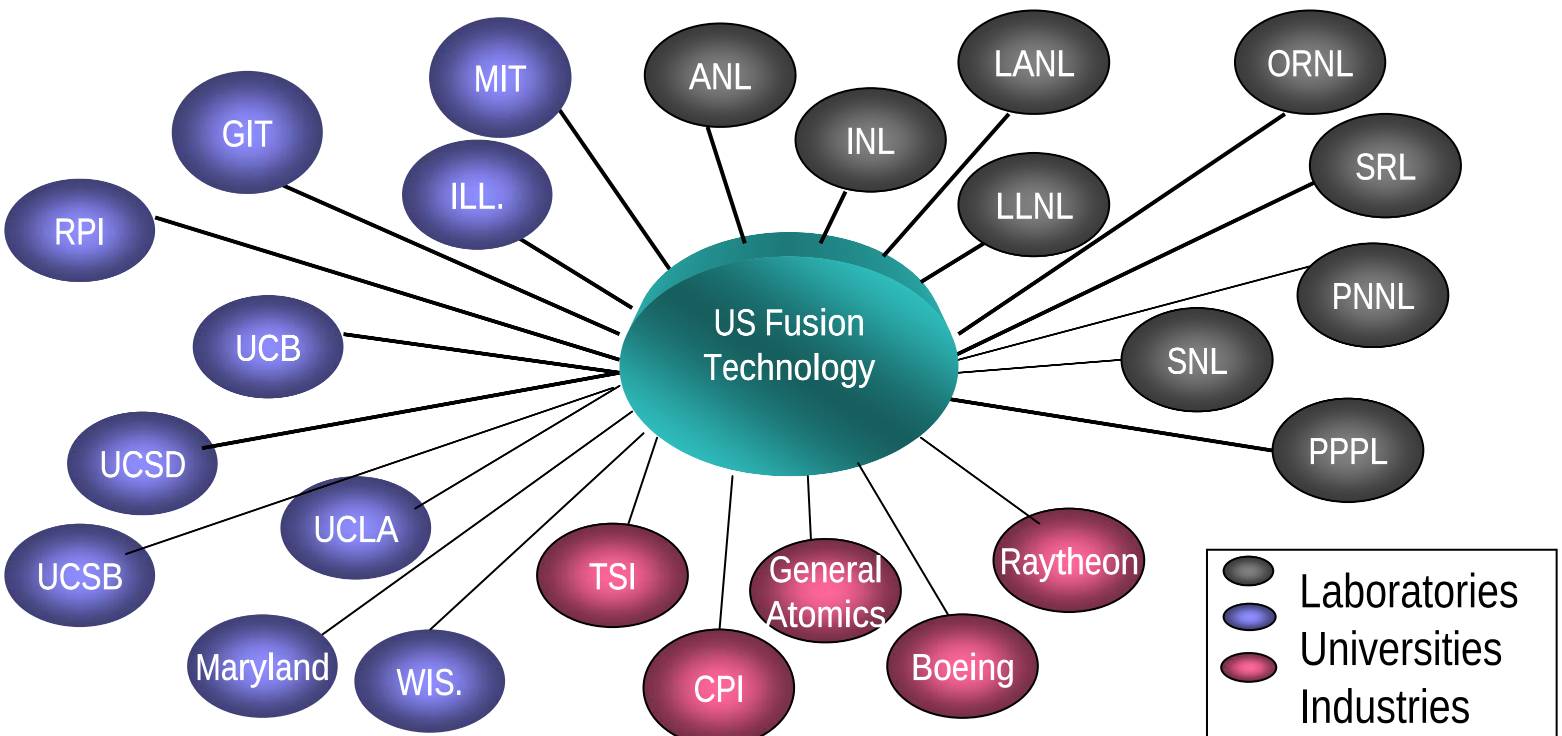
Overview of Fusion Nuclear Technology in the US

Neil B. Morley

University of California, Los Angeles

Presented at:
The 7th International Symposium on Fusion Nuclear Technology
Tokyo, Japan, May 2005



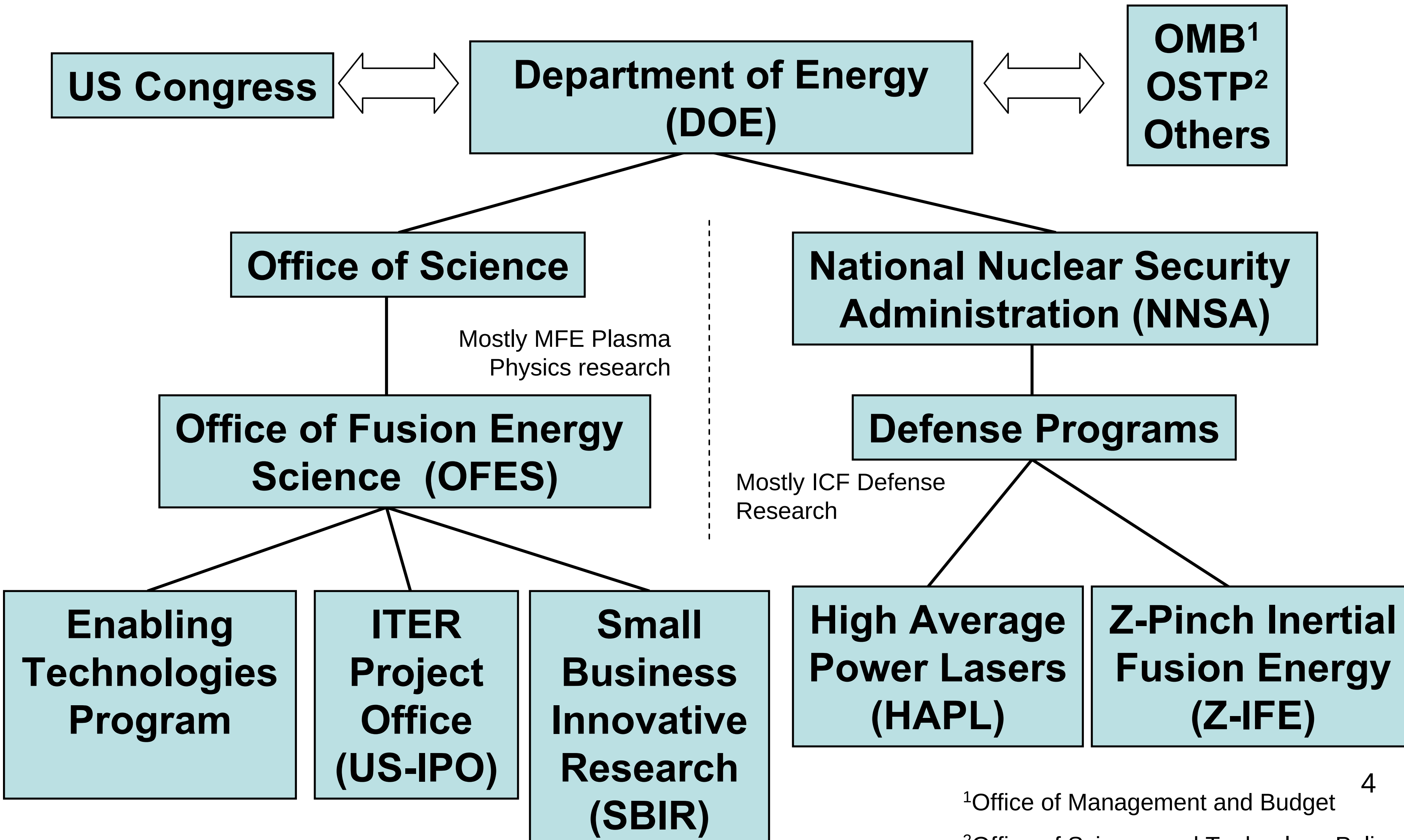
Grateful Acknowledgements

- Contributors and co-authors: M.A. Abdou, A.Y. Ying, P. Calderoni, R. Raffray, S. Willms, R.J. Kurtz, M. Sawan, M. Anderson, R. Nygren, S. Smolentsev, P. Sharpe
- Stan Milora – and contributors to the VLT presentation, US Budget Planning Meeting, March 2005
- N. Sauthoff – US ITER Project Office
- C. Olson and colleagues at SNL – Z-IFE
- J. Sethian and HAPL contributors – HAPL

Outline of the Presentation

- Fusion Research Organization in the US
- Enabling Technologies / VLT Program
  - ITER Test Blanket Module Program
  - JUPITER-2 collaboration with Japan
  - Materials Research
  - Plasma Facing Components Research
  - ARIES Design Studies
  - Neutronics Simulation Tools
- ITER Project Office and US Contributions to ITER
  - First wall / Shield Module 18
  - Tokamak Exhaust Plant
- IFE Technology Research
  - High Average Power Laser
  - Z-Pinch Program
- Summary and Outlook

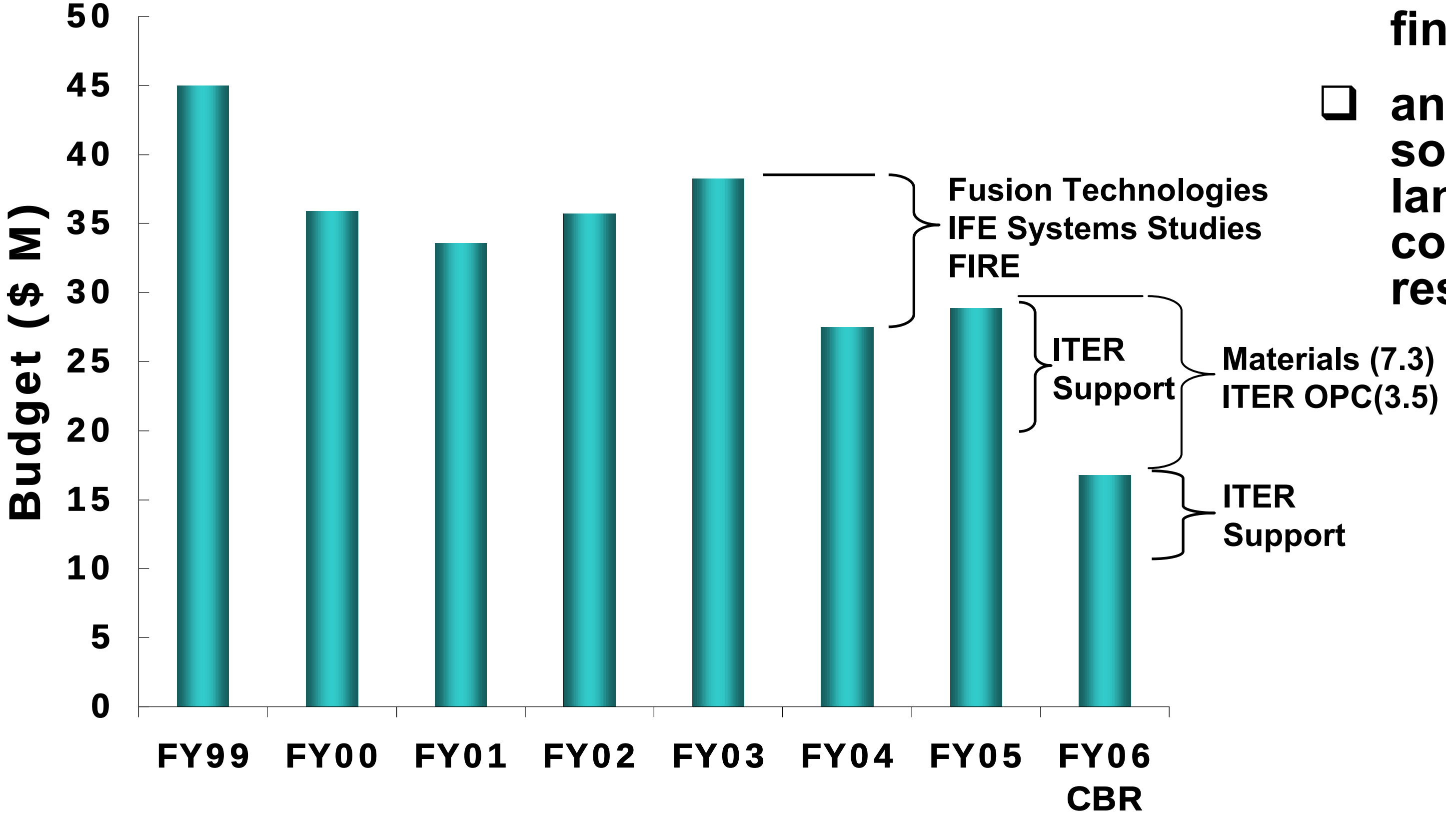
Fusion Nuclear Technology Research Organization in the US



[1]Office of Management and Budget

[2]Office of Science and Technology Policy

Enabling Technology budget erosion and redirection, targeted at longer range technology programs, is a serious concern as US rejoins ITER effort



- FY06 not yet finalized,
- and there is some good language from congress about restoring effort

Enabling Technologies is coordinated by the Virtual Laboratory for Technology - VLT

Director	**S. Milora**
Deputy Director	**D. Petti**

	Program Element	**Element Leader**
Main FNT Research Programs	**Plasma Chamber (Blanket)**	**M. Abdou - *UCLA***
	Safety & Tritium	**D. Petti - *INEEL***
	Materials	**S. Zinkle - *ORNL***
	Plasma Facing Components	**M. Ulrickson - *SNL***
	Tritium Processing	**S. Willms – *LANL***
	ARIES	**F. Najmabadi – UCSD**
	NSO/FIRE	**D. Meade - *PPPL***
	ICH	**D. Swain - *ORNL***
	ECH	**R. Temkin - *MIT***
	Fueling	**S. Combs – *ORNL***
	Magnets	**J. Minervini - MIT**
	Socio-Economic	**J. Schmidt - *PPPL***

Outline of the Presentation

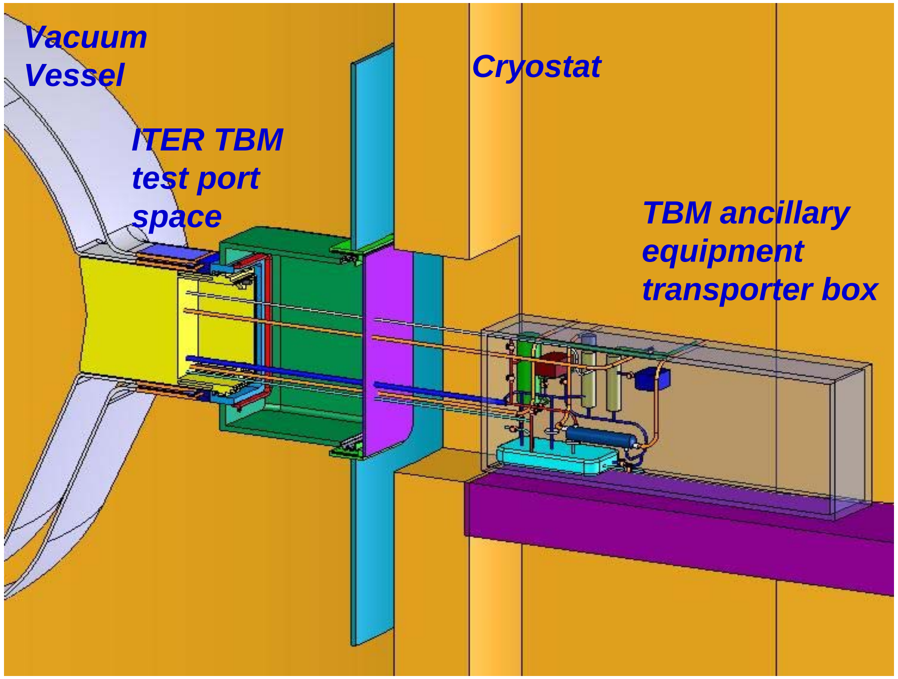
- ❑ Fusion Research Organization in the US
- ❑ Enabling Technologies / VLT Program
  - ITER Test Blanket Module Program
  - JUPITER-2 collaboration with Japan
  - Materials Research
  - Plasma Facing Components Research
  - ARIES Design Studies
  - Neutronics Simulation Tools
- ❑ ITER Project Office and US Contributions to ITER
  - First wall / Shield Module 18
  - Tokamak Exhaust Plant
- ❑ IFE Technology Research
  - High Average Power Laser
  - Z-Pinch Program
- ❑ Summary and Outlook

US Considers Test Blanket Module Program as an Important Utilization of ITER

Integrated experiments on first wall and breeding blanket components and materials in a **Fusion Environment** are a key element of the ITER Mission.

Role of TBM program in the US:

- ❑ Help determine the conditions governing scientific feasibility of the D-T cycle, *i.e.* the **“phase-space”** window of plasma, nuclear, material, and technological conditions in which tritium self-sufficiency can be attained
- ❑ Develop the technology necessary to
  - install breeding capabilities to supply ITER with tritium for any extended phase of operation
  - solve the critical “tritium supply” issue for fusion development beyond ITER



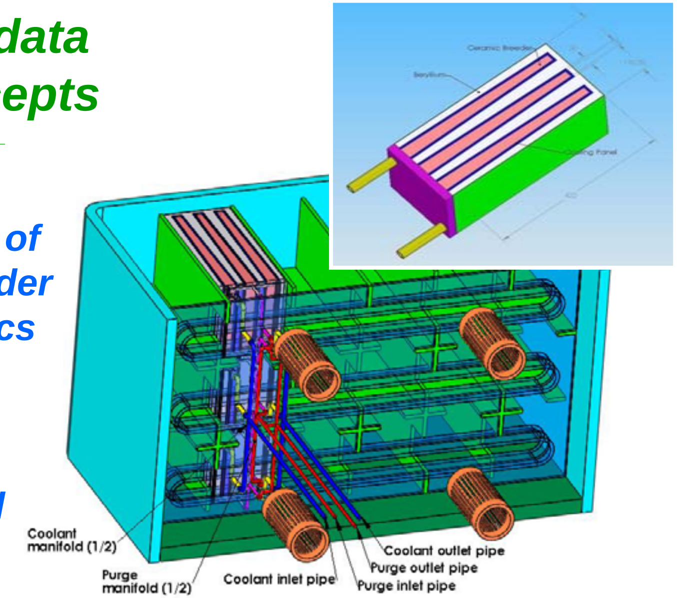
- ❑ **TBM testing starts from Day One of ITER operation**
- ❑ **ITER’s construction plan includes specifications for TBMs because of impacts on space, vacuum vessel, maintenance, equipment, safety, availability, etc.**

US Selected Options for ITER TBM

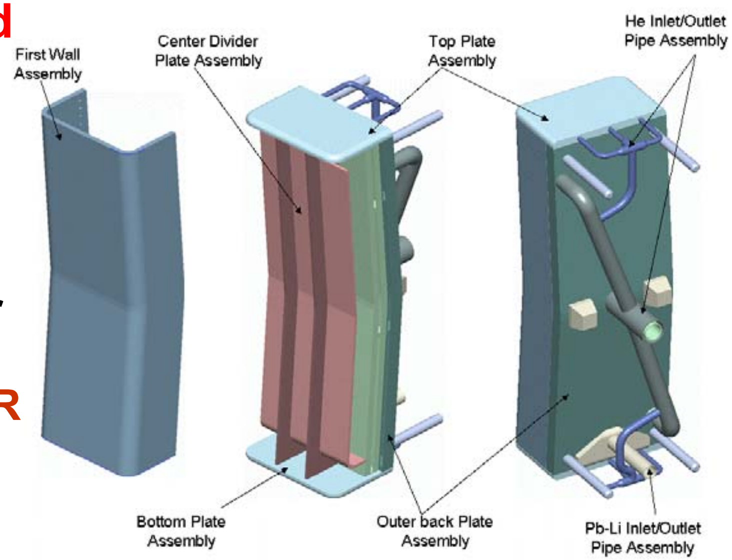
The conclusion of the US community, based on the results of a technical assessment of the available data and analyses to date, is to select two blanket concepts for the US ITER-TBM with the following emphases

- **A helium-cooled solid breeder concept with ferritic steel FW heat sink and blanket structure and beryllium neutron multiplier**

Schematic view of three solid breeder thermomechanics unit cell test articles housed inside the EU's HCPB structural box



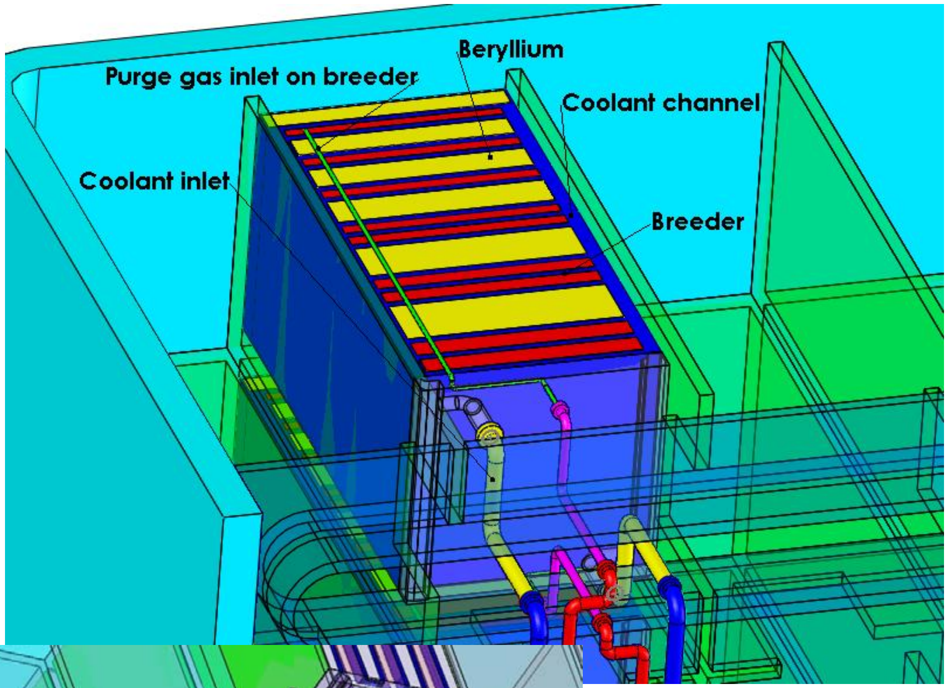
- **A Dual-Coolant Pb-Li liquid breeder blanket concept with self-cooled LiPb breeding zone and flow channel inserts (FCIs) as MHD and thermal insulator**



Dual Coolant Lead-Lithium TBM Views showing complete structure (right), internal poloidal channels (middle) and first wall assembly (left)

A. Ying et al. "Overview of US ITER TBM Program" – Tuesday Morning

US helium-cooled solid breeder test blanket R&D program focused on pebble bed thermomechanics and tritium issues

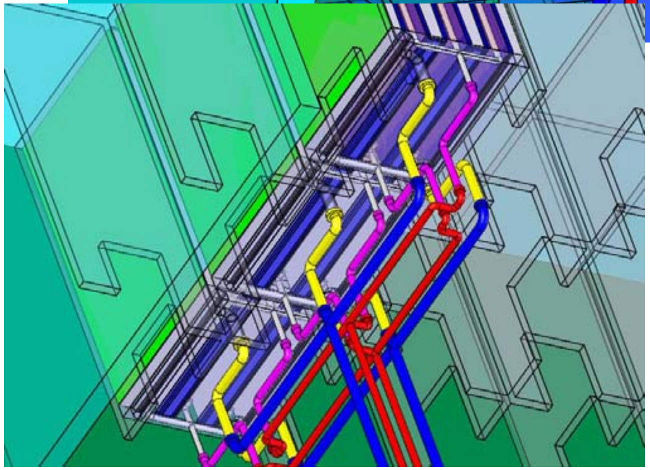


Proposed collaboration involves inserting three "US" unit cells into the EU HCPB structural box

A. Ying et al. "Solid breeder test blanket modules design and analysis" Tuesday Morning

P. Calderoni et al. "Experimental study of interaction of ceramic breeder pebble beds with structural material under thermo-mechanical loads" – Monday Afternoon

M-J. Ni et al. "2D and 3D models for tritium permeation in solid breeder blanket units – Wed. Morning



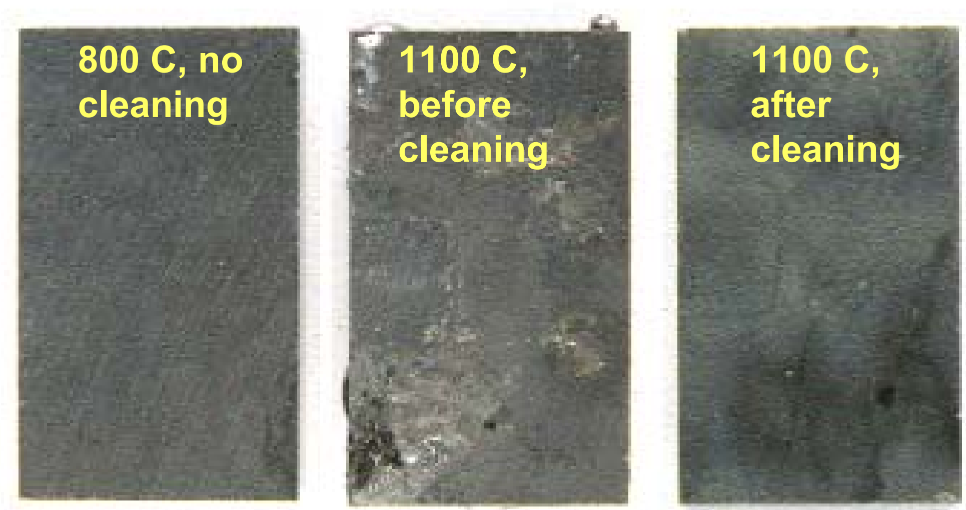
Cooling to each unit cell is done by unit cell array manifold

Unique DCLL R&D issues include MHD effects and FCI/PbLi compatibility

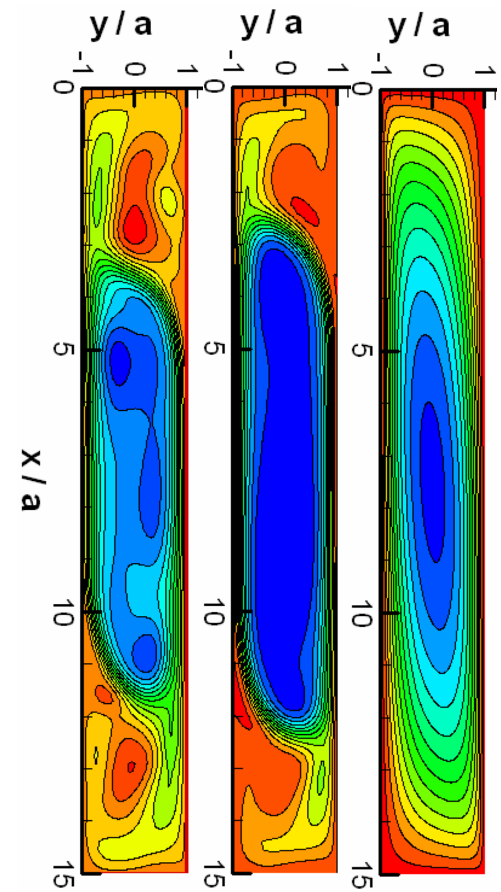
Main Issues

- MHD strongly influence pressurization, heat transfer, corrosion, tritium permeation and ultimate design
- Fabrication, properties and reliability of FCIs
- PbLi must be compatible with FCI material (currently SiC) at ~700-800C

C. Wong et al. "Overview of dual-coolant Pb-17Li breeder first wall and blanket concept development for the US ITER TBM design" – Tuesday Morning



Images of SiC sample surface after exposure

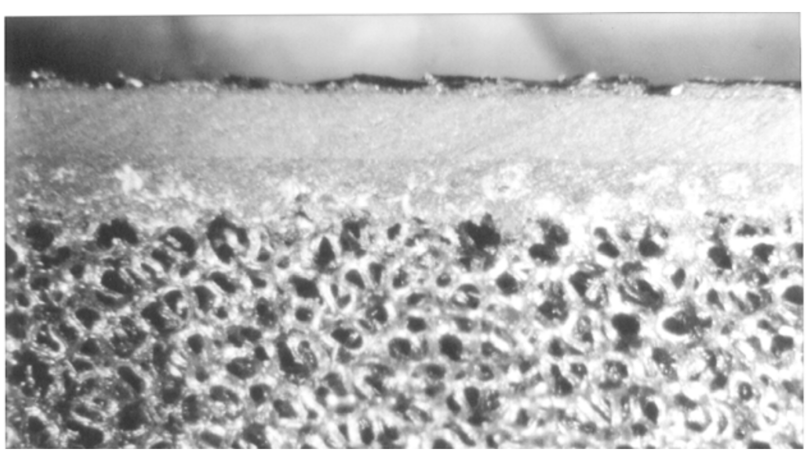


Effect of increasing magnetic field strength on thermal convection driven flow – overall motion is reduced, but not suppressed under reactor conditions

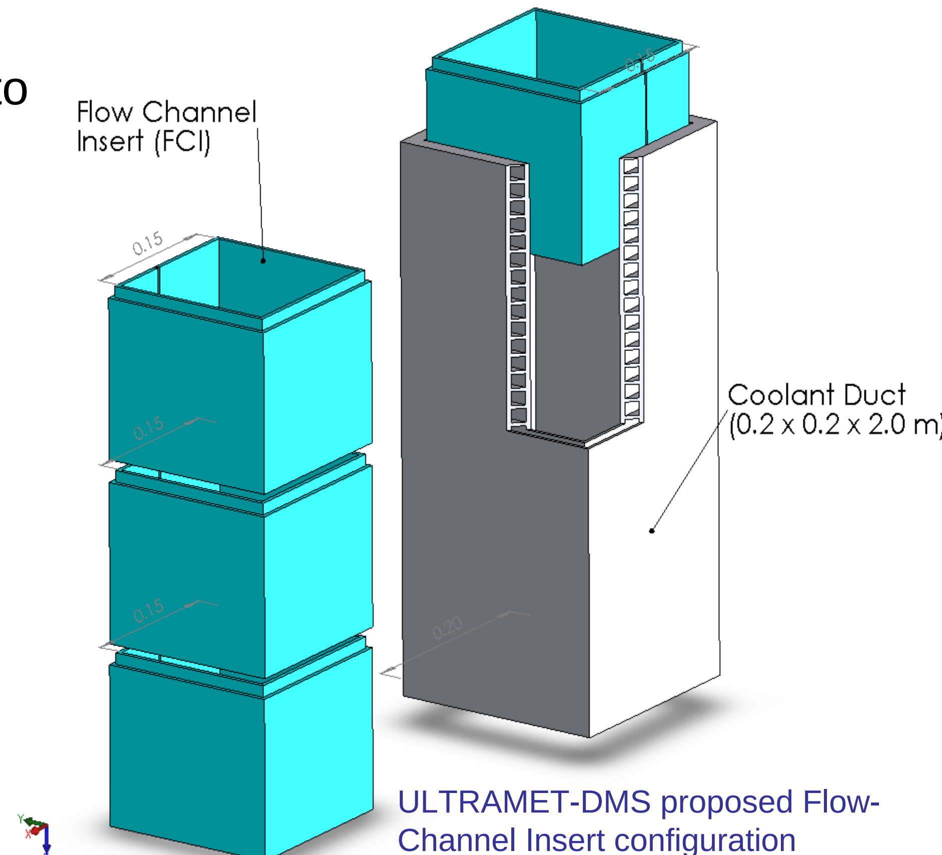
Small Business Innovative Research grants supplement effort in many FNT areas – Like SiC development for DCLL Flow Channel Inserts

Ultramet SBIR looking at feasibility of low density SiC foam cores with impermeable SiC CVD facesheets for DCLL FCIs

- Improved manufacturability compared to SiC/SiC composites
- High strength, stiffness, and thermal stress resistance
- Low thermal and electrical conductivity



CVD SiC closeout layer applied to the surface of SiC foam (20X)



ULTRAMET-DMS proposed Flow-Channel Insert configuration

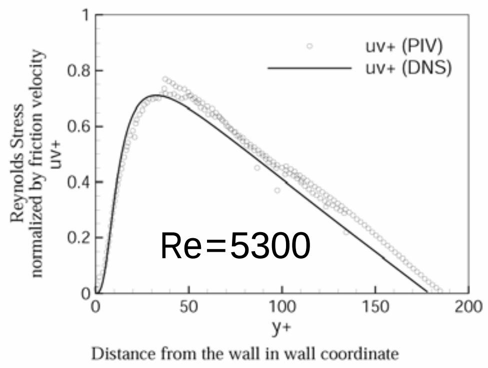
JUPITER-2 Collaboration between US and Japanese Universities beginning 5th year

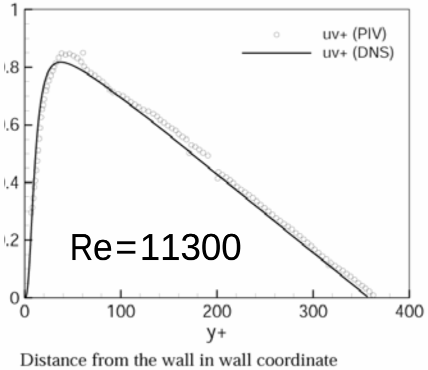
Task 1: Advanced Liquid-Cooled Blankets & Materials	**Task 1-1-A: Flibe Tritium Chemistry & Safety**	INL
	Task 1-1-B: Flibe Thermofluid MHD	UCLA
	Task 1-2-A: MHD Coatings for V/Li Systems	ORNL
	Task 1-2-B: V Capsule Irradiation	PNNL
Task 2: Advanced Gas Cooled Blankets & Materials	**Task 2-1: SiC Fundamental Issues, Fabrication and Materials Supply**	ORNL
	Task 2-2: SiC System Thermomechanics	UCLA
	Task 2-3: SiC Capsule Irradiation	ORNL
Task 3: Advanced Simulation	**Task 3-1: Design-based Integration Modeling**	UCLA
	Task 3-2: Materials Systems Modeling	UCLA

Jupiter-2 MHD Thermofluid experiments exploring effect of strong magnetic fields on Molten Salt turbulence and convective heat transfer

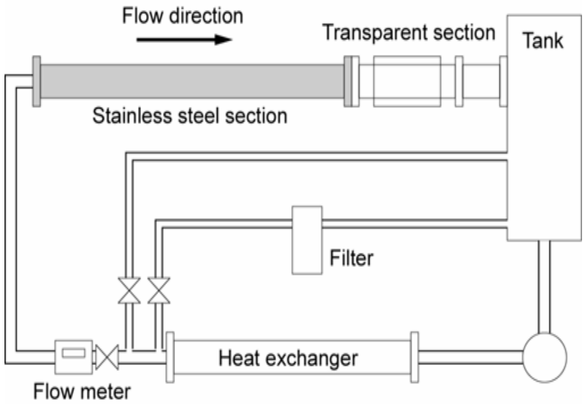
- **Completed!**
  - **benchmarking phase with measurement comparison to theory**
  - **facility transition to MHD operation with installation of 2T gap magnet**
- **MHD turbulence and turbulent heat transfer experiments underway**

J. Takeuchi et al. "Study of turbulent flow and heat transfer for molten salt simulants in a large diameter circular pipe" - Monday afternoon



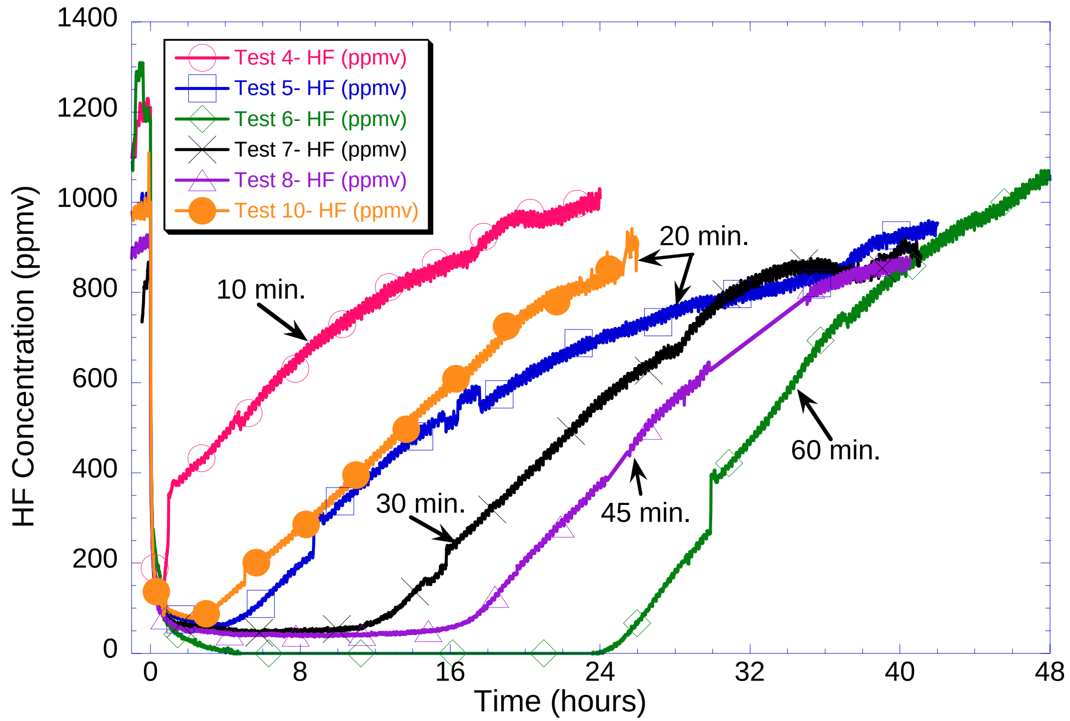


2T Open-Top gap magnet installed in UCLA FLIHY-2 loop for MHD turbulence experiments



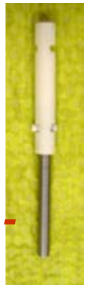
Good agreement between J2 data and intensive DNS simulations

Flibe REDOX control and thermochemistry kinetics/corrosion studies indicate significant amount of dissolved Beryllium in Flibe melt that is very effective at reduction of HF stream



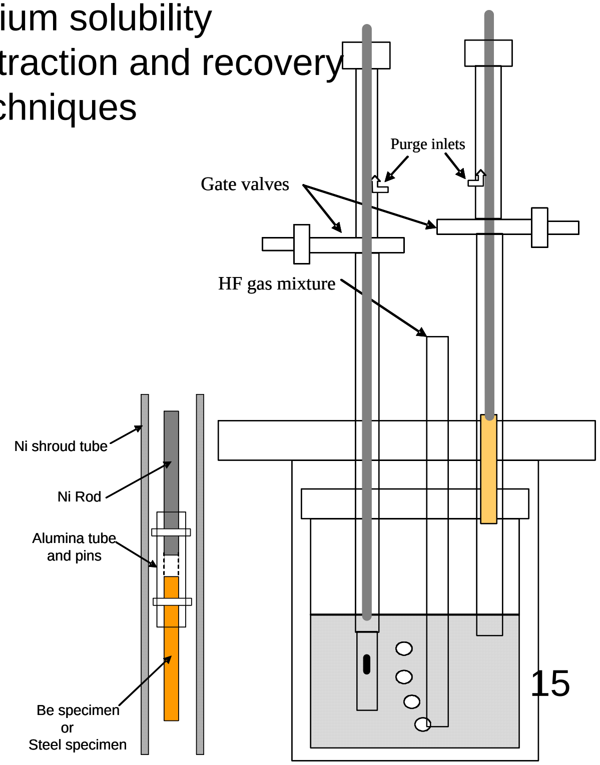
Time-to-return to original HF concentration after immersion/removal of Be rod in Flibe melt

D Petti et al. "Update on Jupiter-II molten salt flibe tritium chemistry and safety experimental program" - Tuesday Afternoon



Planning of next tritium experiments

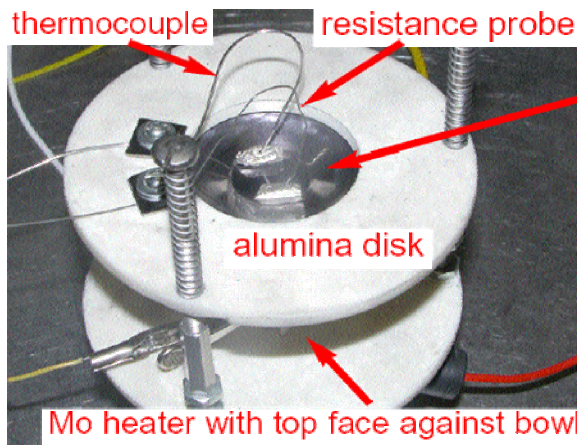
- REDOX control at very low concentrations of TF
- tritium solubility
- extraction and recovery techniques



Jupiter-2 MHD Coatings on Vanadium

- **Li exposures of candidate MHD coatings in this in-situ resistivity measurement apparatus show coatings always short-circuited when in contact with molten lithium**
- **Initial tests with thin vanadium over-layers also showed short circuit when in contact with molten Li**
- **Considering thicker V layers and possibly flow channel inserts**

Progress: Test rig constructed in ORNL glove box
LLNL fabricated bi-layer coating: 8μm Er_2O_3/2μm V
Completed testing of LLNL and Suzuki coatings



Single & double layer MHD coatings tested Jan. 2005:

Suzuki: Er_2O_3 coatings

LLNL: 8μm PVD Er_2O_3
PVD Er_2O_3 + 2μm V

Results: - Unit tested to 600°C, significant Li evaporation >500°C
- Liquid Li forms surface reaction product after 48h at 500°C
- Without Li, coated bowl maintained high resistance at 500°C
- Initial coatings shorted when Li melted (180°C)

B. Pint et al. "Liquid metal compatibility issues for TBMs" – Thursday Morning

Jupiter-2 Irradiation experiments in HFIR

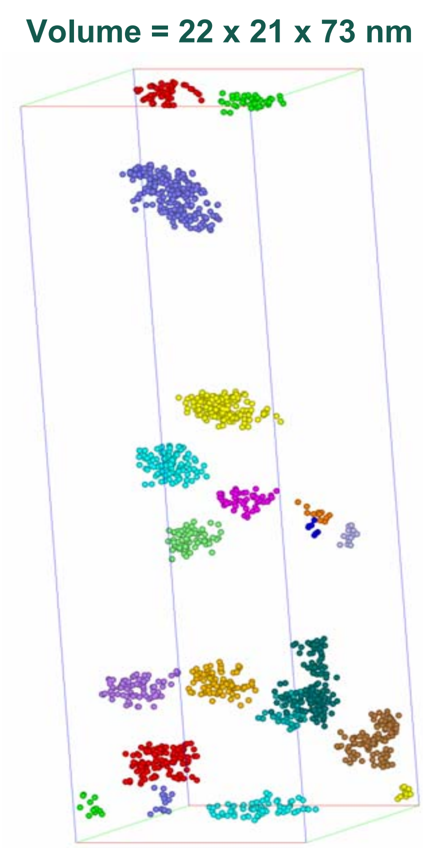
- **V/Li**
  - 17J experiment contains vanadium samples immersed in Li at temperatures of 425, 600 and 700°C.
  - The 17J experiment has completed the 5th of 9 irradiation cycles in HFIR
- **SiC and SiC/SiC composites**
  - Rabbit irradiation in HFIR demonstrated good neutron tolerance of NITE SiC/SiC composite, for the first time.
  - Experimental matrix for 18J has been finalized (to go in-pile in July), including minor modification in support of DCLL blanket FCI R&D
  - 18J experiment will provide knowledge for constitutive modeling and prediction of irradiation effect on strength, detailed fracture properties, thermal conductivity, and electrical conductivity of CVD SiC and CVI or NITE SiC/SiC composites in any architecture.
  - Irradiated compatibility of SiC and lithium ceramics will also be studied.

Vanadium specimens in basket to be immersed in molten lithium during irradiation

Other US Materials Program Research emphases, spanning near-term to long-term

- ITER in-vessel materials research, including
  - assessment of properties data and R&D for ITER design and construction
  - electrical degradation during irradiation in diagnostics & insulators
  - irradiation assisted stress corrosion cracking and fracture toughness in copper alloys
- Material compatibility, structural analysis, and low-dose neutron effects for ITER-TBM including SiC composites, PbLi, Ferritic steels
- Modeling and experiments on key physical mechanisms for flow localization in irradiated metals, which will lead to improved radiation-resistant materials
- 4th-generation radiation-resistant SiC/SiC composites utilizing advanced SiC fibers, SiC multilayer interphases, and novel matrix infiltration methods have been designed

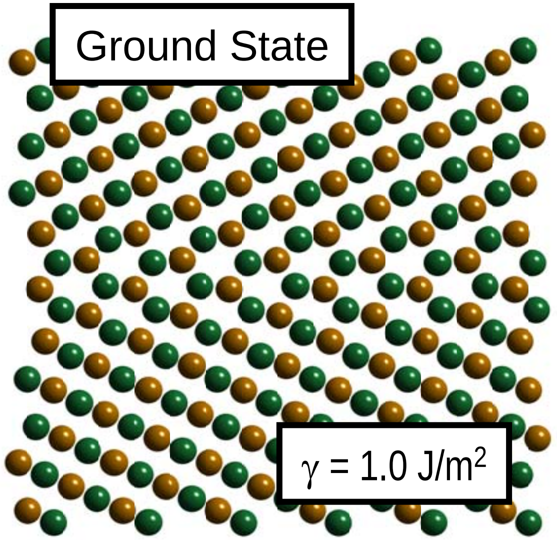
Y. Katoh et al. "Property tailorability for advanced CVI SiC composites for fusion" – Thursday Afternoon



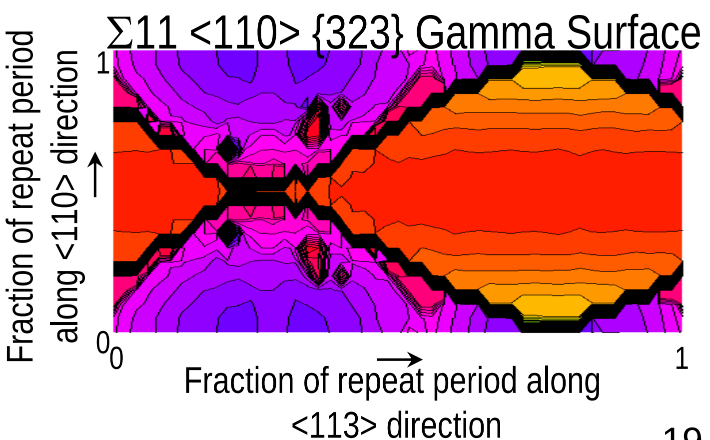
O-Ti-Y nanoclusters in ODS steel possessing long-term stability at high temperatures

Fundamental Material Science: Multi-scale modeling of He transport and fate in ferritic alloys

- The model will be used to predict the performance of irradiated conventional and ODS steels and tested by performing key He effects experiments to gather key information for model validation.
- Ultimately the validated model will be used to develop high-performance ODS steels for fusion.
- Focus is on modeling the trapping and migration of He at important microstructural features in Fe such as dislocations, grain boundaries and coherent nanoclusters.



Atomic model of a grain boundary in iron - different colors of atoms represent different atomic planes



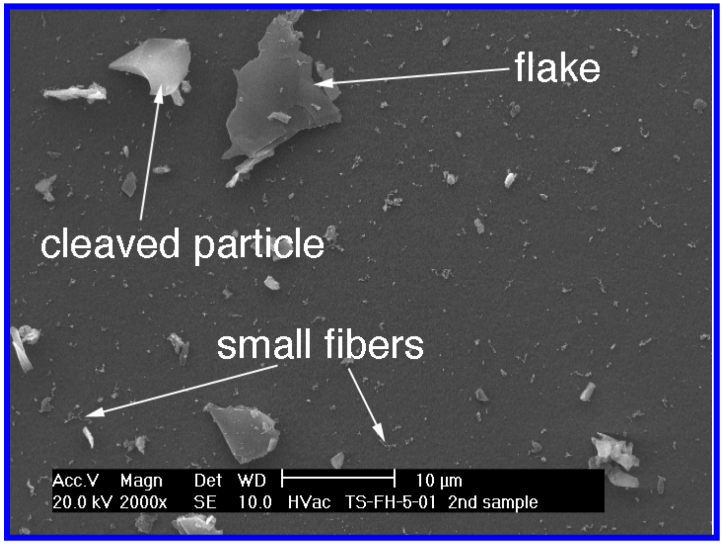
Gamma surface for this grain boundary

Plasma Facing Components Research

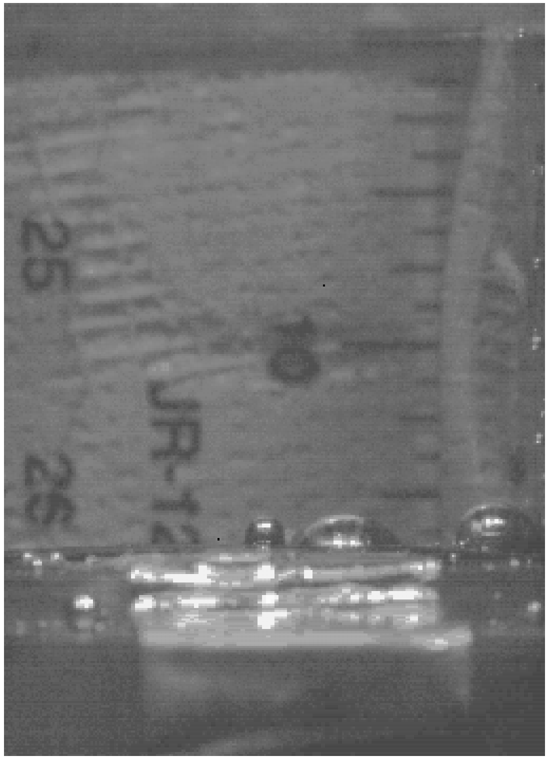
- ❑ **Solid Surface Divertors and First Walls**
  - Improved W rod tiles for C-Mod
  - ELM Testing of ITER PFCs
  - Testing of FW options for ITER Shielding Blanket and ITER TBM
  - Testing of Cu/SS heat sinks for ITER
- ❑ **Advanced Liquid Surfaces**
  - MHD 3-component field flow experiments
  - Improved modeling of liquid MHD
  - Metal PFC melt layer motion
- ❑ **Plasma Materials Interactions Exp.**
  - Tritium experiments on mixed materials
  - Mixed material erosion studies
  - Tokamak Dust
- ❑ **Plasma Materials Interactions Model**
  - Improved edge plasma and PMI codes to include convective SOL transport effects in ITER
  - Modeling of ELM and T retention experiments

M. Ulrickson, "Comparison of liquid and solid surface options for PFCs" – Wed. Morning

Dust Collection and characterization



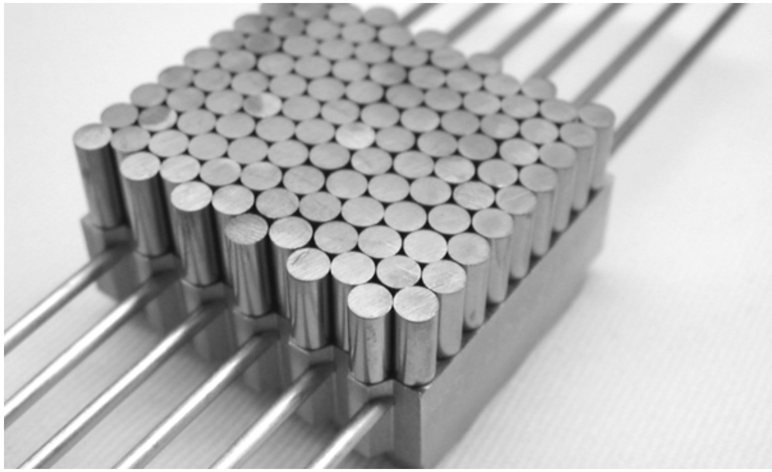
Droplet generation in magnetic field due to stray currents

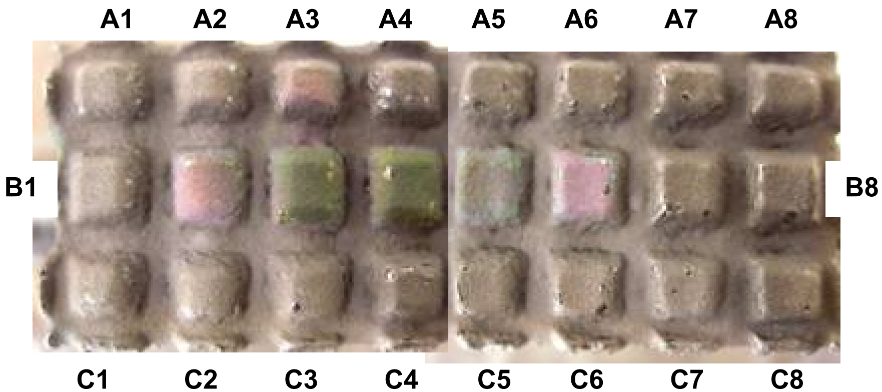


The US has a long standing interest in developing PFCs that includes plasma facing and heat sink materials, fabrication, and interaction with plasmas

Slotted DiMES experiment in DIII-D showed soft layer of Carbon deposited in the slot with a preliminary atomic C/Deuterium ratio of 0.2-0.6

Development of W rod on Inconel Divertor Tiles for CMOD including brazing and HHF testing

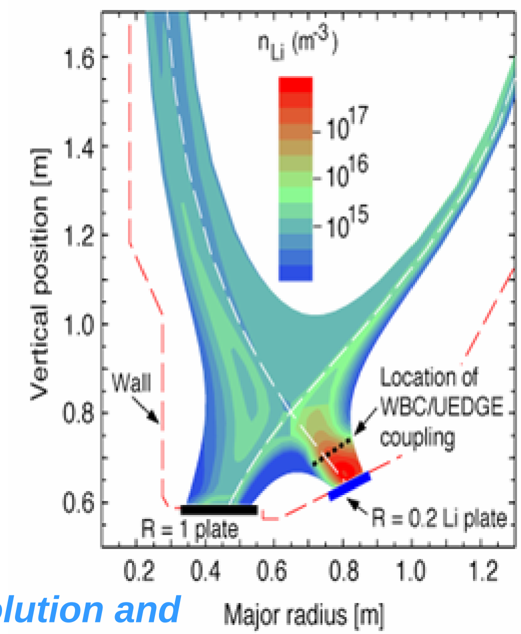




Beryllium near net shaped plasma spray on pre-castellated copper heat sink

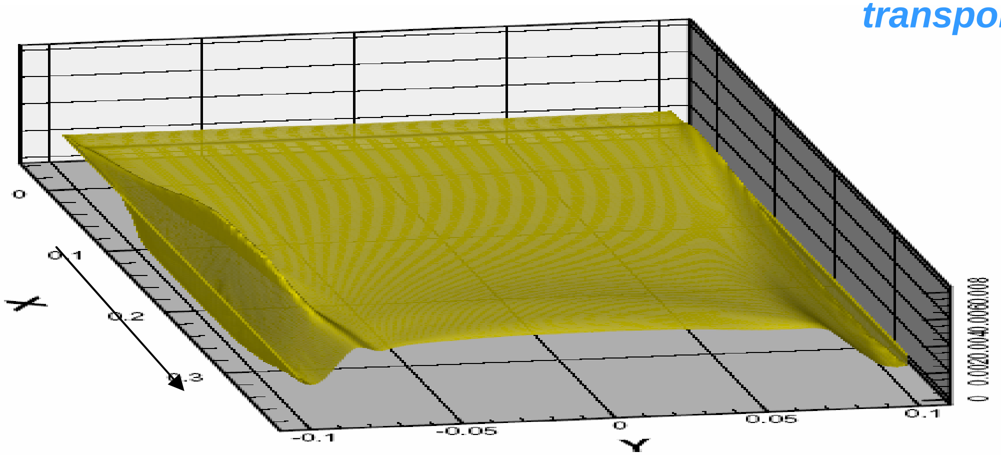
Lithium divertor particle pumping experiments planned for NSTX

- ❑ Test Stages leading to flowing Li Module
  - – Li pellet injection
  - – Module A-1: thin stagnant liquid Li layers on existing carbon tiles
  - – Module A-2: thin stagnant liquid Li layers on heated metallic tiles
  - – Module B: flowing lithium for improved particle pumping and heat removal



Calculated Li evolution and transport from a liquid Li surface on the NSTX divertor

A. Hassanien et al. "LM surfaces in future tokamak operation" – Wed. Afternoon

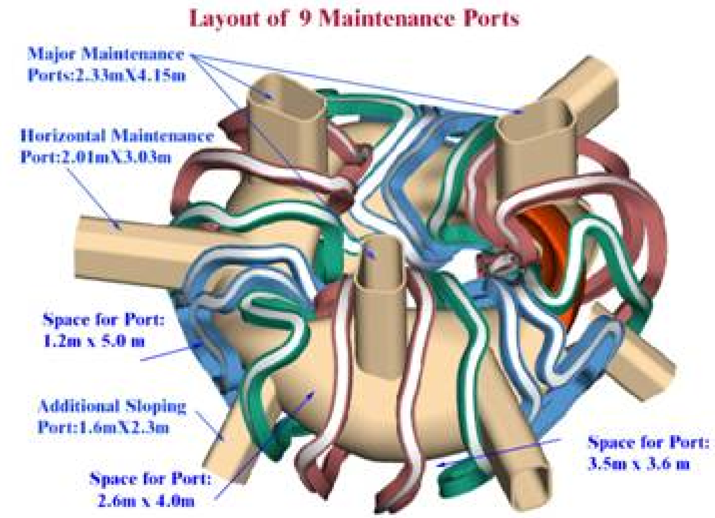


Simulation of lithium film flow in NSTX divertor fields shows separation from sidewalls, but overall acceptable flow

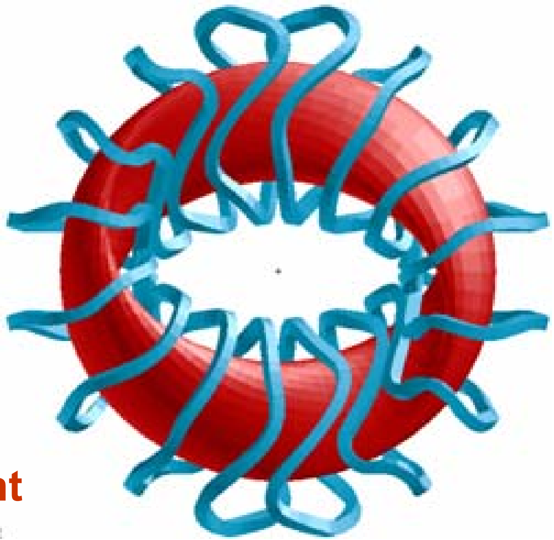
Testing of Lithium evaporator systems for producing thin Li films in NSTX

Advanced Design - ARIES Compact Stellarator Study

- The physics basis of compact stellarator power plants has been assessed. New configurations have been developed, others refined and improved, all aimed at low plasma aspect ratios (A   6), hence compact size.
- Modular coils are designed to examine the geometric complexity and the constraints of the maximum allowable field, desirable coil-plasma spacing, coil-coil spacing, etc.
- Assembly and maintenance appears to be the key issue in configuration optimization.



NCSX-Like

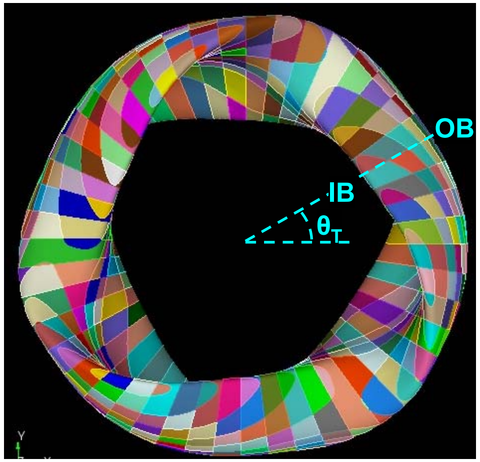


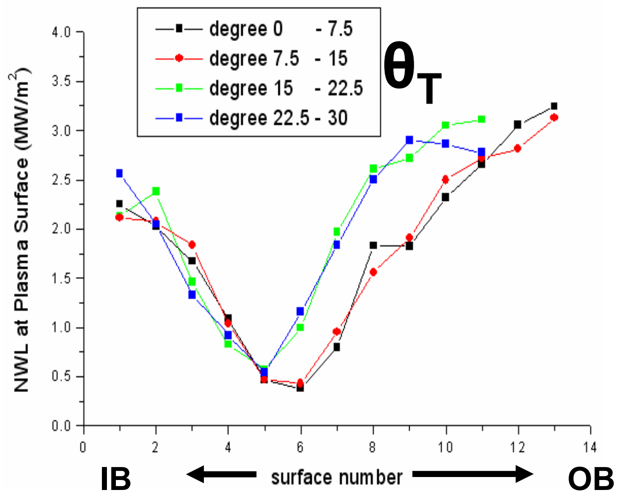
MHH2

R. Raffray et al. "Major integration issues in evolving the configuration design space for the ARIES-CS Power Plant – Thursday Morning

Improved Neutronics Simulation Capability CAD-Based MCNP Development

- **Use Sandia's Common Geometry Module (CGM) interface to evaluate CAD *directly* from MCNP**
  - » **CGM provides common interface to multiple CAD engines, including voxel-based models**
- **Benefits:**
  - » **Dramatically reduce turnaround time from CAD-based design changes**
    - – ***Identified as key element of ITER Neutronics analysis strategy***
  - » **No translation to MCNP geometry commands**
  - » **Can handle unsupported 3D models**
- **Issues/plans:**
  - » **Benchmarking the current prototype version of MCNP/CGM for ITER analyses**
  - » **Slower than MCNP alone. The focus will be to speed up the ray-tracing portion of the code**





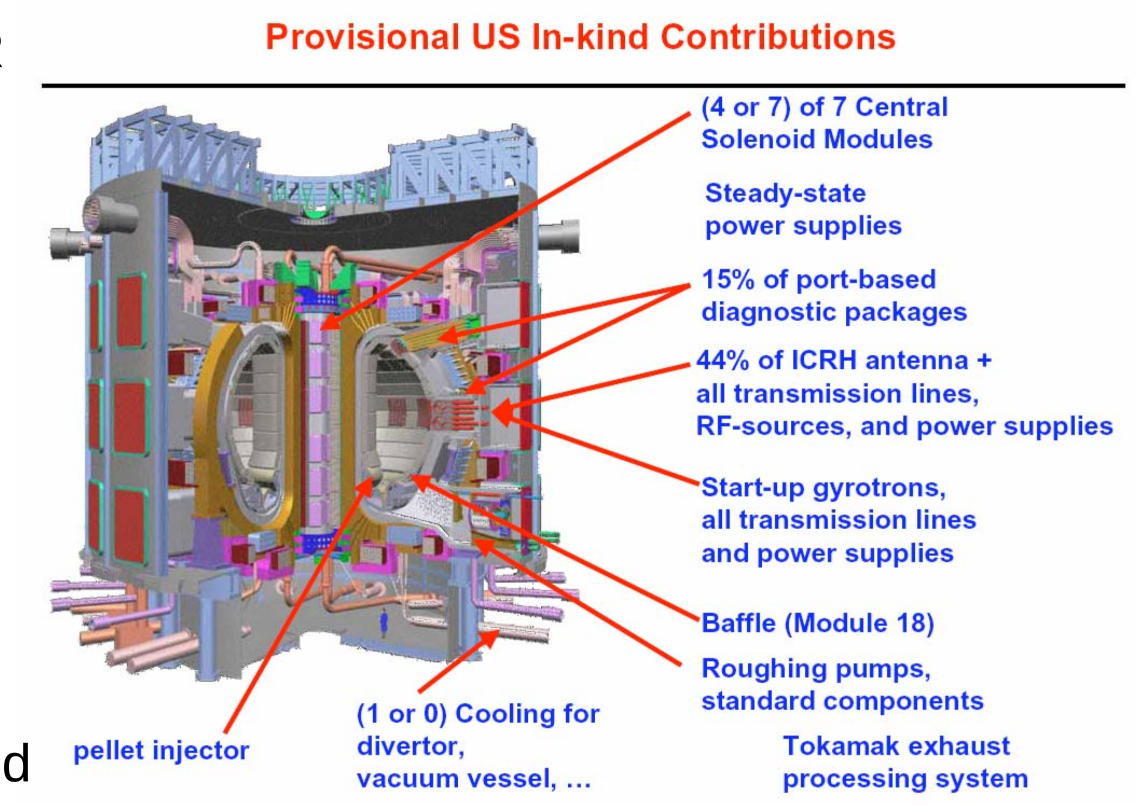
MCNP/CGM applied to complex geometry of ARIES-CS

Outline of the Presentation

- ❑ Fusion Research Organization in the US
- ❑ Enabling Technologies / VLT Program
  - – ITER Test Blanket Module Program
  - – JUPITER-2 collaboration with Japan
  - – Materials Research
  - – Plasma Facing Components Research
  - – ARIES Design Studies
  - – Neutronics Simulation Tools
- ❑ ITER Project Office and US Contributions to ITER
  - – First wall / Shield Module 18
  - – Tokamak Exhaust Plant
- ❑ IFE Technology Research
  - – High Average Power Laser
  - – Z-Pinch Program
- ❑ Summary and Outlook

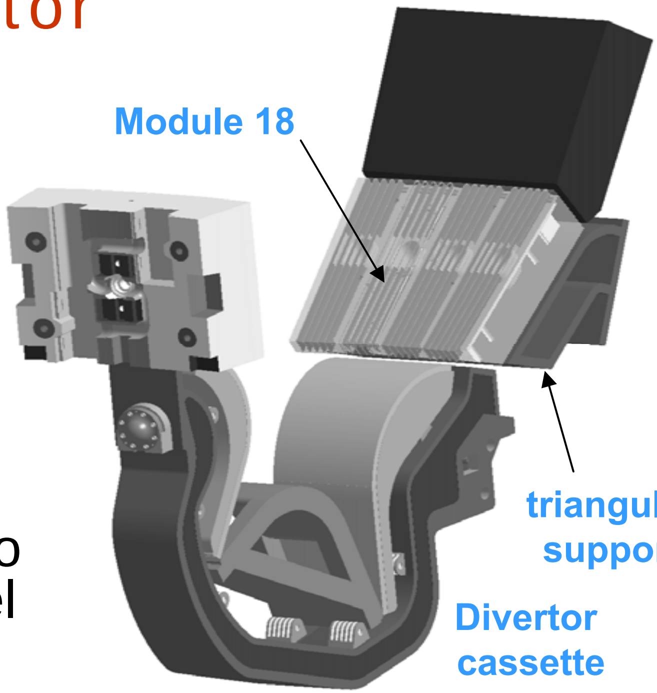
US ITER Project Office and FNT Research

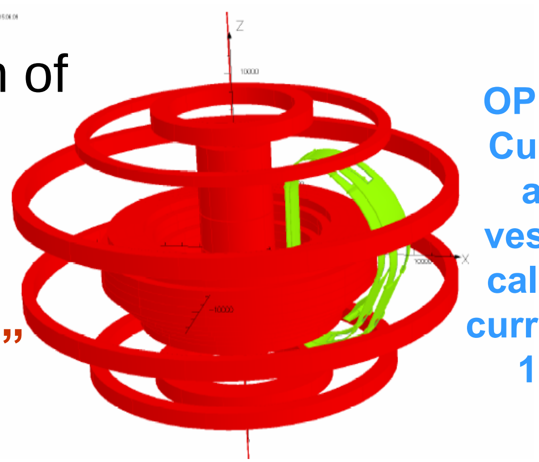
- US-ITER Project Office awarded to PPPL/ORNL consortium, Led by Ned Sauthoff
- FNT contributions to ITER by the US
  - 10% FW/Shield module, Baffle Module 18
  - Tokamak Exhaust Processing System
  - 15% of port based diagnostic packages including required plasma facing surfaces and neutron shields
  - 44% of ICRH antennae including plasma facing surfaces
- TBM program linked to and coordinated with US-IPO



The US will develop the design of ITER FW Module 18 – the lowest outboard module just above the divertor

- Mod18 is unique from other FW modules in that it
  - is mounted on the triangular support, an appendage on the vacuum vessel wall
  - is thinner (400 vs. 450mm) than other modules, has various port penetrations
  - has part of its lower surface in addition to the front face is exposed to the plasma.
- The FW's CuCrZr heat sink must be joined to beryllium armor, and internal cooling channel liners and a return manifold of 316LN-IG.
- A key issue is eddy current control and determination of the number and position of cuts in the metal block.
  - Model development and analysis is underway with the OPERA® code





OPERA model of Current sources and Vacuum vessel sector for calculating eddy currents in Module 18 and TBMs

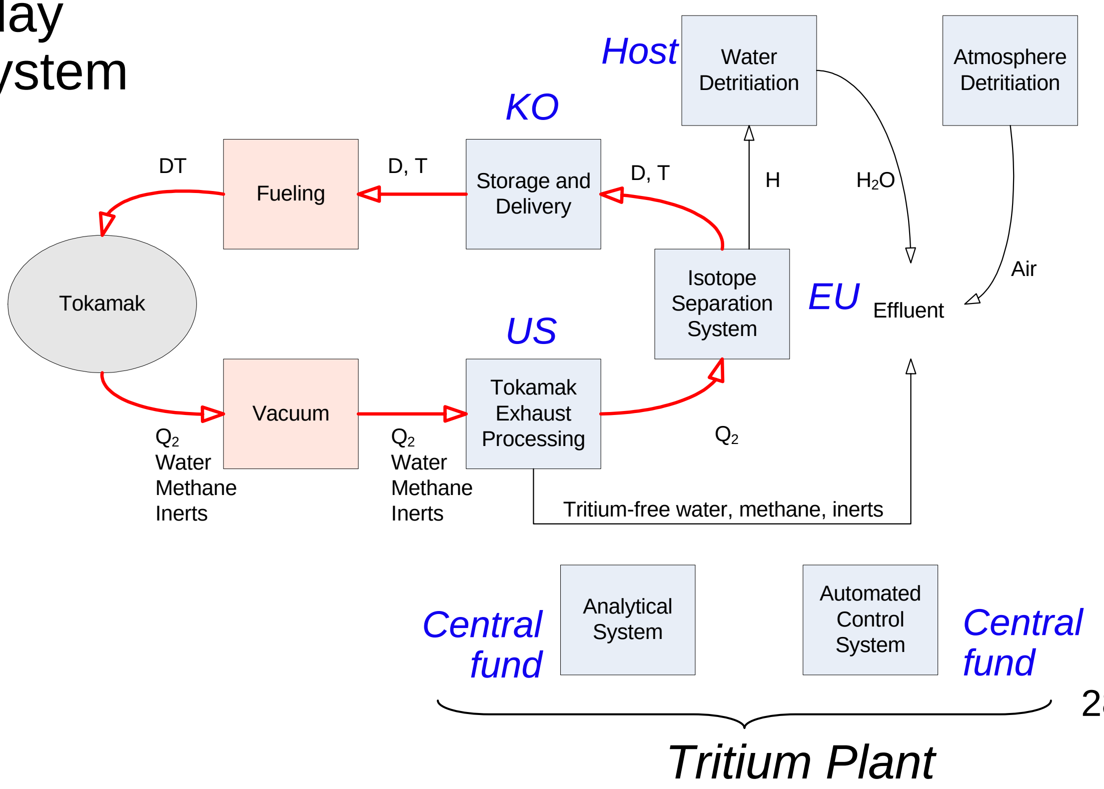
R.E. Nygren et al., "ITER First Wall Module 18" - Thursday Morning

Tokamak Exhaust Processing System responsibility of the US

- US is participating in the Tritium Plant Working Group to plan out the overall TEP procurement project and to prepare for TEP design work (to begin soon)

Key Tokamak Exhaust Processing System Design Specifications

- Lose no more than 1 Ci/day to the Vent Detritiation System
- Overall TEP decontamination factor (DF) of 10^8
- Process gas from 450 s and 3000 s pulses at a flowrate of ~75 SLPM
- Recently design flowrate was ~150 SLPM

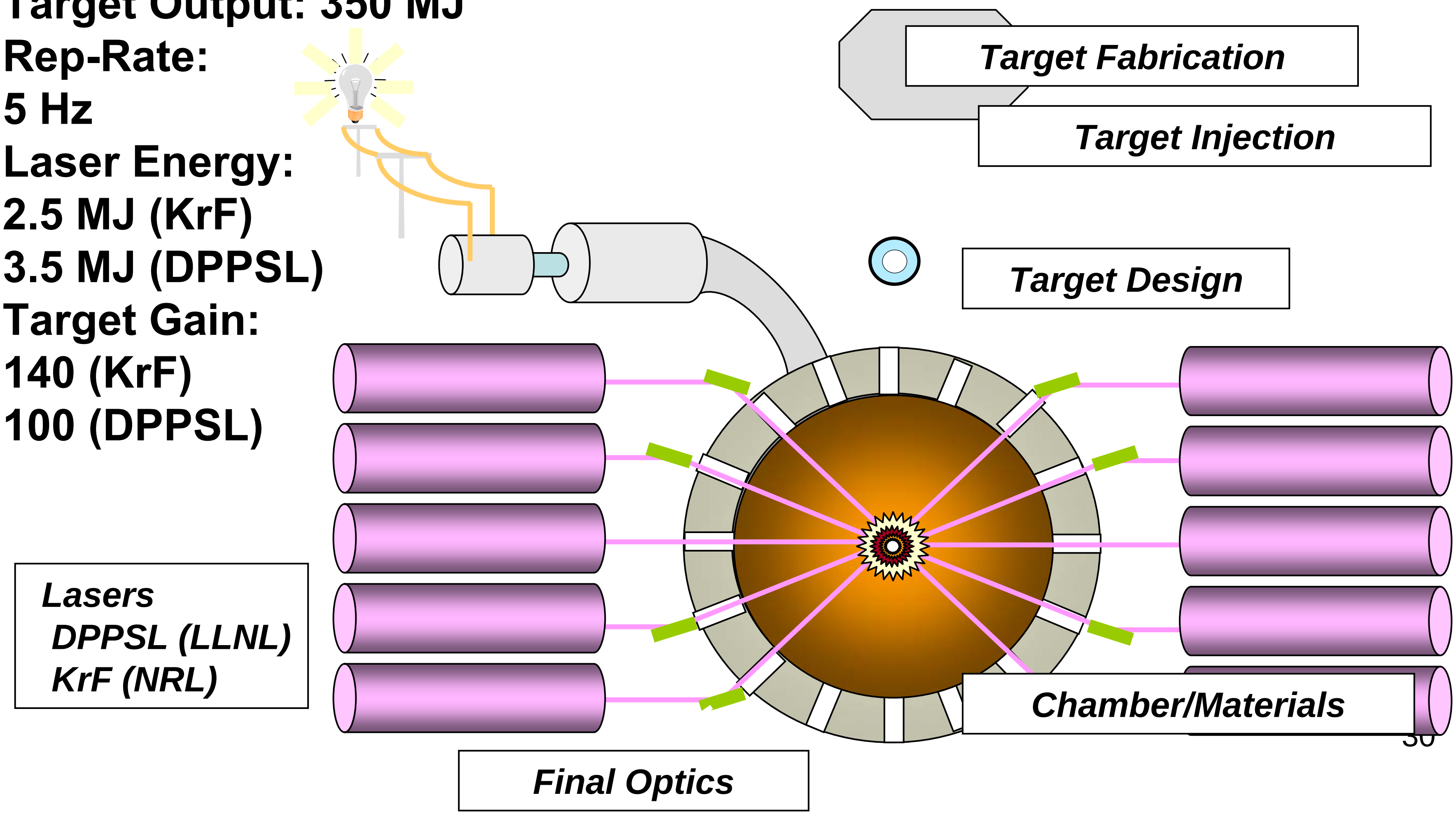


Outline of the Presentation

- ❑ Fusion Research Organization in the US
- ❑ Enabling Technologies / VLT Program
  - – ITER Test Blanket Module Program
  - – JUPITER-2 collaboration with Japan
  - – Materials Research
  - – Plasma Facing Components Research
  - – ARIES Design Studies
  - – Neutronics Simulation Tools
- ❑ ITER Project Office and US Contributions to ITER
  - – First wall / Shield Module 18
  - – Tokamak Exhaust Plant
- ❑ IFE Technology Research
  - – High Average Power Laser
  - – Z-Pinch Program
- ❑ Summary and Outlook

The High Average Power Laser (HAPL) Program is developing unique technologies for Inertial Fusion Energy

- ❑ **Plant Output: 500-800 MWe**
- ❑ **Target Output: 350 MJ**
- ❑ **Rep-Rate:**
  5 Hz
- ❑ **Laser Energy:**
  2.5 MJ (KrF)
  3.5 MJ (DPPSL)
- ❑ **Target Gain:**
  140 (KrF)
  100 (DPPSL)

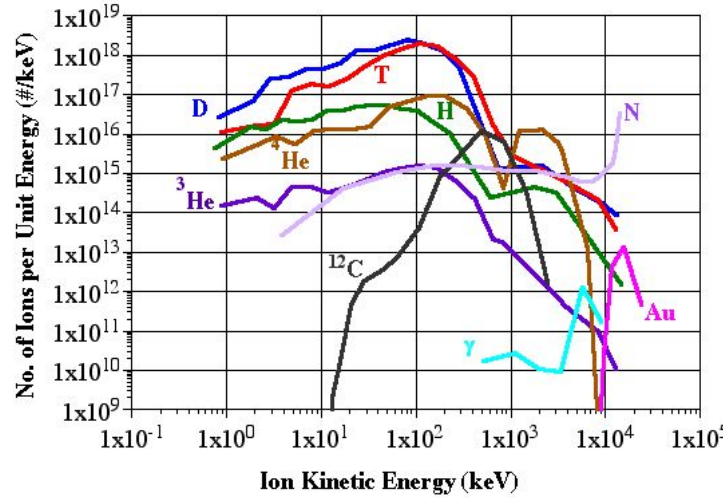


A key FNT issue is survival of the tungsten armor under the cyclic X-ray and ion threat spectra

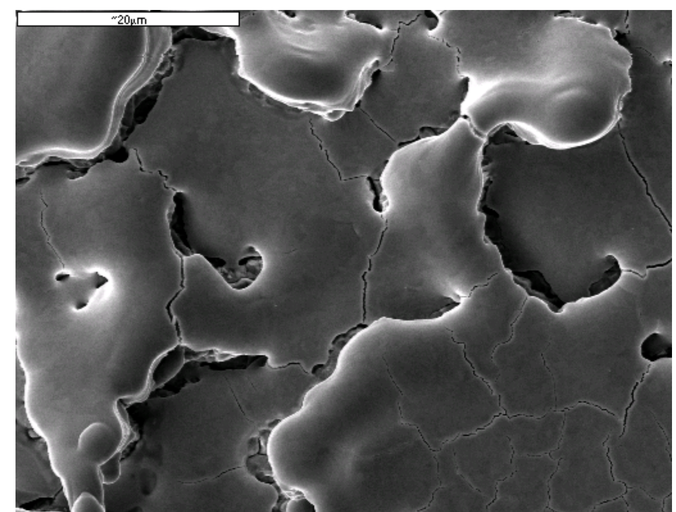
- ❑ Several possible mechanisms affect the armor survival
  - – Ablation, melting, surface roughening
  - – Cyclic thermal stress fatigue
  - – Accumulation of implanted helium
  - – Armor/substrate bond fatigue.
- ❑ Research effort includes modeling and experimental testing of the armor thermo-mechanical behavior in facilities utilizing ion, X-rays and laser sources to simulate IFE conditions.
- ❑ Significant progress has been made recently toward solving helium retention and bond fatigue
  - – but long term survival of the armor remains a key unresolved issue.

R. Raffray et al. “Progress towards realization of a laser IFE solid wall chamber”, and

M Andersen et al. “Thermomechanical analysis of a micro-engineered tungsten foam” – Tuesday Morning



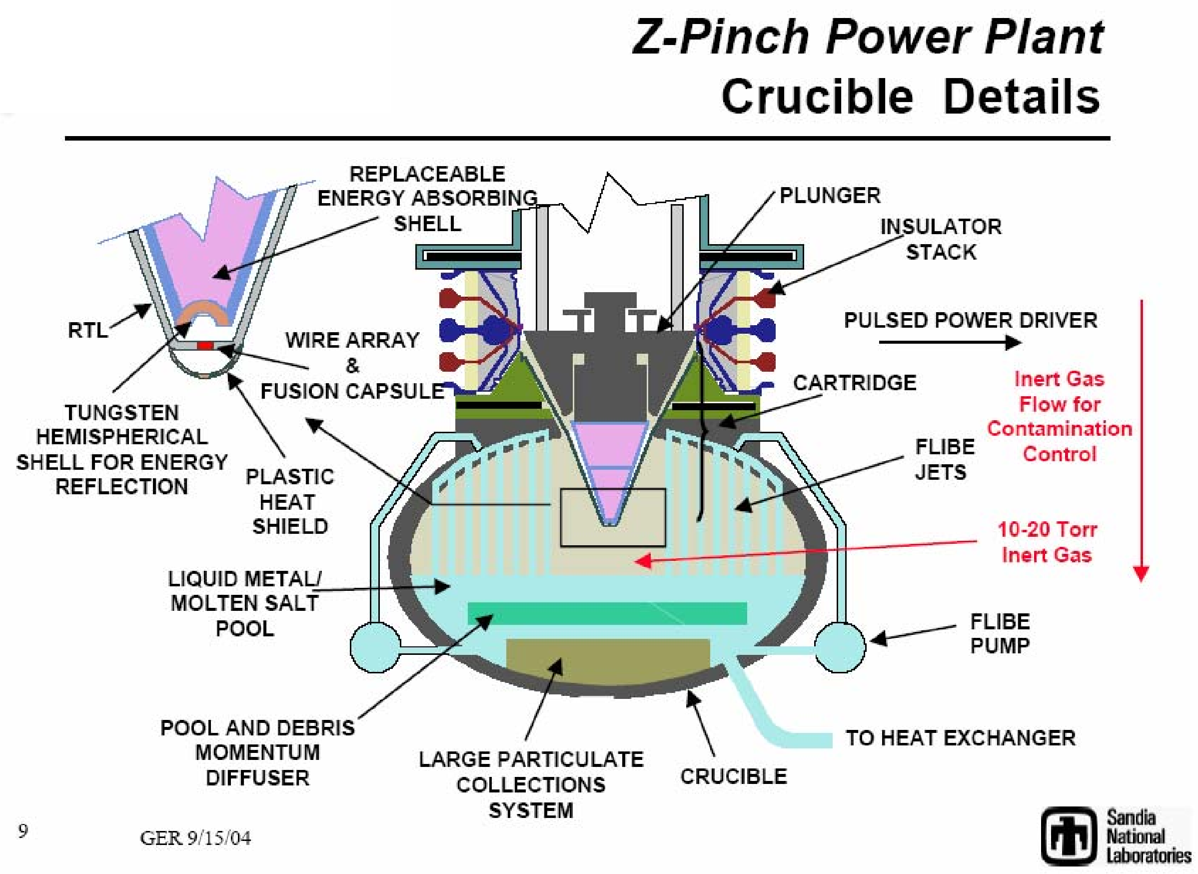
Ion species and spectra at chamber wall



Flaking of W armor after 1600 N+ ion beam pulses in RHEPP, SNL (2000x mag)

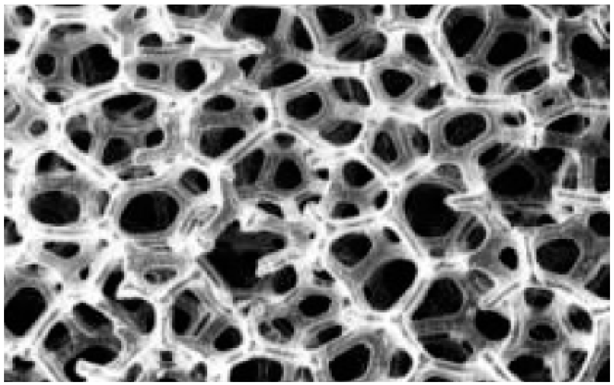
Z-IFE FNT Research Effort is focused on key issues

- **Feasibility of the Recyclable Transmission Line and full RTL cycle (fire RTL/z-pinch, remove RTL remnant, insert new RTL/z-pinch)**
- **Successful mitigation of shock from the high-yield target (~3 GJ) to protect the chamber structural wall**

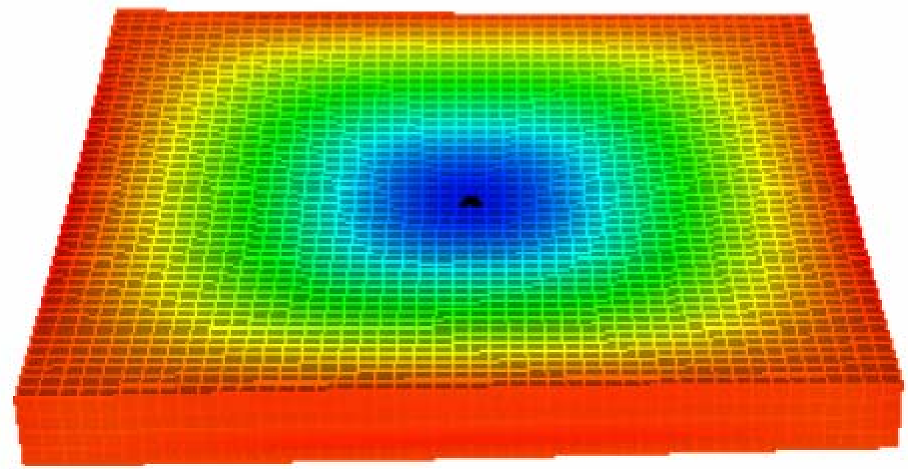


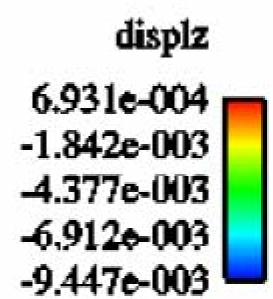
- **Achieve Proof-of-Principle of Z-IFE: 1 MA, 1 MV, 100 ns, 0.1 Hz**

Shock experiments on porous metal foams and multiple liquid layers show effectiveness in mitigating shocks for Z-pinch



Typical foam





Deformation of aluminum foam after passage of weak shock 1.34 Ma

Summary and Outlook

- Reported here is a wide variety of fusion nuclear technology R&D activities in the US

- The emerging importance of the ITER basic machine in the efforts of the US Enabling Technology program is readily apparent
  – Long term reactor relevant R&D efforts have been shifted and focused to those first wall and blanket concepts and materials that will be needed for or tested in ITER
  – IFE FNT R&D programs have been shifted to other funding sources in Defense Programs.

- There are major concerns among the US scientists and engineers that the recent policy trend of eliminating research on "long term" technologies and technical issues will have negative consequences on the ability of the US fusion program to realize its goal of demonstrating the potential of fusion as a viable and attractive energy source for many decades to come.

- My Opinion: despite these concerns, the capabilities, enthusiasm, and commitment of fusion nuclear technology researchers in the US remains strong owing to the prospect of contributing to ITER and utilizing the ITER fusion environment to advance the understanding and development of fusion nuclear technology.