Overview of US and UCLA Plasma Chamber Systems Program
(and UCLA work under PFC)

Briefing for Gene Nardella
Office of Fusion Energy Science, DOE

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Washington, D.C.
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Outline of Presentation

- Mission and History of the US Plasma Chamber Program
  - Current Elements of US Plasma Chamber Program
  - Interaction of Plasma Chamber with other US Programs
- Current UCLA Activities in the Plasma Chamber Program for MFE
- Current UCLA Activities in the Plasma Chamber Program for IFE
- Summary
Scope of Plasma Chamber Research

Plasma Chamber Research embodies the scientific and engineering disciplines required to understand, design, develop, test, build, and operate safely and reliably the systems that surround a burning plasma. PC includes all components and functions from the edge of the plasma to the magnets, including:

- first wall
- blanket (breeding and non-breeding)
- conducting shells
- vacuum vessel
- radiation shielding
- nuclear part of RF antenna, etc.
- cooling systems
- electric/thermal insulators
- tritium barriers and processing
- tritium fuel cycle
- support structure & remote maintenance

PC also includes design and integration for Chamber Components.
Mission of US Plasma Chamber Programs

- Advance the engineering sciences and develop technologies for plasma chamber systems that allow current and future plasma experiments (e.g., ITER) to achieve their goals and improve their performance potential.

- Support the ITER mission: “The ITER should serve as a test facility for neutronics, blanket modules, tritium production and advanced plasma technologies.” as stated by the ITER Quadripartite Initiative Committee (QIC), IAEA Vienna 18-19 October 1987.

- Resolve key feasibility issue for DT fusion: Ensure that tritium can be sufficiently produced, efficiently extracted and safely controlled, while simultaneously extracting heat at high temperature, in a practical engineering system surrounding the DT plasma and compatible with its operation under extreme conditions of high heat and particle fluxes, high temperature, strong magnetic field, and ultra low vacuum.

- Advance technologies of plasma chamber systems to realize an economically and environmentally attractive fusion energy source.
Since the Early 1970’s, Plasma Chamber Technology Research has had a Fundamental and Major Impact on:

1. The Direction and Emphasis of Plasma Physics R&D
2. The Direction and Emphasis of other Fusion Technology Programs
3. Identifying and Resolving Critical Issues in Fusion, many of which are “Go, No-Go” issues
4. Shaping our vision today of a burning plasma device and fusion power plant
Recent History of US Plasma Chamber Program

- The US Plasma Chamber Program enthusiastically worked on CDA and EDA phases of ITER, playing several critical roles.

- The Plasma Chamber Program suffered a MAJOR Budget cut in 1996 (reduced from annual budget of $6M to $1M). OFES and the community worked very hard to maintain a “bare minimum of critical skills” (skills that took 30 years to develop).

- Over the past 5 years, Plasma Chamber research focused on developing innovative ideas for the Chamber that could substantially improve the plasma performance and our vision of fusion.

- Some of this work on lithium walls has been embraced by plasma physics community and is continuing as part of the PFC program.

Now, a New Emphasis is required:

As we move forward with joining ITER, the Plasma Chamber community is already to restructuring its activities to best support the new initiatives.
Remarkable Events over the Past year!!

- Amazing Year!!
- FY04 Budget for Chamber was submitted to congress as ZERO in February. This erroneous, irrational decision shocked each and every person “who knew anything” about Fusion.
- The situation has been only partially rectified in the FY04 initial financial plan.
  - Thanks to the efforts of OFES, many of the fusion community leaders, and the proactive efforts of some senior members of the Chamber/Blanket Communities!

(How and Why such an irrational decision was made last December/January needs to be clearly understood by the Chamber/Blanket community in order to avoid such disasters in the future! Topic for social hour. But for now we need to move on.)

- The US rejoined ITER negotiations. ITER blanket testing came to focus and presented a great opportunity to move forward.
Redirecting Chamber Technology
Effort to support ITER

- With the US rejoining ITER, the Blanket/Chamber community concluded that it is very important for the US to participate in the ITER Test Blanket Module (TBM) Program.
- Reached consensus on a general framework for the direction of activities in the US Chamber/Blanket Program
- Key elements of the emerging framework are:
  - Provide fusion nuclear technology (FNT) support for the basic ITER device as needed
  - Participate in ITER TBM program and redirect good part of resources toward R&D for TBM
  - Enhance international collaboration between all ITER Parties to in carrying out the R&D and construction of the test facilities and modules.
  - Examples include JUPITER-II program between US and Japanese Universities and new opportunities with Korea and China
Outline – Current US Program

- Mission and History of the Plasma Chamber Program
- Current Elements of US Plasma Chamber Program
  - Critical Issues
  - ITER TBM
  - ITER Basic Machine FNT support
  - Jupiter-II Collaboration
  - IEA Activities
  - SBIR Activities
- Interaction of Plasma Chamber with other US Programs
- Current UCLA Activities in the Plasma Chamber Program for MFE
- Current UCLA Activities in the Plasma Chamber Program for IFE
- Summary
Remaining Critical R&D Issues for Plasma Chamber

1. Remaining Engineering Feasibility Issues, *e.g.*
   - feasibility, reliability and MHD crack tolerance of electric insulators
   - tritium permeation barriers and tritium control
   - tritium extraction and inventory in the solid/liquid breeders
   - thermomechanics interactions of material systems
   - materials interactions and compatibility
   - synergistic effects and response to transients

2. **D-T fuel cycle tritium self-sufficiency in a practical system**
   depends on many physics and engineering parameters/details: *e.g.* fractional burn-up in plasma, tritium inventories, FW thickness, penetrations, passive coils, and many more variables.

3. **Reliability/ Maintainability/ Availability**: failure modes, effects, and rates in blankets and PFC’s under nuclear/thermal/mechanical/electrical/magnetic/integrated loadings with high temperature and stress gradients. Maintainability with acceptable shutdown time.

4. **Lifetime of blanket, PFC, and other FNT components**
What is the ITER Test Blanket Module (TBM) Program?

- The ITER Test Program is managed by the ITER Test Blanket Working Group (TBWG) with participants from the ITER Central Team and representatives of the Parties.

- Breeding Blankets will be tested in ITER, starting on Day One, by inserting Test Blanket Modules (TBM) in specially designed ports.

- Each TBM will have its own dedicated systems for tritium recovery and processing, heat extraction, etc. Each TBM will also need new diagnostics for the nuclear-electromagnetic environment.

- Each ITER Party is allocated limited space for testing two TBM’s. (No. of Ports reduced to 3. Number of Parties increased to 6)

- ITER’s construction plan includes specifications for TBM’s because of impacts on space, vacuum vessel, remote maintenance, ancillary equipment, safety, availability, etc.
ITER’s Principal Objectives Have Always Included Testing Tritium Breeding Blankets

• “The ITER should serve as a test facility for neutronics, blanket modules, tritium production and advanced plasma technologies. The important objectives will be the extraction of high-grade heat from reactor relevant blanket modules appropriate for generation of electricity.”
  —The ITER Quadripartite Initiative Committee (QIC), IEÅ Vienna 18–19 October 1987

• “ITER should test design concepts of tritium breeding blankets relevant to a reactor. The tests foreseen in modules include the demonstration of a breeding capability that would lead to tritium self sufficiency in a reactor, the extraction of high-grade heat and electricity generation.”
  —SWG1, reaffirmed by ITER Council, IC-7 Records (14–15 December 1994), and stated again in forming the Test Blanket Working Group (TBWG)
Highlights of US Strategy for ITER TBM
(Evolved over the past several months by the community, DOE and VLT)

- The US will seek to maximize international collaboration. There is a need for all parties to collaborate, and to possibly consider a more integrated plan among the ITER parties for carrying out the R&D and construction of the test modules.
- ITER TBM should be viewed as a collaborative activity among the VLT program elements. While the Blanket/Chamber Program provides the lead role for ITER TBM, major contributions from other programs, e.g., Materials, Safety, PFC, are essential.
- The US must reconsider its previously preferred two blanket concepts in view of new technical results obtained over the past few years.
- The US community has now reached consensus on preferred options for ITER TBM (see next slide)
US Selected Options for ITER TBM

The initial conclusion of the US community, based on the results of the technical assessment to date, is to select two blanket concepts for the US ITER-TBM with the following emphases:

• Select a helium-cooled solid breeder concept with ferritic steel structure and neutron multiplier, but without a fully independent TBM. Rather, plan on unit cell and submodule test articles that focus on particular technical issues of interest to all parties. (All ITER Parties have this concept as one of their favored options.)

• Focus on testing Dual-Coolant liquid breeder blanket concepts with ultimate potential for self-cooling. Develop and design TBM with flexibility to test two options:
  – a helium-cooled ferritic structure with self-cooled LiPb breeder zone that uses SiC insert as MHD and thermal insulator (insulator requirements in dual-coolant concepts are less demanding than those for self-cooled concepts)
  – a helium-cooled ferritic structure with low melting-point molten salt. The choice of the specific lithium-containing molten salt will be made based on near-term R&D experiments and modeling. Because of the low thermal and electrical conductivity of molten salts, no insulators are needed.
## Port Allocations for ITER TBM

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<tr>
<th>Port A</th>
<th>Port B</th>
<th>Port C</th>
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<tbody>
<tr>
<td>He-Cer (1)</td>
<td>H$_2$O-Cer</td>
<td>Li/V</td>
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<tr>
<td>He-Cer (2)</td>
<td>He-LiPb</td>
<td>Dual Coolant (LiPb or Molten Salt)</td>
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<tr>
<td>Port Master A: Boccaccini</td>
<td>Port Master B: Enoeda</td>
<td>Port Master C: Kirillov</td>
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<tr>
<td>Working Group*</td>
<td>2 Working Groups*</td>
<td>2 Working Groups*</td>
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<tr>
<td>Cer/He</td>
<td>H$_2$O-Cer</td>
<td>He-LiPb</td>
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</tbody>
</table>

*Members nominated by each interested party (not necessarily members of TBWG).*
FIVE TBWG WORKING GROUPS

*Broad US representation in Working Groups*

- **Cer / He**: All parties - Boccaccini (Leader), Ying
- **He / LiPb**: EU, US, China, Korea, RF - Poitevin (Leader), Wong
- **H₂O / Cer**: Japan, China - Enoeda (Leader)
- **Li / V**: RF, US, Japan, China, Korea - Kirillov (Leader), Sze
- **Molten Salt**: US, Japan, RF, China? - Sze (Leader), Petti
ITER Test Blanket Module Activities

- **Motivation**
  - Utilization of ITER fusion nuclear environment
  - Tritium supply for fusion development
  - 1st steps in establishing tritium self-sufficiency

- **Activities**
  - Active participation in ITER test blanket working group (TBWG) for test and infrastructure planning
  - Evaluate blanket options for DEMO and evaluate R&D results for key issues to select primary US blanket concepts
  - Perform concurrently R&D on the most critical issues required
    - MHD flow with insulators and inserts
    - Tritium recovery and control
    - SiC inserts compatibility and failure modes
    - Solid breeder / multiplier / structure / coolant interactions
  - Develop engineering scaling and design, in collaboration with ITER partners, for TBMs.
# TBM Roll Back from ITER 1st Plasma

Shows CT R&D must be accelerated now for TBM Selection in **2005**

EU schedule for Helium-Cooled

Pebble Bed TBM (1 of 4 TBM's Planned) → **ITER First Plasma**

| Year | 02 | 03 | 04 | 05 | 06 | 07 | 08 | 09 | 10 | 11 | 12 | 13 | 14 | 15 | 16 | 17 | 18 | 19 | 20 | 21 | 22 | 23 | 24 | 25 |
|------|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|
| PB Material Fabrication and Char. (mech., chem, etc) | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| Out-of-pile pebble bed experiments | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| Pebble bed Irradiation Programme | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| Modelling on Pebble beds including irradiation effects | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| Key issues of Blanket Structure Fabr. Tech. | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| Develop. and testing of instrumentation for TBM | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| Develop. and testing of components of Ext. Loops | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| TBM and Ext. Loop Mock-up Design | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| TBM and Ext. Loops Mock-up Fabrication | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| Operation of TBM and Ext. Loop Mock-ups | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| Final Design of TBM | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| Fabrication and qualification of TBM and Ext. Loops | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| Operation in the Basic Performance Phase of ITER | | | | | | | | | | | | | | | | | | | | | | | | | | | |


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**HCPB Programme**

- a final decision on blanket test modules selection by 2005 in order to initiate design, fabrication and out-of-pile testing

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**HCPB Programme for ITER**

- TBM Roll Back from ITER 1st Plasma
- EU schedule for Helium-Cooled Pebble Bed TBM (1 of 4 TBM's Planned)
- ITER First Plasma
- A final decision on blanket test modules selection by 2005 in order to initiate design, fabrication and out-of-pile testing

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(UCLA)
Tritium Consumption in Fusion Is HUGE!
- 55.8 kg per 1GW-year fusion power

Production & Cost from fission are LIMITED & EXPENSIVE!
- CANDU Reactors: 27 kg over 40 years, $30M/kg (current)
- Fission reactors: few kg per year, $84M - $130 M per kg (per DOE Inspector General)

World Tritium Supply Exhausted by 2025 by ITER at 1000MW at 10% Availability or ITER at 500 MW at 20% Availability

Conclusion
Chamber technology is not just for power reactors. It is essential for continuing burning plasma and fusion research
Molten Salt Blankets Assessment

- We completed the conceptual design of a re-circulating FLiBe self-cooled blanket using Pb as the neutron multiplier and advanced nano-composited ferritic steel (AFS) (T limit @ 800°C) as structural material.

- The design can handle max. $\Gamma_n$ of 5.4 MW/m$^2$ and max. first wall heat flux of 1 MW/m$^2$ with gross efficiency of 46%.

- We will need to develop the AFS material and methods of fabrication and confirm the allowable interface temperatures for FLiBe/AFS and Pb/AFS. Be multiplier is also a credible option.

We are evaluating three reduced activation ferritic steel MS blanket options for DEMO and ITER test module: Dual coolant (He and FLiBe) with neutron multiplier options of Be and Pb, and a self-cooled FLiNaBe option with Be.
Support for the basic ITER device

Provide more accurate prediction in the nuclear area for critical ITER components as we move toward construction

- Diagnostics damage
- Personnel access, activation to assess site specific safety issues
- Provide support for US procurement packages – e.g. shielding blanket modules
- Predictive capabilities and tools needed by elements of fusion program
  - neutronics, activation, neutron-material interactions,
  - heat transfer, fluid mechanics, MHD,
  - tritium recovery and control, fuel cycle dynamics,
  - reliability and availability.
Thick first walls (>1cm) seriously threaten the ability to attain tritium self sufficiency, hence the feasibility of DT fusion.

Real Engineering Design of breeding blankets is needed as part of evaluating blanket options.
JUPITER-II collaboration for Plasma Chamber Systems

All experiments are directed to solve key feasibility issues for the molten salt, Li/V and solid breeder ITER test modules.

1. Flibe REDOX control
   - Completed flibe purification process.
   - Completed flibe mobilization experiment.
   - Started hydrogen isotopes (D) permeation, diffusion and solubility measurements.
   - Preparation for the REDOX experiments.
   - Presented two papers at ICFRM and Be workshop.

2. Flibe heat transfer and flow mechanics
   - Measured straight pipe velocity profile and turbulent statistics.
   - Constructed 304SS heat transfer test section, with some initial data available.
   - New acrylic PIV attachment section constructed and tested.
   - Prepared for the MHD experiment with a US supplied magnet.

3. MHD coating development
   - Coatings of AlN, Y2O3 and Er2O3 have been tested in Li up to 800C.
   - Vacuum distillation system developed and tested to remove residue lithium from test coupons.
   - Resistivity experiments conducted for Er2O3, Y2O3 and (Y,Sc)O3 to confirm sufficient resistivity for MHD coating.

4. SiC/pebble bed thermomechanics experiments
   - Two configurations were developed.
IEA Activity and the Subtask Group on Solid Breeder Blankets

(1) IEA Implementing Agreement on Nuclear Technology of Fusion Reactor is initiated in 1994 among Japan (JAERI), USA, Euratom and Canada with Annex I.

(2) Russian Federation joined in 1996.

(3) According to the progress of TBWG activity, the importance of the promotion of collaborative R&D was recognized. Solid Breeder Blanket subtask group started its activity in 1997.

(4) Annex I is renewed in 1999.

(5) Currently, 4 subtask groups (Solid Breeder Blanket, Liquid Breeder Blanket, Tritium Technology and Neutronics) are working on information exchange, workshop, collaborative experiments etc. under Annex I.

(6) The ninth Solid Breeder Blanket subtask group meeting was held in 2003 in Kyoto.
IEA collaboration on solid breeder pebble bed time dependent thermomechanics interactions/deformation research

**Primary Variables**
- Materials
- Packing
- Loadings
- Modes of operation

**Primary & Secondary Reactants:**
- Temperature magnitude/gradient
- Differential thermal stress/contact pressure
- Plastic/creep deformation
- Particle breakage
- gap formation

**Single/multiple effect experiments**
(NRG, UCLA)

**Finite Element Code**
(ABQUS, MARC)
(NRG, FZK, UCLA)

**Discrete Element Model**
(UCLA)

**Thermo-physical and Mechanical Properties**
Consecutive equations

**Partially integrated out-of-pile and fission reactor tests**
(NRG, ENEA)

**Design Guideline and Evaluation**
(out-of-pile & in-pile tests, ITER TBMs)

**Database Experimental Program**
(FZK, JAERI, CEA, UCLA)

**Goal:**
Performance/Integrity prediction & evaluation

**Irradiation Effect**
(NRG)
SBIR Awards in Plasma Chamber Area

- Hypercomp, Inc. Phase I and Phase II to continue development of various aspects of the HIMAG free surface and closed channel MHD code for complex geometries
  - New solvers for B-formulation, MHD turbulence, and analytic treatment of boundary layers
  - Complete inclusion of multiple conducting solid materials and heat transfer
  - Better parallel matrix solvers and iteration technique for highly skewed meshes

- Metaheuristics, Inc. Phase II to continue development of alternative Lattice Boltzman model for fluid flow with
  - MHD in closed channels with wetting and chemical reactions
  - Turbulence models
  - High parallel efficiency

- Plasma Processes. Phase I for liquid metal flow in porous coatings

Topics for next year solicitation
- SiC structures for flow channel inserts
- Virtual TBM
Outline – Current US Program

- Mission and History of the Plasma Chamber Program
- Current Elements of US Plasma Chamber Program
- Interaction of Plasma Chamber with other US Programs
  - Budget distribution
  - Interaction of programs
- Current UCLA Activities in the Plasma Chamber Program for MFE
- Current UCLA Activities in the Plasma Chamber Program for IFE
- Summary
Plasma chamber program
distribution of effort

- Plasma Chamber FY05 is planned at ~$1894k and is roughly split:
  - TBM and DEMO designs: $100k GA, $310k UCLA, $59k ORNL, $195k UW. Total: $664
  - J2 Thermofluid and Thermomechanics experiments: $590k UCLA
  - Other Thermomechanics, Thermofluid, Neutronics model development and experiments (e.g. for IEA): $640k UCLA

- Additional related work is supported via other programs
  - D-K Sze participation in TBM and J2 supported by VLT
  - INEEL participation in TBM and J2 supported by Safety
  - LANL participation in TBM supported by Tritium
  - Additional ORNL participation in TBM supported by Materials
  - SNL and UCLA (work on free surface MHD) supported by PFC
Technology Programs are Highly Interrelated and Interactive
(Take as an analogy a “three-legged stool”: PFC, Chamber Tech, and Materials) (Many Other “3-legged stool” examples can be shown with other parts of the fusion program, e.g. with Safety and Design Studies Programs)

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<thead>
<tr>
<th>Area</th>
<th>Plasma Technology</th>
<th>Fusion Technology</th>
<th>Materials Program</th>
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<tr>
<td>Key Issue</td>
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<td>First Wall/Blanket/Shield</td>
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<td>Tritium breeding and neutron multiplier materials R&amp;D</td>
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<td>Radiation shielding (components and personnel radiation protection, design and R&amp;D)</td>
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<td>Neutronics, photonics, and neutron material interactions (transport, DPA, He, H, transmutation, etc.)</td>
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<td>Blanket structural materials (development, properties and irradiation)</td>
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<td>Coolant/Multiplier/Breeder/Structure interactions and compatibility</td>
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<td>Tritium extraction, inventory, and control</td>
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<td>Thermofluid effects (heat transfer, fluid mechanics, MHD)</td>
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<td>Heat removal and thermal efficiency</td>
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<td>Vacuum Vessel</td>
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<td>Configuration and engineering design</td>
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<td>Plasma Facing Components</td>
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<td>Plasma materials interactions R&amp;D (effects of PFM on core plasma)</td>
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<td>Erosion of PFM and impurity control</td>
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<td>Joining of Plasma Facing Materials (PFM) to heat sink, thermal fatigue life</td>
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<td>Reliability, Availability, and Maintainability</td>
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<td>Methods, analysis, and R&amp;D (failure modes, effects, and rates; reliability growth; maintenance and availability; etc.)</td>
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<td>Remote maintenance technology</td>
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- **P** - Primary role for resolving issue,  
- **S** - Supporting role in resolving issue
Interaction also strong between Plasma Chamber and PFC community

- Next stage of NSTX needs a better mechanism to control particles and impurities – an experimental flowing lithium module in place of cryo-pumping is being considered
  - Technical support from PFC community building on liquid wall work in Plasma Chamber

- US work on ITER shielding blanket modules
  - significant overlap with Plasma Chamber interests and expertise
Outline – UCLA Program in MFE

- Plasma Chamber Program Mission and History
- Current Elements of US Plasma Chamber Program
- Interaction of Plasma Chamber with other US Programs
- Current UCLA Activities in the Plasma Chamber Program for MFE
  - ITER Test Blanket Module R&D
  - Molten Salt Thermofluid MHD (Jupiter-II)
  - Solid Breeder / SiC Thermomechanics (Jupiter-II)
  - Solid Breeder / Steel Thermomechanics (IEA)
  - ITER Basic Machine and Procurement Package Support
  - Plasma Facing Components
- Current UCLA Activities in the Plasma Chamber Program for IFE
- Summary
UCLA activities for US ITER TBM Program

• Leadership in:
  – Solid breeder TBM scaling, test plan development, test module design and supporting R&D
  – Liquid metal and molten salt thermofluid MHD experiments and simulations
  – US TBWG representation

• Participation in study for selection of liquid breeder option
  – Ancillary equipment definition for liquid breeder options
  – Critical issue groups for assessing state of R&D needs for liquid breeder TBMs
  – Simulation of thermofluid behavior and neutronics of DEMO blanket options
Solid breeder TBM research and development

All parties (6) are interested in testing helium-cooled solid breeder (HCSB) blanket modules in ITER

Why Solid Breeder TBM?

- Can provide an immediate solution to fusion chamber technology (tritium supply and heat removal)
- The remaining feasibility issue for solid breeder concerns solid breeder dimensional stability at high burnups (tritium inventory in beryllium at high fluence is an issue)

Goals

- Insert a quarter port size solid breeder submodule into ITER on day 1 of ITER operation
- ITER testing serves as a reality check for the integrated design concept (notice that there is no fusion relevant experimental result prior to ITER testing) and provides data for code benchmarking and performance evaluation
- The goal of ITER first phase testing is to produce an experimentally verified and optimized helium-cooled solid breeder blanket design concept for Demo under a neutron wall load of 3 MW/m²
  - Initial exploration of performance in a fusion environment
  - Initial check of codes and data
  - Develop experimental techniques and test instrumentation
Previous U. S. Activities in Solid Breeder Blanket R&D

Prior to 96, large effort existed in broader Solid Breeder R&D areas:

- Solid Breeder Irradiation and Analysis, including In-situ Tritium Recovery (BEATRIX II, etc.)

- Ceramic Breeder Materials Thermodynamics - Investigation of formation of lithium hydroxide and material compatibility issues

- Tritium Inventory and Modeling - A comprehensive code MISTRAL was developed and used for DEMO and ITER(EDA) blanket tritium transport analysis

- Experiments, modeling and analysis of solid breeder blanket material system thermomechanics, to resolve thermal control issues for solid breeder blankets

- Solid Breeder Blanket Design Activities
Present U.S. Solid Breeder Blanket R&D

Carried out mostly in collaboration with other countries (IEA, JUPITER-II)

Solid Breeder Blanket Specifics:

- Focus on niche areas of solid breeder blanket material system thermomechanics interactions (Primary organizations: UCLA, Support: ORNL, PNL)
  - design database on effective thermo-physical and mechanical properties for breeder and beryllium pebble beds
  - experiments and modeling development on evaluation of thermomechanical states of blanket element pebble beds under different loading conditions

Material/Blanket Experiments Interface:

- Development of Web based Integrated Fusion Materials Database (UCLA, UCBS, ANL, ORNL)
- Construction of VISTA (VIRTUAL INTERNATIONAL STRUCTURAL TESTING ASSEMBLY) modelling tool, to evaluate a range of potential interactions and failure paths (perform “Virtual Experiments”). (UCLA, UCBS, FZK, ANL, ORNL)
Database Assessment: Existing data for Be pebble bed $K_{eff}$ shows large discrepancy

- Discrepancy as high as 50% found in obtained values of effective conductivity as a function of temperature and applied external pressure
- Experiments planned to resolve the discrepancy, as well as to address cycling effects

**Similar bed properties (particle size and packing fraction)**

**$k_{eff}$ vs. Temperature**

**$k_{eff}$ vs. Applied Pressure**

- Discrepancy as high as 50% found in obtained values of effective conductivity as a function of temperature and applied external pressure
- Experiments planned to resolve the discrepancy, as well as to address cycling effects
Experiments, Microscopic and Macroscopic Modeling efforts simultaneously underway to Understand and Predict Solid Breeder Blanket Pebble Bed Thermomechanics Interactions

Stress exerted on the wall at different bed temperatures

Average stress exerted on the particles at initial time and at time 2000 minutes

Stress relaxed as creep initiated
Stress magnitude profiles at different times

Force distribution inside the particles with 1% compressive strain

Test Article for Deformation Study

Solid breeder pebbles after the tests

MARC calculations

DEM calculations
MAJOR TASKS

Breeder and Pebble Bed Characterization and Development

Multiplier and Pebble Bed Characterization and Development

Blanket Thermal Behavior

Advanced In-Situ Tritium Recovery (Fission Tests)

Nuclear Design and Analysis (Modeling Development)

Fusion Test Modules Design Fabrication and Testing

Material and Structural Response

Tritium Permeation and Processing

LEGEND

Initiate Task
Terminate Task
Evaluation Point
Operate Major Experiment
Terminate Major Experiment
Information Flow

Test Sequence for Major Solid Breeder Blanket Tasks


**Additional R&D Tasks For Solid Breeder Test Modules**

- **Submodule/module definition** for ITER Solid Breeder Blanket TBMs
- Design, construction, and operation of **out-of-pile experiments** for more integrated tests (interactions among elements)
- Participate in submodule **experiments in fission reactors**
  [advanced in-situ tritium release, material interactions, and synergistic effects]
- Develop **plan for construction** and testing of ITER TBMs

*The US will collaborate with other parties on the solid breeder blanket TBMs*
Pb-17Li dual coolant blanket concepts have international support

- Nearly all ITER parties have interest in Pb-17Li blankets – either separately cooled or dual coolant concepts
- UCLA is participating in development of test plan, scaling and TBM design for dual-coolant PbLi concepts in collaboration with interested US institutions and international partners.
- UCLA is leading assessment in US program on the feasibility of the SiC insert based on current state of the R&D (part of the selection study in the US)
- UCLA is working on development of simulation capability and experimental test plan for LM-MHD effects in closed channels with complex geometry and heterogenous wall conductivity (with strong participation of HYPERCOMP SBIR).
UCLA MTOR can be for basic flow physics, free surface and TBM module simulation experiments

- Large magnetic volume for complex geometry modules
- Higher field smaller volume regions for higher MHD interaction experiments
- 30 liter gallium alloy flowloop
Jupiter-II Flibe Thermofluid Simulation: Task 1-1-B

- Measurement and simulation of Flibe turbulent flow, heat transfer, and heat transfer enhancement characteristics in magnetic fields using flibe simulant water/KOH
- Important molten salt feasibility issue and ITER TBM R&D issue

**US-Side**
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Z. Kawara, S. Ebara,  
H. Nakaharai
Thermofluid Task Schedule for 6 year collaboration

<table>
<thead>
<tr>
<th>Year</th>
<th>Non-magnetic Phase</th>
<th>Magnetic Phase</th>
</tr>
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<tbody>
<tr>
<td>FuY 2001</td>
<td>Turbulence Visualization Experiments</td>
<td>Turbulence Visualization Experiments</td>
</tr>
<tr>
<td>FuY 2002</td>
<td>Heat Transfer Experiments</td>
<td>Heat Transfer Experiments</td>
</tr>
<tr>
<td>FuY 2003</td>
<td>Pipe flow geometries with innovative heat transfer enhancement configurations</td>
<td>Same geometries as 2001-03 with magnetic field</td>
</tr>
</tbody>
</table>

Check & Review

- Continue with heat transfer, or another option
- Continue with MHD, or another option?
- Flibe Loop, or another option?

Facility: FLIHY-2 (UCLA)
J2 Thermofluid Test Section
Annual work scope for
Jupiter-II Thermofluid in 2004-05

1) Complete plans and construction for adding magnetic capability – done at PPPL with guidance from UCLA
2) Heat transfer experiment schedule
   Heat transfer experiment for straight pipe w/o magnetic field will be performed through summer 2004
   Heat transfer experiment start with magnetic field from Dec 2004
3) Development of PIV measurement under magnetic field
4) Consideration of heat transfer enhancement
5) Direct numerical simulation for turbulent pipe flow with magnetic field will be performed in Japan

Turbulence visualization experiments with PIV showed pretty good agreement with DNS
Flow facility will require modification for magnetic operation

- Existing 600 kW magnet power supply and cooling water must be redirected to J2 thermofluid test stand
- J2 experiments next year will focus on MHD effects on heat transfer in simulated flibe blanket flows!
- Magnet to be operational Dec 2004.
- Magnet is also extremely useful for LM-MHD experiments in support of TBM research on PbLi as well as molten salt simulants.
JUPITER II Task 2.2 SiC system thermomechanics tasks in FuY2003

- Box Furnace (up to 1100 °C), Power Controller, TCs, Capacitance Sensor, & Data Acquisition
JUPITER II Task 2.2 SiC system thermomechanics
test stand under construction

Controlled mechanical constraint, environmental conditions, expanded diagnostics capability, glove box set-up for Be handling

SiC/ SiC and CB or Be pebble bed test article
IEA Research Focus: ceramic breeder and beryllium pebble beds/structural wall time-dependent thermomechanics interaction

Provide Important Links to International Programs through IEA

Allow the US, through modest investments to gain access to much larger international program

- Stress generated as a combined effect of differential thermal expansion and the applied mechanical boundary conditions could lead to particle rearrangement, cracking and time-dependent deformation, which alters the thermomechanical state of the pebble bed and its associated effective thermo-physical and mechanical properties.
- Modeling development provides tools to predict and understand the critical effects of thermomechanics interactions
- Experimental data guide and verify modeling development

Worldwide Solid Breeder Activities

Breeder/beryllium material development and characterization
Permeation barrier development
Irradiation testing
Large-scale testing
Tritium release and recovery
Thermomechanics Modeling and Exp.

US Participation
UCLA support ITER Nuclear Design and Shielding Blanket

- As ITER moves toward construction it will need more accurate predictions in the nuclear area, e.g.
  - computation of radiation field, radiation shielding, nuclear heating, penetrations, materials radiation damage, decay heat, radwaste, maintenance dose, tritium fuel cycle, tritium permeation and inventories, basic device non-breeding blanket issues and performance.

- UCLA continues to assist in resolve remaining issues in ITER design, in particular those related to US bid procurement packages.
  - Support for the design and testing of shield blanket “baffle” modules if of key interest to UCLA precursor to TBM modules, which will also have plasma exposure.
  - flexibility in non-breeding blanket design needs to be to ensure reliable and safe operation, and possibly even the feasibility for change to breeding blanket in an extended operation phase.
Neutronics Capabilities

- Developed and used state-of-the-art computational tools and data bases for nuclear analyses:

  **Transport codes**: MCNP (Monte Carlo), TORT, DORT, ANISN, DANTSYS (Deterministic Methods) for 1D, 2D, 3-D modeling

  **Activation/Dose Codes**: DKR-Pulsar, ALARA, REAC

  **Data Processing**: NJOY, TNANSX, AMPX

  **Sensitivity/Uncertainty analyses**: FORSS, UNCER

  **Cross Section Data bases/libraries**: ENDF/B-VI, FENDL-2

- 35 years Experience in Nuclear analyses and Design of Tokamak machines: ITER-CDA, ITER-EDA, ARIES series, INTOR, UWMAK series, etc.

- Extensive experience in analyses of integral experiments using 14-MeV neutron sources: US/JAERI and IEA collaborations, ITER R&D neutronics experiments on shielding blanket experiments

- **Major Contribution to ITER**: ITER Test Blanket Module (till 1998), Nuclear analyses of ITER basic and breeding blanket design, and Dose calculation in ITER building during operation and after Shutdown (till 1998).
ITER Diagnostics for Nuclear Environment

In addition UCLA proposes to help develop diagnostics, diagnostic techniques, and nuclear analysis for plasma diagnostic ports for the magneto-nuclear environment of fusion devices (ITER, CTF, etc.). Such diagnostic systems are needed for both basic machine operation, productive experiments in TBM systems, and tritium fuel cycle data collection.

(this important work is currently not funded)
UCLA research on plasma facing components

- Focus is on continuing combined numerical and experimental effort to understand and develop predictive capability for NSTX flowing Lithium module stages
  - Stage 1: thin stagnant liquid Li test
  - Stage 2: flowing lithium used to solve NSTX particle pumping
  - Stage 3: flowing lithium for improved plasma performance
  - Stage 4: flowing lithium solves heat removal problem in NSTX for long pulse operation

The concept: NSTX Lithium Free Surface Module (ORNL)
Experiments on film flows show formation of 2D turbulence structures

- Turbulent fluctuations organize into 2D structures with vorticity along the magnetic field
- Corner vortices and small surface disturbances suppressed
- Flow can Pinch-IN in field gradients and separate from the wall
- Drag can be severe, slowing film down by 2x or 3x
LM jets streams less sensitive to magnetic field gradients and direction changes

- Jets are stabilized by reduction of turbulence and secondary flows in nozzle
- All return current paths in free jet must be in the liquid itself.

LM Jet flows in MTOR under increasing field strength (Bmax varies from 0 to 1.1 T left to right)
UCLA is collaborating on HIMAG 3D - a complex geometry simulation code for free surface MHD flows

- Simulations are crucial to both understanding phenomena and exploring possible flow option for NSTX Li module
- Problem is challenging from a number of physics and computational aspects requiring clever formulation and numerical implementation

Unstable MHD velocity profiles in gradient magnetic fields breakdown into instability

Complex geometry: Free surface flow around cylindrical penetration
HIMAG has already successfully simulated aspects of experiments in MTOR

- Wave crests running into the center of flow when gradient region is reached, very similar to video image
- Detailed analysis of numerical and experimental data the subject of UCLA PhD thesis

- 5 cm x 40 cm insulated trough
- Initial velocity, 2 m/s
- Density ratio 6400, Ga-In-Sn in air
- Surface normal magnetic field only
- Free surface initial thickness, 2 mm
- 900,000 cells – 16 processors
Tasks on PFC work for next year

- Conversion of MTOR ¼ segment to higher field for wide channel film flow experiments
- Initial experiments for wicked flow through porous weaves to evaluate this technique for early stage nearly-stagnant NSTX experiments
- Wide channel flow in surface-normal field gradient with quantitative laser and inductive surface height measurements
- Continue simulating experimental results and adding capabilities to HIMAG code
Outline – UCLA Program in IFE

- Mission and History of the Plasma Chamber Program
- Current Elements of US Plasma Chamber Program
- Interaction of Plasma Chamber with other US Programs
- Current UCLA Activities in the Plasma Chamber Program for MFE
- **Current UCLA Activities in the Plasma Chamber Program for IFE**
  - Chamber clearing rates
  - Modeling mass transfer at liquid free surfaces
  - Z-pinch

- Summary
Vapor Condensation Studies for HIF concepts: chamber clearing rates

Developed an innovative and inexpensive scheme to generate flibe vapor in conditions relevant to fusion technology design studies involving a liquid protection scheme (HIF, IFE, Z-pinch)

Measured flibe vapor clearing rates suggest that high repetition rates in HIF power plants are feasible provided that high purity of the molten salt is ensured

Found that for flow conditions characterized by high kinetic energy flibe vapor condensation is partially inhibited on metal surfaces perpendicular to the main component of the vapor velocity

Results presented at the 15th International Symposium onHeavy Ion Inertial Fusion (Princeton University, June 2004)

SEM characterization of condensed material
Novel Diagnostic for Time-Resolved Li and Be Density Measurements

Measured line intensity at 670 nm related to BeLiF3 Vapor Pressure as a function of liquid Flibe Temperature

Results presented at the 16th International Conference on Plasma Surface Interaction in Controlled Fusion Devices (Portland, May 2004)
Modeling development for free surface flow with mass transfer

• This continuing fundamental research (graduate student PhD thesis) is aimed at:
  – spray droplet condensation efficiency in IFE
  – droplet heat transfer enhancement of free surface liquid divertors in MFE
  – Developing a suitable model that can be incorporated into HIMAG capabilities at a later stage

• During the course of model development, numerical simulation of the heat and mass transfer capabilities of droplets sprayed onto the free surface will be addressed.

• The focus of FY 05 is to adopt a high order numerical method (Ghost fluid method) in the modeling to better resolve the contact discontinuity boundary conditions.
UCLA is interested to continue IFE research if possible

- If funds are restored to IFE technology, UCLA is very interested to continue fundamental research for IFE
  - P. Calderoni has been hired to work on other research in our group, but is available to continue research on vapor condensation that was just coming to its productive stage
  - The facility developed for the condensation research is still active and available (see Z-pinch work, next page)
  - X. Liu (PhD student) is continuing to develop numerical techniques for assessing aspect of spray assisted condensation.
  - Other numerical tools developed for MFE have significant overlap with IFE needs.
Investigation of flibe properties for Z-pinch Recyclable Transmission Lines

GOAL 1: Determine chamber clearing rates in conditions relevant to Z-pinch power plant chamber (1-10 Torr Ar background gas, 600-700 C flibe temperature)

GOAL 2: Investigate flibe electrical properties, such as voltage breakdown over the liquid surface

GOAL 3: Characterize the interaction of carbon steel condensing vapor with liquid flibe for recycling operations
Summary

- Mission and History of the Plasma Chamber Program
- Current Elements of US Plasma Chamber Program
- Interaction of Plasma Chamber with other US Programs
- Current UCLA Activities in the Plasma Chamber Program for MFE
- Current UCLA Activities in the Plasma Chamber Program for IFE

Summary