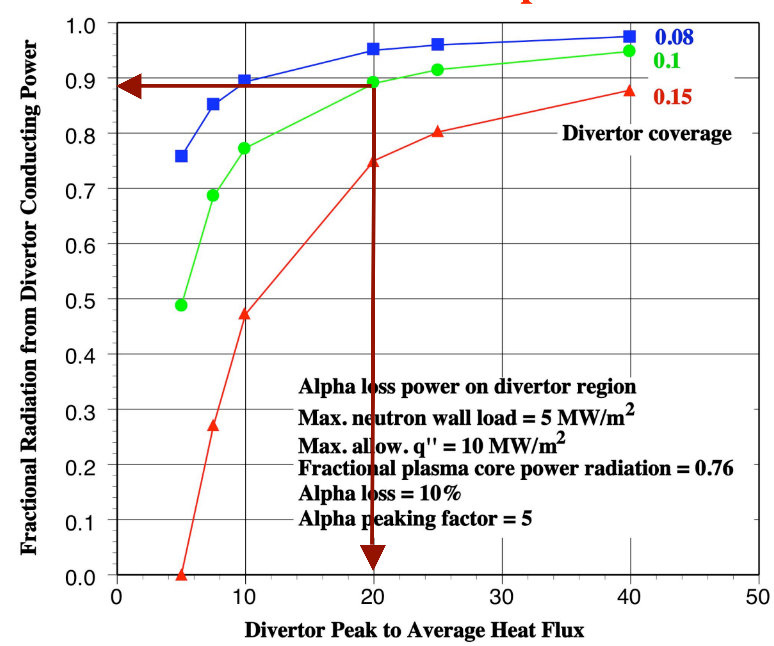
Separate Divertor and Alpha Modules

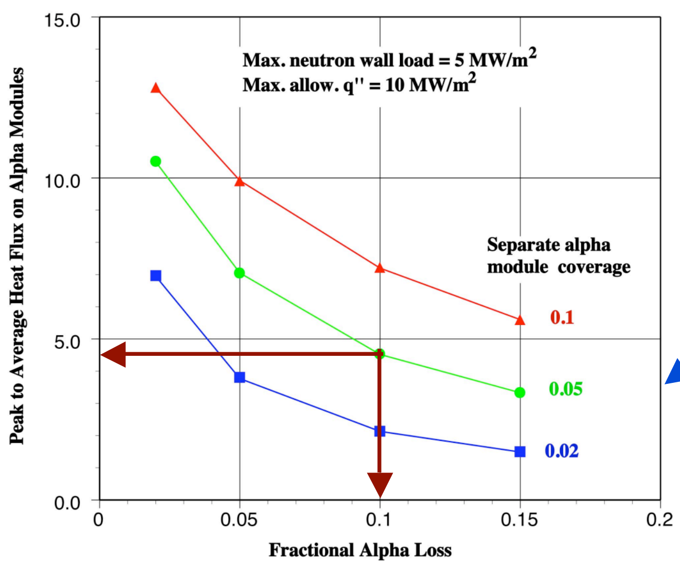


Example parameters for separate divertor modules:
Divertor coverage = 0.1
Divertor peaking factor = 20
Fractional radiation in divertor region ~ 0.76

Example parameters for combined divertor + modules:
 loss = 0.1
 peaking factor = 5
Divertor+ coverage = 0.1
Divertor peaking factor = 20
Fractional radiation in divertor region ~ 0.895

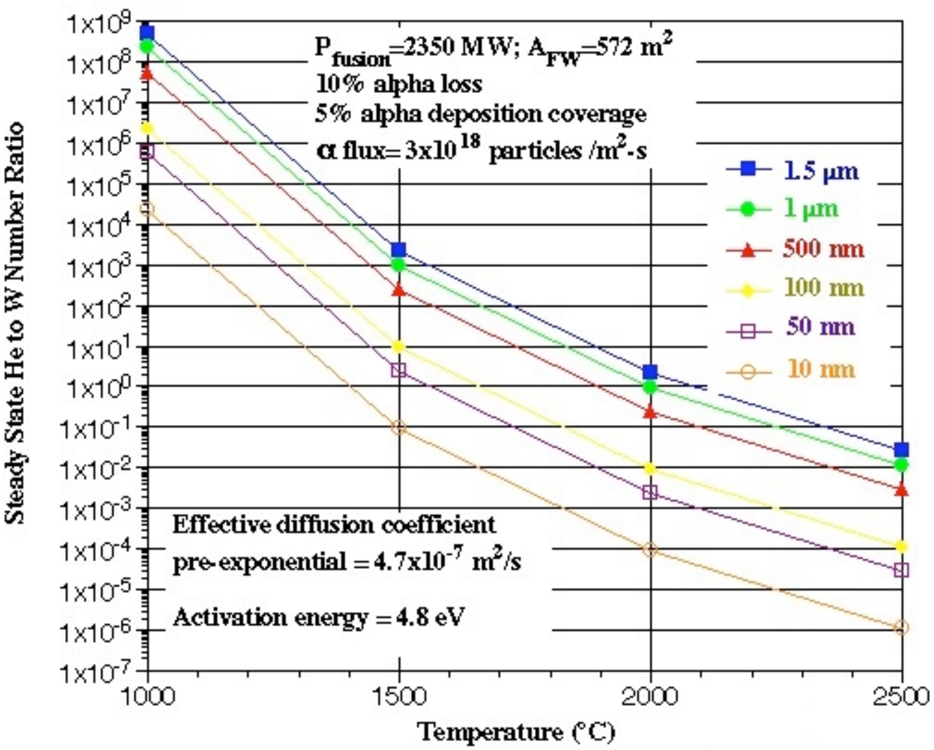
Combined Divertor and Alpha Modules





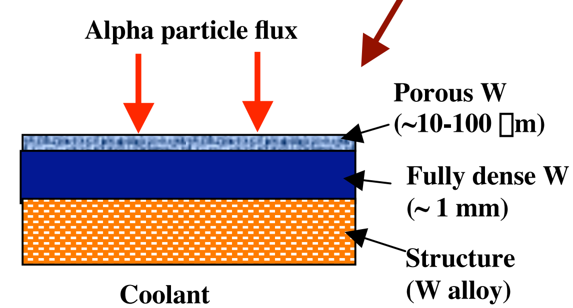
Example parameters for separate modules:
 loss = 0.1
 module coverage = 0.05
 flux peaking factor ~ 4.5

He Implantation in Armor from Prompt □-Losses



- Simple effective diffusion analysis for different characteristic diffusion dimensions for an activation energy of 4.8 eV
- Not clear what is the max. He conc. limit in W to avoid exfolation (perhaps ~0.15 at.%)
- High W temperature needed in this case
- Shorter diffusion dimensions help, perhaps a nanostructured porous W (PPI)
- e.g. 50-100 nm at ~1800°C or higher

- An interesting question is whether at a high W operating temperature (>~1400°C), some annealing of the defects might help the tritium release.
- This is a key issue for a CS which needs to be further studied to make sure that a credible solution exists both in terms of the alpha physics, the selection of armor material, and better characterization of the He behavior under prototypic conditions.

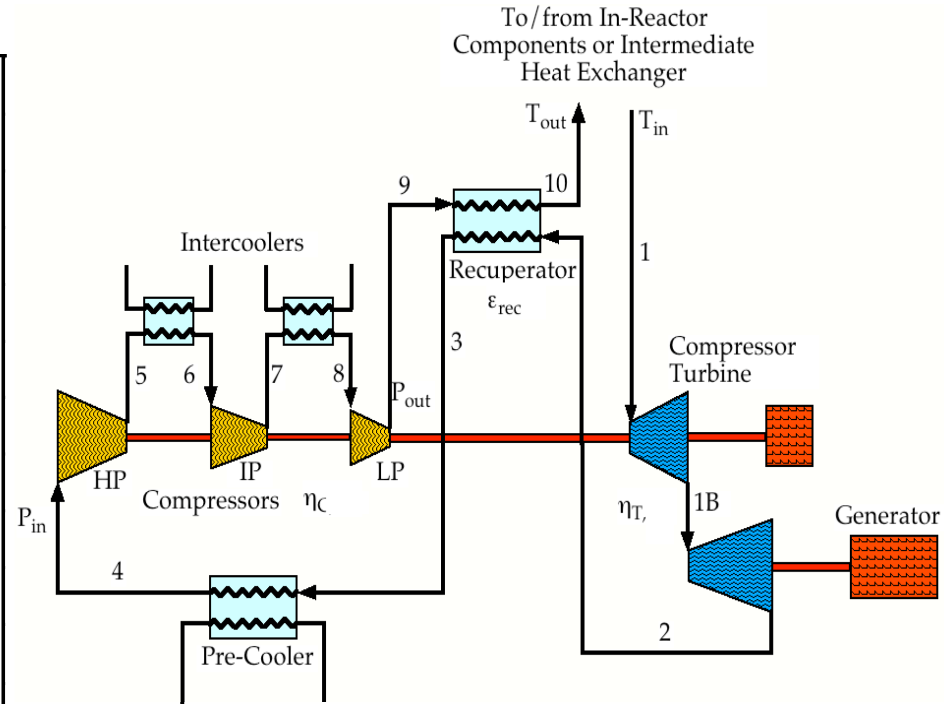


(paper accepted for presentation at 17th PSI, China, May 2006)

Optimization of DC Blanket Coupled to Brayton Cycle

- **Net efficiency calculated including friction thermal power from He coolant flow and subtracting required pumping power to estimate net electrical power.**
- **Brayton cycle configuration and typical parameters summarized here.**

Number of Compression Stages	**3**
Number of Expansion Stages	**1**
HX Min. Temp. Difference between Hot and Cold Legs	**30°C**
Compressor Efficiency	**0.89**
Turbine Efficiency	**0.93**
Recuperator Effectiveness	**0.95**
Total Compression Ratio	**3.5**
Cycle Lowest He Temperature	**35°C**
Fractional Cycle He Pressure Drop	**0.045**
Cycle He Pressure	**15 MPa**
Gross Efficiency for Example Case ($\Gamma_{N,max}$=5 MW/m^2)	**~0.42**
Net Efficiency (+ contribution of blanket & divertor He friction thermal power – pumping power)	**~0.39**



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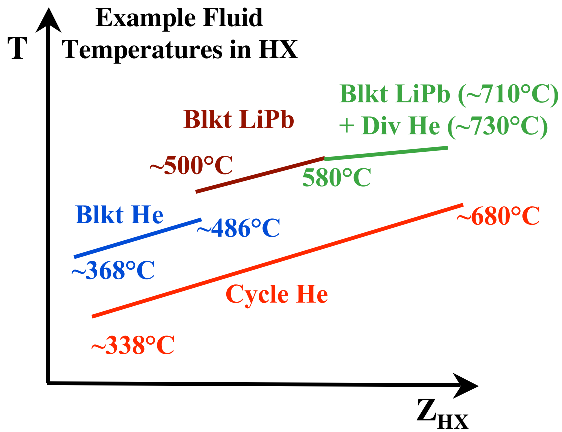
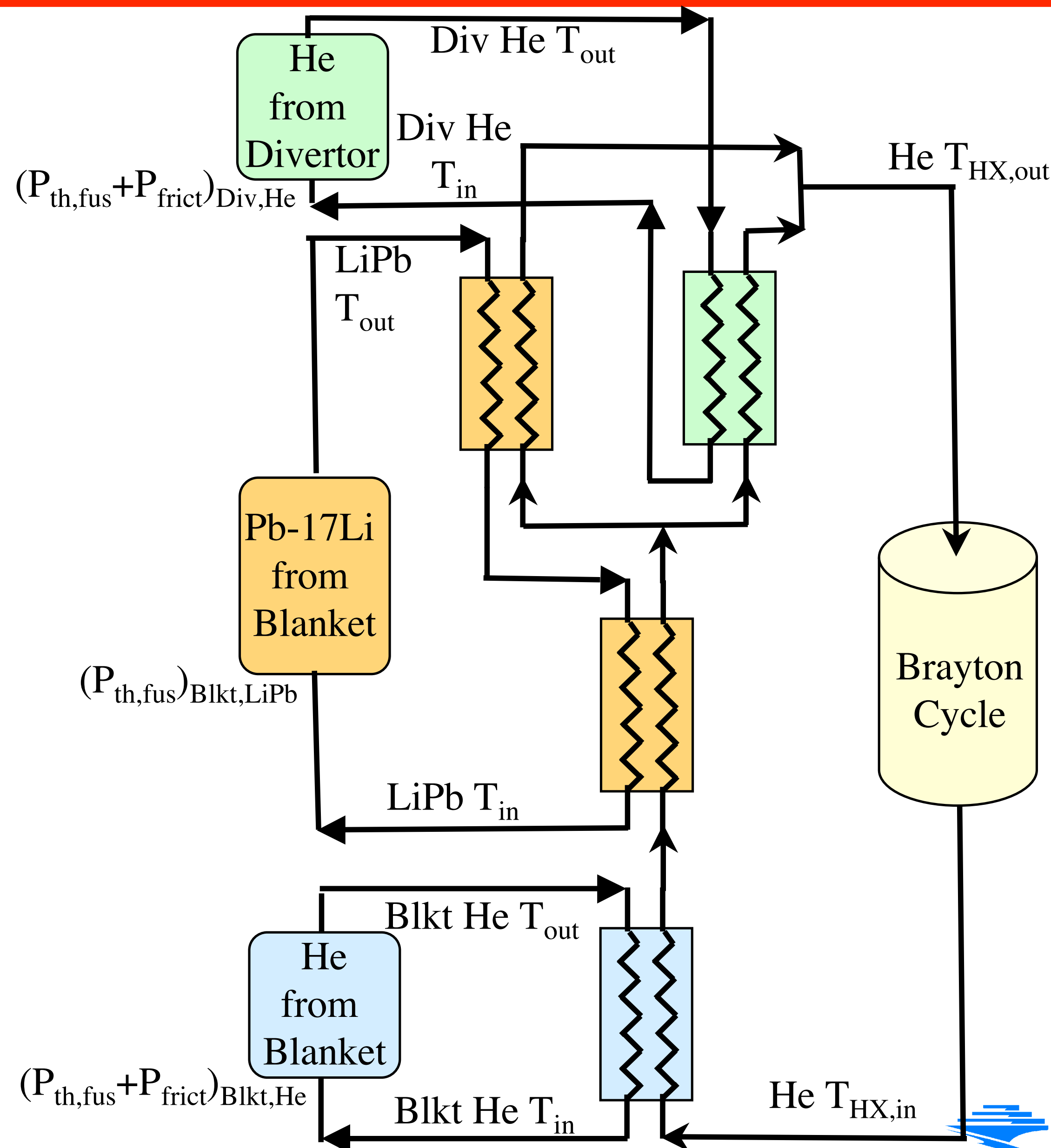
Details of Coolant Routing Through HX Coupling Blanket and Divertor to Brayton Cycle

- **Div He T_{out} ~ Blkt Pb-17Li T_{out}**
- **Min. DT_{HX} = 30°C**
- **$P_{Friction}$ ~ □$_{pump}$ x P_{pump}**



Power Parameters for Example Case

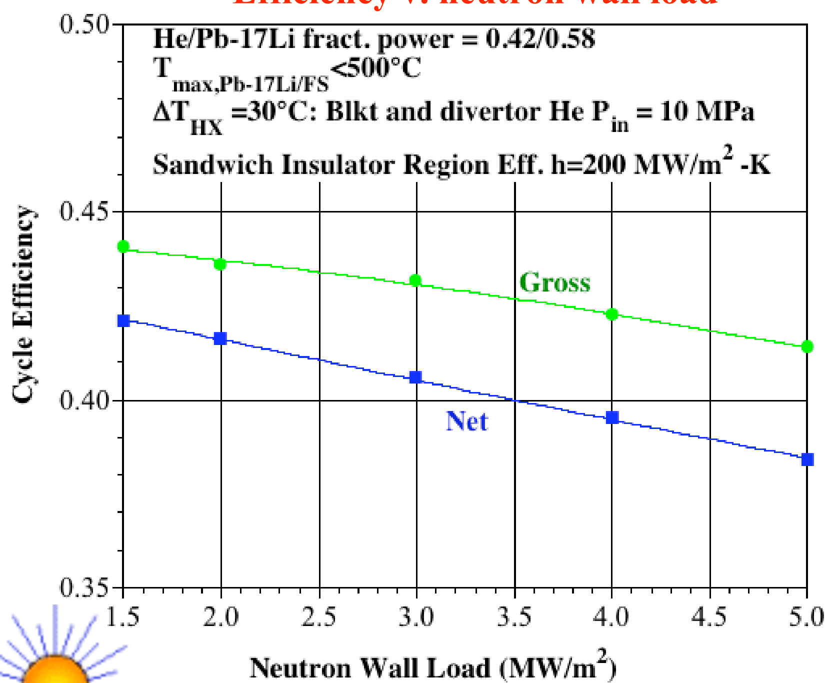
Fusion Thermal Power in Reactor Core	2604 MW
Fusion Thermal Power Removed by Pb-17Li	1373 MW
Fusion Thermal Power Removed by Blkt He	994 MW
Friction Thermal Power Removed by Blkt He	87 MW
Fusion Thermal Power Removed by Div He	237 MW
Friction Thermal Power Removed by Div He	26 MW
Fusion + Friction Thermal Power in Reactor Core	2717 MW



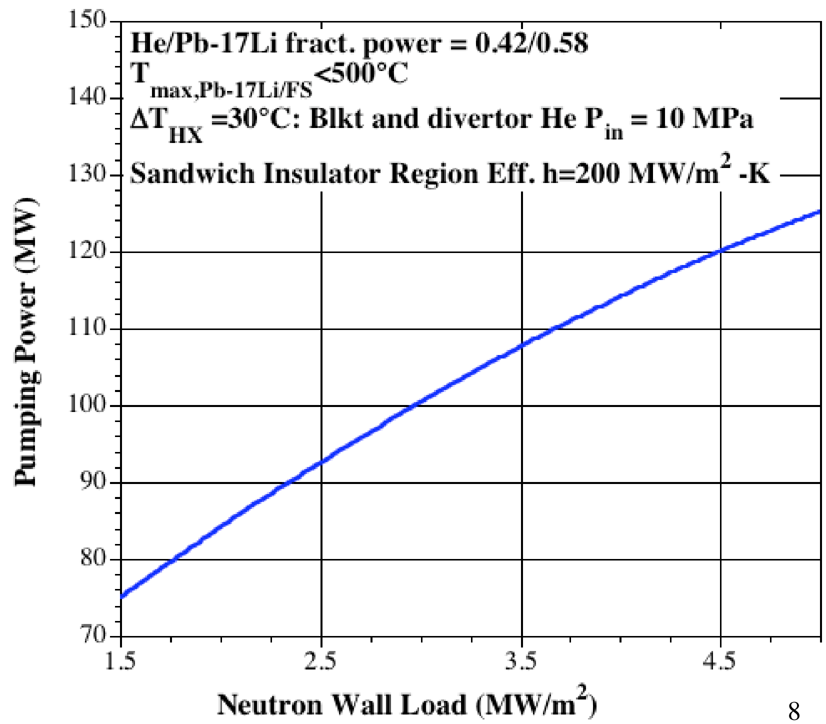
Optimization of DC Blanket Coupled to Brayton Cycle Assuming a FS/Pb-17Li Compatibility Limit of 500°C and ODS FS for FW

- $\eta_{Brayton,gross} = P_{elect,gross} / P_{thermal,fusion}$
- $\eta_{Brayton,net} = (P_{elect,gross} - P_{pump}) / P_{thermal,fusion}$
- Use of an ODS FS layer on FW allows for higher operating temperature and a higher neutron wall load.
- The optimization was done by considering the net efficiency of the Brayton cycle for an example 1000 MWe case.

Efficiency v. neutron wall load

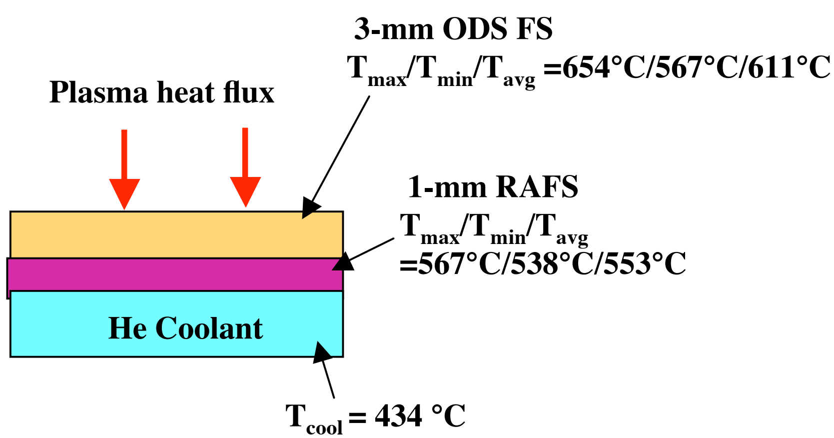


Total in-reactor pumping power v. neutron wall load



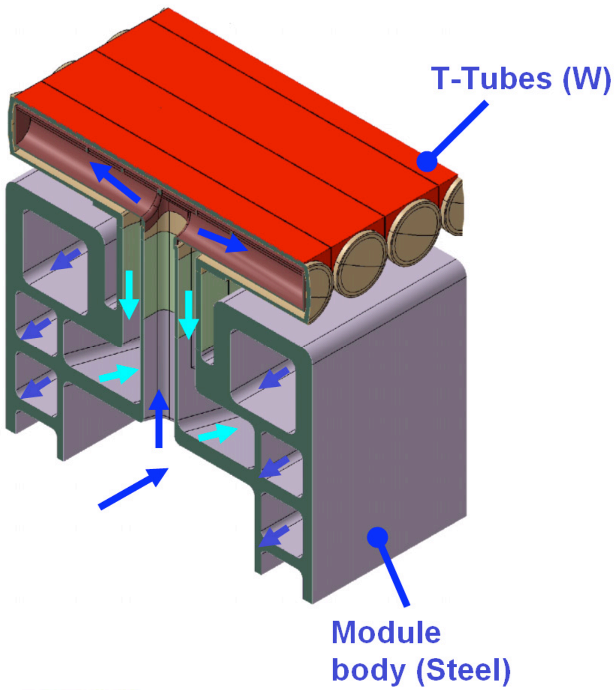
Summary of Parameters for Dual Coolant Concept for Example Case

- **Avg/max. wall load =3.3/5 MW/m^2**
- **Avg/max. plasma q'' = 0.57/0.86 MW/m^2**

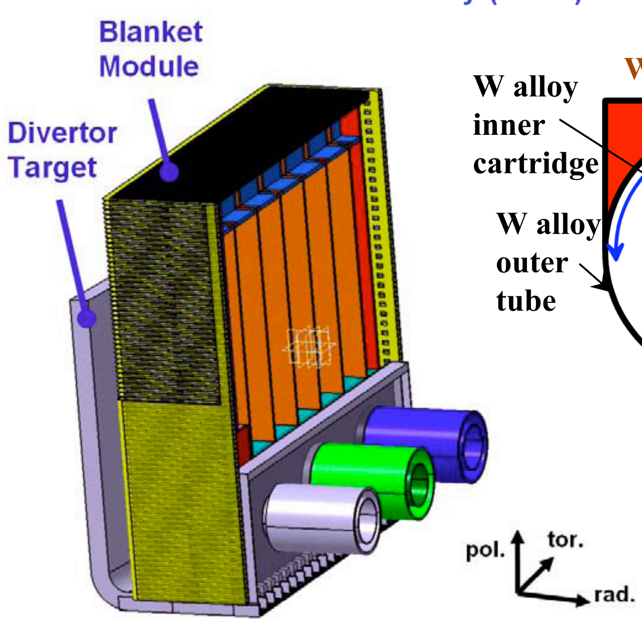


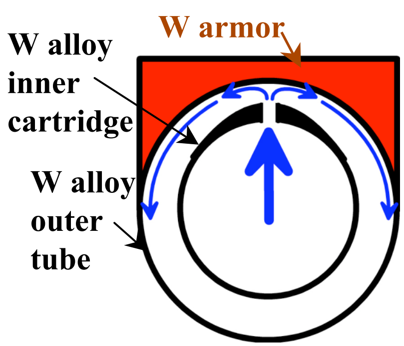
Blanket	
Typical Module Dimensions	2 m x 2 m x 0.62 m
Tritium Breeding Ratio	1.1
Fusion Thermal Power in Blanket	2367 MW
Blanket Pb-17Li Coolant	
Pb-17Li Inlet Temperature	500°C
Pb-17Li Outlet Temperature	710°C
Pb-17Li Inlet Pressure	1 MPa
Typical Inner Channel Dimensions	0.26 m x 0.24 m
Thickness of SiC Insulator in Inner Channel	5 mm
Effective SiC Insulator Region Conductivity	200 W/m^2-K
Average Pb-17Li Velocity in Inner Channel	~0.1 m/s
Fusion Thermal Power removed by Pb-17Li	1373 MW
Pb-17Li Total Mass Flow Rate	35,100 kg/s
Pb-17Li Pressure Drop	~1 kPa
Pb-17Li Pumping Power	~ 4.7 kW
Maximum Pb-17Li/FS Temperature	500°C
Blanket He Coolant	
He Inlet Temperature	368°C
He Outlet Temperature	486°C
He Inlet Pressure	10 MPa
Typical FW Channel Dimensions (poloidal x radial)	2 cm x 3 cm
He Velocity in First Wall Channel	56.6 m/s
Total Blanket He Pressure Drop	0.34 MPa
Fusion Thermal Power Removed by He	994 MW
Friction Thermal Power Removed by He	87 MW
Total Mass Flow Rate	1830 kg/s
Pumping Power	97 MW
Maximum Local FS Temperature at FW	654°C
Radially Averaged FS Temperature at T_{max} Location	596°C

Divertor Study



- **Good progress on engineering design and analysis (but further progress much needed on physics study).**
- **Also integration with blanket needs further study, including more detail on coolant piping layout.**
- **Plan for experimental verification of thermal-hydraulic performance (Georgia Tech.)**





Divertor	
Divertor T-Tube Unit Cell Dimensions	9 cm (tor.) x 1.6 cm (pol.)
He Inlet Temperature	580°C
He Outlet Temperature	730°C
He Inlet Pressure	10 MPa
Typical FW Channel Dimensions	0.5 mm
He Jet Velocity	200 m/s
Average Jet Flow Heat Transfer Coefficient	~ 17,000 W/m^2-K
He Pressure Drop	0.4 MPa
Fusion Thermal Power in Divertor	237 MW
He Friction Thermal Power in Divertor	26 MW
Total Mass Flow Rate	338 kg/s
Pumping Power	28 MW
Maximum W Alloy Temperature	<1300°C
Maximum Primary + Secondary stresses	<370 MPa

Data for System Code

Dual Coolant Blanket Pumping Power
(add ~28 MW for divertor pumping power)

Max. Neutron Wall Load	Cycle Efficiency	Blanket Pumping Power (MW)
1.5 MW/m^2	0.441	48
2 MW/m^2	0.436	53
3 MW/m^2	0.432	74
4 MW/m^2	0.423	85
5 MW/m^2	0.414	97

Self-Cooled Pb-17 Li with SiC_f/SiC
(In-reactor pumping power is negligible)

Max. Neutron Wall Load	Cycle Efficiency
2 MW/m^2	0.63
3 MW/m^2	0.61
4 MW/m^2	0.59
5 MW/m^2	0.56

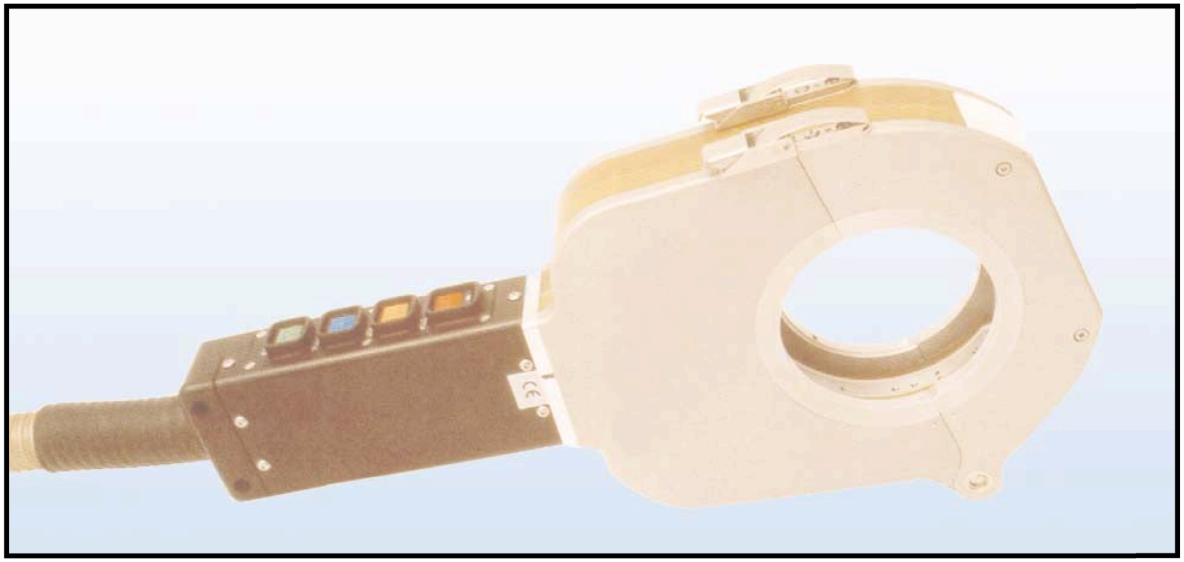
Web Search for Orbital Tool for Pipe Cutting and Rewelding from Outside

- Several off-the-shelf orbital tools available from different companies
  e.g. Astro Arc Polysoude, Magnatech and Pro-fusion (shown here)

- In communication with them to determine possibility for larger pipe size ~ 30 cm and the clearance requirements (Siegfried's e-mail)

7000 Series Orbital Weldheads

(For 4½" and 6½" Tube)



- High Duty Cycle due to superior internal, weldhead body and rotor water cooling (U.S. Patented)
- Remote Control built-in to the Handle
- Quick-change collets that require no tools for replacement
- Very rugged construction: mechanical and thermal protection
- 15 degree slant for easier loading/unloading and tungsten setting
- Collets for nearly all standard elbows and fittings

Part #	Item and Description	Tube Size Range	Pipe Size Range
PF-7004	4½" Orbital Weldhead for MK Power Supplies	1" to 4½"	1" to 4"
PF-7006	6½" Orbital Weldhead for MK Power Supplies	1" to 6½"	1" to 6"

Pro-Fusion Technologies, Inc.
3825 Old Conejo Road, Newbury Park, CA 91320 USA
Tel. (805) 376-8021 Fax (805) 376-0619
Web site: www.Pro-FusionOnline.com

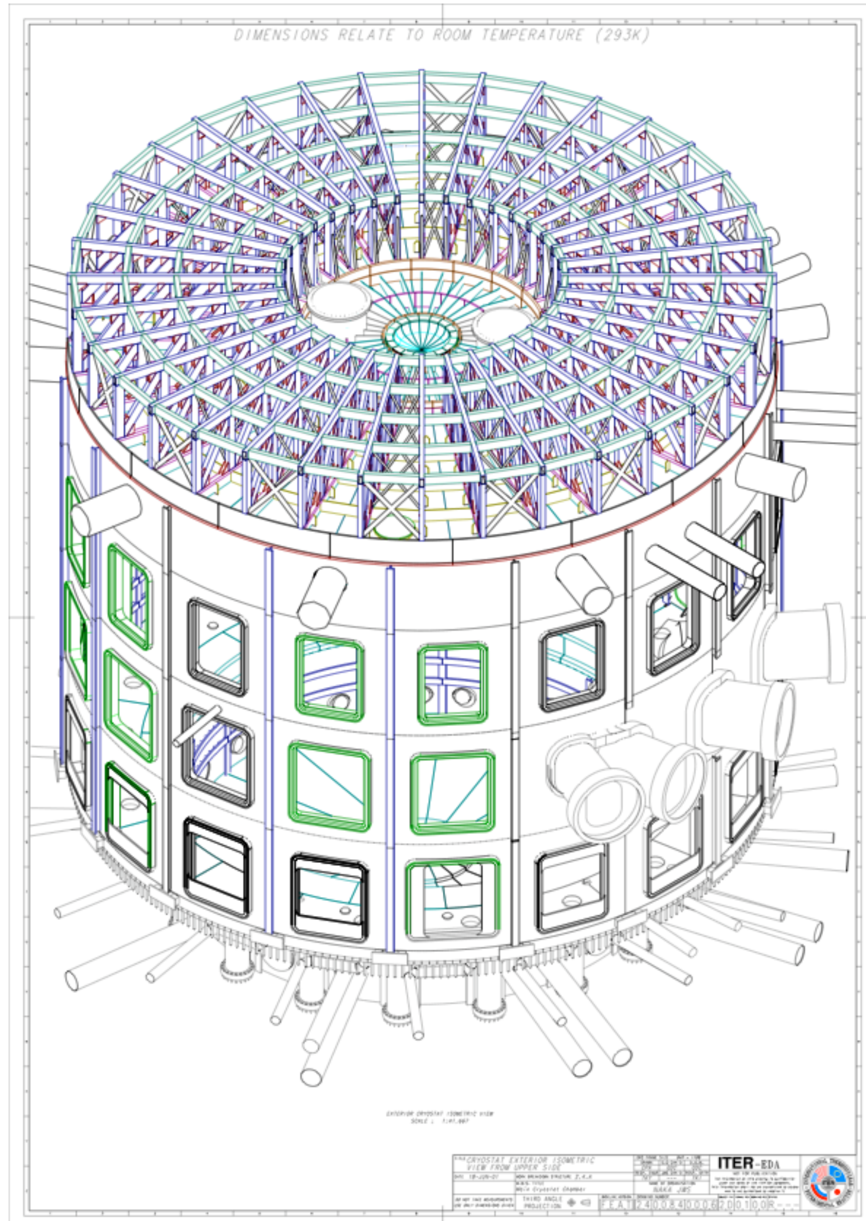
Cryostat Size and Configuration: ITER Experience

- **Main functions:**
  - Provide the vacuum insulation environment for the operation of the superconducting coils
  - Provide the second boundary for the confinement of the radioactive inventory inside the ITER VV

- **Design for 0.1 MPa external pressure and 0-0.2 MPa internal pressure.**

- **Shell + ribs + structure around penetrations.**

- **For cylindrical radius of 14.2 m:**
  - Min. Shell thickness ~ 1.2 cm based on internal pressure and ~3.2 cm based on external pressure.

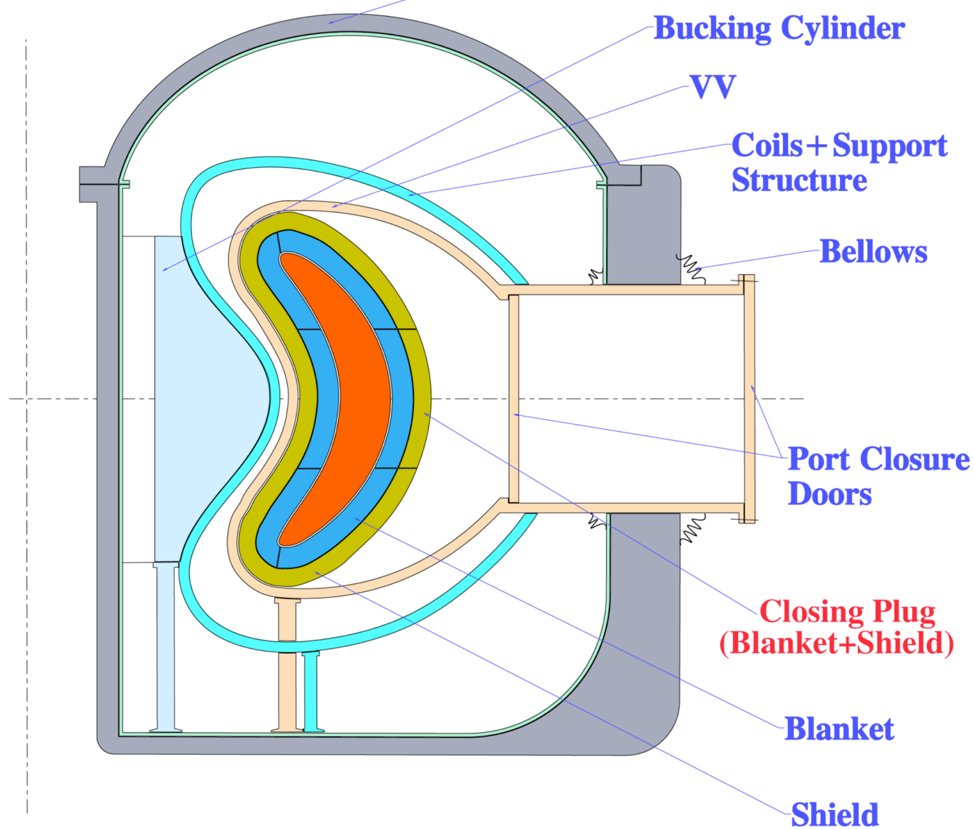


Cryostat Size and Configuration for Port-Based Maintenance

- **Simple hoop stress calculations based on 0.3 MPa internal pressure (conservative based on B. Merrill's estimates)**
  - Max. allow. stress = 115 MPa
  - Assumed minor radius of cryostat shell ~10 m
  - Min. thickness ~ 0.3 x 10/115 ~ 0.026 m

- **In addition, can we attach the shell to the biological shield to minimize the load requirement of the cryostat?**

- **What would be reasonably conservative for the system code? ~ 3-5 cm**



Work Documentation

- **Updated reference write-ups consistent with latest design parameters (system) which can be used in presentations and publications**
  - **Maintenance, dual coolant blanket, divertor and alpha particle (René)**
  - **Neutronics (Laila)**
  - **Safety (Brad)**
  - **Magnet design and material (Leslie)**

- **Work described in several publications over the last year, including:**

1. **F. Najmabadi, A. R. Raffray, L-P Ku, S. Malang, J. F. Lyon and the ARIES Team, "Exploration of Compact Stellarators as Power Plants: Initial Results from the ARIES–CS Study," presented at the 47th Meeting of the APS Plasma Physics Division, Denver, CO, October 2005, to appear in *Physics of Plasma*, 2006.**
2. **F. Najmabadi, A. R. Raffray, and the ARIES Team, "Recent Progress in ARIES Compact Stellarator Study," presented at the 15th International Toki Conference on Fusion & Advanced Technology, Toki Gifu, Japan, December 2005, to be published in *Fusion Engineering & Design,* 2006.**
3. **L. El-Guebaly, P. Wilson, D. Paige, ARIES Team, Z-Pinch Team, "Evolution of Clearance Standards and Implications for Radwaste Management of Fusion Power Plants," *Fusion Science & Technology*, Vol. 49 (1) 62-73, January 2006.**
4. **A.R. Raffray , L. El-Guebaly, S. Malang, F. Najmabadi, X. Wang and the ARIES Team, "Major Integration Issues in Evolving the Configuration Design Space for the ARIES-CS Compact Stellarator Power Plant," *Fusion Engineering & Design* , in press, 2006.**
5. **T.K. Mau, H. McGuinness, A. Grossman, A. R. Raffray , D. Steiner and the ARIES Team, "Divertor Heat Load Studies for Compact Stellarator Reactors," to appear in Proceedings of the 21th IEEE/NPSS Symposium on Fusion Engineering, Knoxville, TN, September 26-29, 2005.**
6. **T. Ihli, A. R. Raffray , and the ARIES Team, "Gas-Cooled Divertor Design Approach for ARIES-CS," to appear in Proceedings of the 21th IEEE/NPSS Symposium on Fusion Engineering, Knoxville, TN, September 26-29, 2005.**
7. **X. R. Wang, S. Malang, A. R. Raffray , and the ARIES Team, "Modular Dual Coolant Pb-17Li Blanket Design for ARIES-CS Compact Stellarator Power Plant," to appear in Proceedings of the 21th IEEE/NPSS Symposium on Fusion Engineering, Knoxville, TN, Sept. 26-29, 2005.**
8. **L. El-Guebaly, R. Raffray , S. Malang, J.F. Lyon, L.P. Ku, and the ARIES Team, , "Benefits of Radial Build Minimization and Requirements Imposed on ARIES Compact Stellarator Design," *Fusion Science & Technology* ,47 (3), 432-439, April 2005.**
9. **L. El-Guebaly, P. Wilson, D. Paige, the ARIES Team, "Initial Activation Assessment for ARIES Compact Stellarator Power Plant, *Fusion Science & Technology*, Vol. 47 (3), 640-444, April 2005.**
10. **Mengkuo Wang, Timothy J. Tautges, Douglass L. Henderson, Laila El-Guebaly, Xueren Wang, "Three-Dimensional Modeling of Complex Fusion Devices Using CAD-MCNPX Interface, "*Fusion Science & Technology* ,47 (4), 1079-4183, May 2005.**
11. **J. F. Lyon et al., "Optimization of Stellarator Reactor Parameters, "*Fusion Science & Technology*, Vol. 47 (3) 414-421, April 2005.**
12. **A. R. Raffray , S. Malang, L. El-Guebaly, X. Wang, and the ARIES Team, "Ceramic Breeder Blanket for ARIES-CS," *Fusion Science & Technology* ,47 (4), 1068-1073, May 2005.**
13. **A. R. Raffray , L. El-Guebaly, S. Malang, X. Wang, and the ARIES Team, "Attractive Design Approaches for a Compact Stellarator Power Plant," *Fusion Science & Technology* ,47 (3), 422-431, April 2005.**
14. **X.R. Wang, S. Malang, A. R. Raffray , and the ARIES Team, "Maintenance Approaches for ARIES-CS Power Core," *Fusion Science & Technology* ,47 (4), 1074-1078, May 2005.**

ARIES-AT Special Issue of Fusion Engineering & Design Published, January 2006

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