Idaho National Laboratory

Japan-US Joint Project TITAN 2007-2012

US/Japan TITAN Program:

Explorations toward Fusion Power Performance

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ARIES-Pathways Meeting

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TITAN Program Background

- Joint endeavor in the US and Japan to study *"Tritium and Thermofluid Control in Magnetic and Inertial Fusion Energy Systems"*
- 6-year R&D program involving several Japanese Universities/ Institutes and US Fusion R&D Facilities
- Follow-on to the JUPITER and JUPITER-II collaborations; JUPITER-II focused on each key technologies of materials, tritium and thermofuild for advanced blankets.
- In the next step, the principle concern is matching the first wall, blankets, and systems for tritium recovery and thermal performance, advancing the technologies for an integrated fusion power handling system.
- Short name TITAN derived from *"**T**ritium, **I**rradiation, and **T**hermo-fluid studies for **A**merica and **N**ippon"*

Program Objectives

- To obtain fundamental understanding for establishing tritium and thermo-fluid control throughout the first wall, blanket, heat exchanger, and tritium recovery system of MFE and IFE systems by experiments under specific conditions to fusion, such as irradiation, pulse high heat flux, circulation and high magnetic fields.
- The results will be applied through the integrated modeling to advancement of design for tritium and heat control of MFE and IFE systems.

This program initiates a certain level of technology integration based on presently-available capabilities, with the idea of DEMO-relevance in mind…

A related consequence is the usefulness of this R&D to various ITER Test Blanket Module concepts…

Task Structure and Research Items

Task	Subtask	US Facilities	Research items
Task 1 Tritium and Mass Transfer Phenomena	**1-1 Tritium and mass transfer in first wall**	STAR/TPE PISCES	Tritium retention and transfer behavior and mass transfer in first wall
	1-2 Tritium behavior in blanket systems	STAR	Tritium behavior through elementary systems of liquid blankets
	1-3 Flow control and thermo-fluid modeling	MTOR	Flow control and thermo-fluid modeling under strong magnetic fields
Task 2 Irradiation-Tritium Synergism	**2- 1 Irradiation-tritium synergism**	HFIR STAR	Irradiation effects on tritium retention and transfer behavior in first wall and structural materials
	2-2 Joining and coating integrity	HFIR	Synergy effects of simultaneous production of tritium and helium on healthiness of joining and coating integrity
	2-3 Dynamic deformation	HFIR	Effects of irradiation and simultaneous production of tritium and helium on dynamic deformation of structural materials
Common Task System Integration Modeling	**MFE/IFE system integration modeling**		Integration modeling for mass transfer and thermo-fluid through first wall, blanket and recovery systems of MFE/IFE

Task 1-1 Studies: ***Tritium and Mass Transfer in the First Wall***

Important issues of PWI for ITER and DEMO:

- **Mixed material effects**
  - W, C, Be for ITER, ?? for DEMO
  - He effects
    - Material degradation, Erosion, T retention, T permeation
- **Neutron irradiation effects**
  - Radiation damage, Transmutation effects (W→Re,Os)
    - Material degradation, Erosion, T retention, T permeation
- **Pulsed heat load effects**
  - ELM, (Disruption)
    - Material degradation (cracking, dust formation, etc.)

Interest centers on advanced (e.g. UFG) W, a potential candidate for DEMO FW. Also study W-coated RA mtls to understand fuel retention behavior on the interface.

Task 1-1 Studies, cont.:

Research Subjects of Task 1-1

- **Plasma Driven Permeation/Retention**
  - non-irradiated mono materials (W, RAF)
    - Irradiated materials for Task 2-1
  - Mixed (duplex) materials (W on RAF, W on SiC/SiC)
    - Evaluation of first wall materials for blankets
  - He ion irradiation effects on permeation/retention
- **Material Mixing and PSI phenomena for Be/W/C system**
  - Formation conditions
  - Erosion and morphological change by D/He/Be mixed plasma
  - Retention of D and He in mixed layer
- **Pulsed heat load effects**
  - Effects on mixed layer formation
  - Effects of surface morphology and material degradation
  - Ablation, dust generation, redeposition

Task 1-1 Studies, cont.:

Role of US and Japan

- High density tritium plasma experiment exclusively in TPE
  - Use of tritium greatly increases sensitivity
    - Permeation measurements with very low permeating flux
    - Tritium depth distribution measurements by IP (Imaging Plate)
  - Capability to test irradiated samples
- PMI study with Be plasmas in PISCES-B
  - Contribution to ITER
  - Mixed material effects with Be, in-situ surface analysis
  - Pulsed heat load operation
- Experiments without T and Be, devices in Japan also contribute
  - Linear and torus plasma devices (NAGDIS） (Nagoya Univ.)
  - Mixed Ion Beam device (HiFIT) (Osaka Univ.)
  - Pulse laser device (Kyusyu Univ., Osaka Univ.)

Task 1-1 Studies, cont.:

Experiment plan for PISCES (to mid-term review)

- Y1: Fine-grain tungsten (from Japan) exposure in PISCES-B
  - Vary Be %, fluence, plasma species mix, surface temp., D retention
- Y1: Select and acquire high-power laser, install in PISCES Lab, design safety upgrades for P-B enclosure, test fire into P-A, develop fast diagnostics on P-A
- Y2: Fine-grain W exposure with heat pulses (material effects:P-B)
- Y2: Witness plate TEM sample collection (in P-B)
- Y2: Modify P-B enclosure for laser operation
  - Install required fast diagnostics for P-B
  - Preliminary test shots into P-B

Task 1-1 Studies, cont.:

Experiment plan for TPE (to mid-term review)

Tritium Retention Studies

- Y1: Procure, setup, and calibrate Image Plate analyzer
- Y1: Setup cutting and polishing equipment for tritiated samples in glovebox and ventilated hood
- Y1-Y3: Depth profiling of tritium in W by IP analysis
  (estimate at least 12 runs at various fluence and sample surface temperature)
- Y1-Y3: Corresponding tritium retention measurements (TDS) in W

Plasma-driven Permeation Studies

- Y1: Design of in-situ plasma permeation target assembly
- Y2: Fabrication and thermal testing (with deuterium) of assembly
- Y3: Permeation tests on W and RAFS (with tritium)

Task 1-2 Studies: *Tritium Behavior in (liquid) Blanket Systems*

Key technical items for tritium in liquid blanket systems

- Solubility in PbLi
  - typical measurements performed at relatively high hydrogenic partial pressure ($\sim 10^1$-10^4 Pa) are extrapolated to much lower partial pressures required for tritium inventory control
  - deviance from Sievert's Law is possible (based on other LM results, e.g. Li)
  - measurements at extremely low concentrations require tritium
- Recovery methods from Pb-Li (as well as other liquid breeders) and He flows
  - inadequate mass transport across liquid-vapor interface for vacuum disengagement or window permeators in PbLi
  - oxidation or cryogenic systems for He, with structural and power implications
  - ingenious techniques for high recovery efficiencies are needed
- Transport barriers resistant to thermal cycling and irradiation
  - minimum required PRF ~ 100, needs robustness or self-healing attributes
  - success (or lack thereof) greatly influences direction of blanket system design
- Permeation behavior at very low partial pressures over metals
  - linear vs. Sievert's behavior? transport related to dissociation/recombination rates becomes non-equilibrium?
  - influence of surface characteristics and treatment

Task 1-2 Studies, cont.:

What is needed?

- Solubility of tritium in LiPb at very low (<< 1 Pa) partial pressure
- Efficient tritium recovery method
  - — Much work has been done, mainly in EU, to develop an efficient tritium recovery process
  - — However, the efficiency of the recovery process is only about 20%, whereas DEMO target recovery efficiency is about 75%
  - — Improvement of the EU process, or the development of a new process, will be necessary
- Development of reliable tritium diffusion barriers
  - — Successful development presently achieves PRF ~ 100, but only over small surface areas
  - — Substantial failure observed under irradiation
  - — Under TITAN program, Japan side will develop coating, and US side will test the effectiveness of the coating (possibly irradiated, also thermal cycles)
- Establish the pressure dependency of tritium permeation
  - — important for tritium control and the tritium recovery method (e.g. based on permeation window)

Task 1-2 Studies, cont.:

Research Project Areas

- Solubility of T in Pb-Li at Blanket Conditions
  - Low pressure region of hydrogen isotopes using tritium
  - Confirmation of Sieverts' Low, Isotherm of Pb-Li and T system
- Concentration Effects of T Permeation in Structural Materials and TPB Coating
  - Wide T pressure range covering several kinds of liquid breeders
  - Performance test on SM as well as TPB coating (to be developed in Japan)
- Tritium Extraction from Pb-Li and Other Liquid Breeders at Blanket Conditions
  - Mass transfer kinetics
  - Permeation window, gas engager, etc.
  - Performance test on a loop which is constructed inside or outside the budget
- Modeling and System Design for Tritium Behavior at Blanket Conditions

Task 1-3 Studies: Flow Control and Thermo-fluid Modeling

Motivation for collaboration in this area

- Critical feasibility area for any self-cooled or dual-coolant liquid metal blanket concept:
  - pressure drop and pressure drop reduction affecting primary stress and pumping power
  - flow profiles and flow distribution affecting heat transfer and hot spot formation, corrosion, tritium transport and permeation
- Strong interest by both parties
- Unique facility in the US
- Largely generic to selection of liquid metal
- A very suitable subject for basic scientific investigation, ideal for university research.
- A logical extension of Jupiter-II program and facilities, with shifted emphasis to Liquid Metals

Task 1-3 Studies, cont:

Research Project Areas

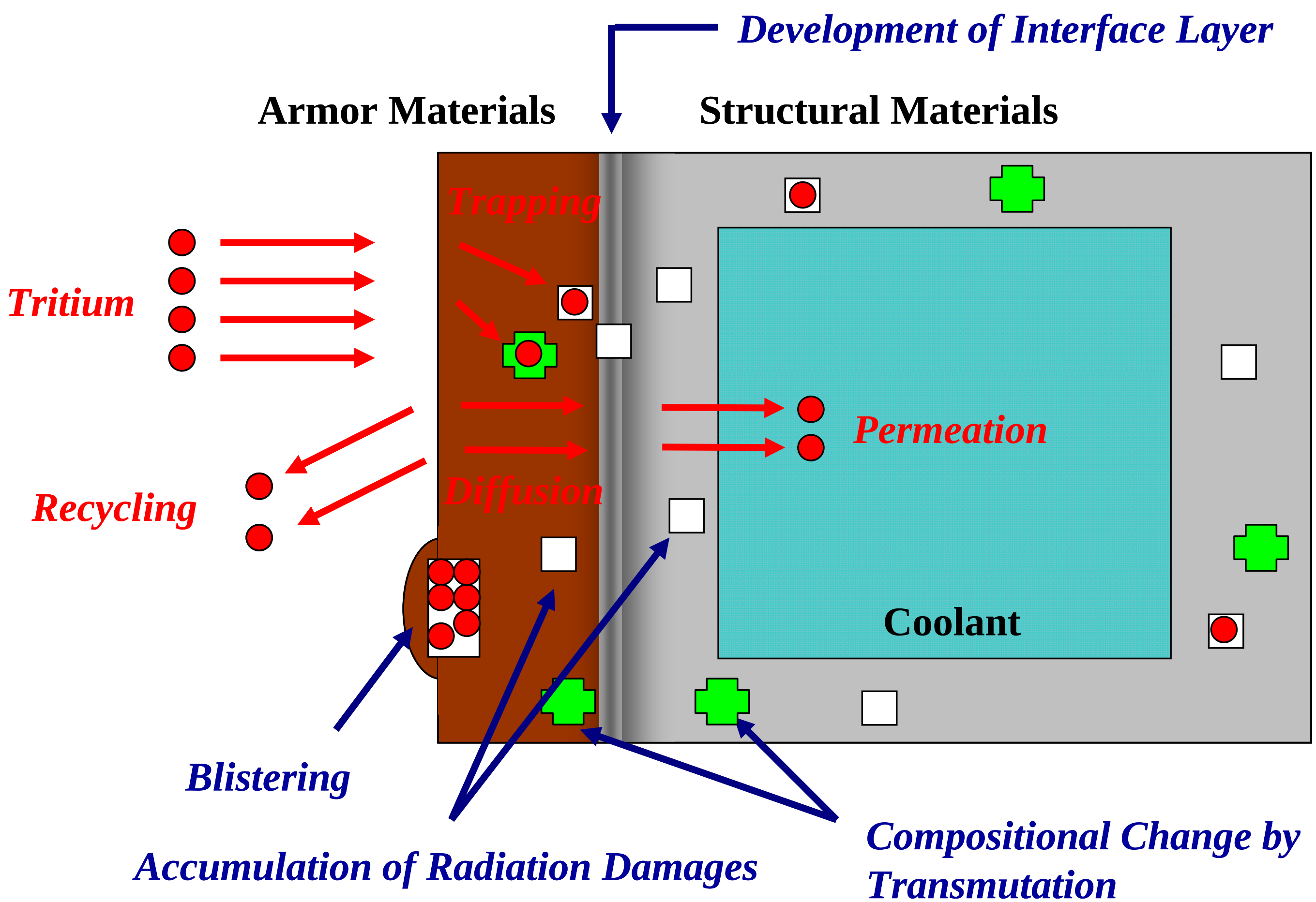
Selected to provide the basis for developing practical LM blankets of interest to US and Japan

- Fundamental Studies of 3D flow structures
  - Manifolds (consisting of expansions, bends, parallel channels, etc.)
  - Heat Transfer Enhancement structures (steps, ribs)
- Insulation techniques for pressure drop (and thermal) control
  - Technique 1: SiC/SiC or other FCI
  - Technique 2: Multi-layer ceramic/metallic coatings (or FCI)
- Heat and mass transfer studies
  - Turbulent heat transfer in poorly conducting channels
  - MHD Mixed convection in fusion blankets
- Computational Model Development
  - High Hartmann number complex geometry simulations
  - Fundamental turbulence models in different regimes
  - Transport effects (heat, tritium, corrosion) due to MHD velocity / temperature coupling – towards integrated system modeling and virtual TBM

Task 2-1 Studies: *Irradiation-tritium Synergism*

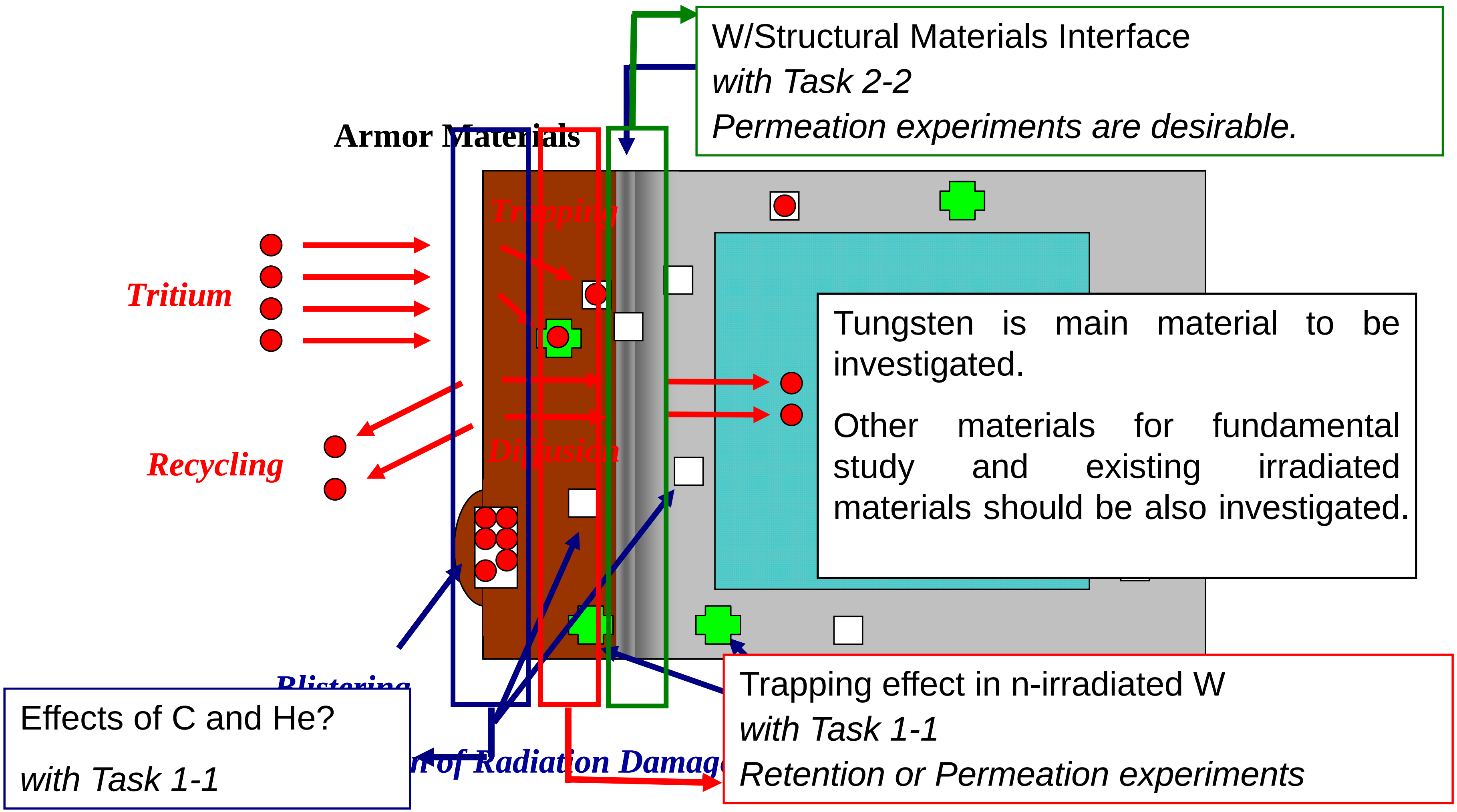
Motivation for collaboration

Purpose is to understand T behavior in first-wall components after n-irradiation.



Task 2-1 Studies, cont.:

Integration with other tasks



Task 2-1 Studies, cont.:

Planned Activities

- Neutron irradiation by rabbit capsules in ORNL
- Tritium Retention/Permeation tests at INL

Developments of techniques for tritium experiments with small samples are challenging.

First stage: Retention experiments with TPE

Second stage: Permeation experiments with Ion Beam Device

Sample holders for small specimens are required.

- Tritium exposure experiments on irradiated samples will occur both with TPE and TRitium Ion Implantation eXperiment (TRIIX)

Task 2-1 Studies, cont.:

Planned Activities, cont.

- Irradiation tests of pure W
  - PM, recrystalized or stress-relieved
  - doses of 1 and 5 dpa and temperatures of 300˚C and 500˚C
- Coating Material (with Task 2-2)
  - W on ferritic steel, ODS, SiC or V alloys by plasma spray etc.
- Mixing layer on W
  - Preliminary idea is to implant C ions on W surface prior to neutron irradiation
- Other materials
  - Investigation on Ni including single crystal has been proposed as fundamental study because of small surface effects
  - Existing irradiated materials from previous collaborations

Task 2-2 Studies: *Joining and Coating Integrity*

Planned Activities

- Dissimilar Joint Development:
  - To develop and evaluate the irradiation performance of dissimilar joints of plasma facing materials and first wall structural material
- Material Research/Design Interaction:
  - Close collaboration of materials research and reactor engineering design provide a highly efficient manner of advanced materials research/development
- Fundamental studies to cover wide materials:
  - Cost effective approach to perform experimental/modeling studies on other joint materials
- Main issues for direct investigation:
  - coating development
  - irradiation and PIE for mechanical properties/behavior
  - helium/hydrogen synergism effects
  - hydrogen behavior at weld interfaces

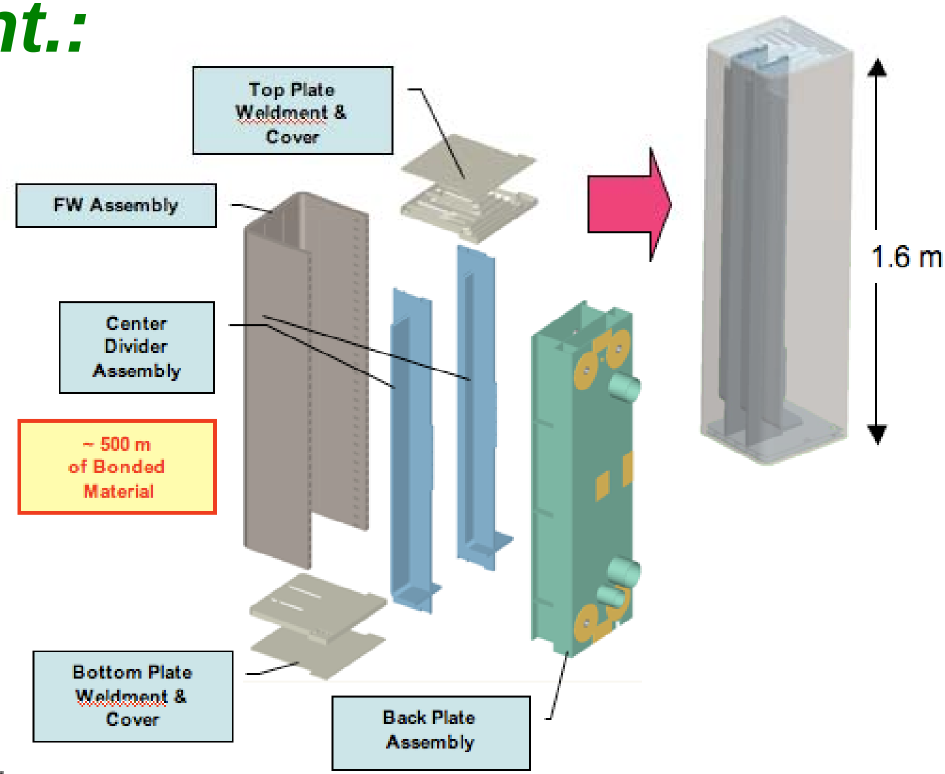
Task 2-2 Studies, cont.:

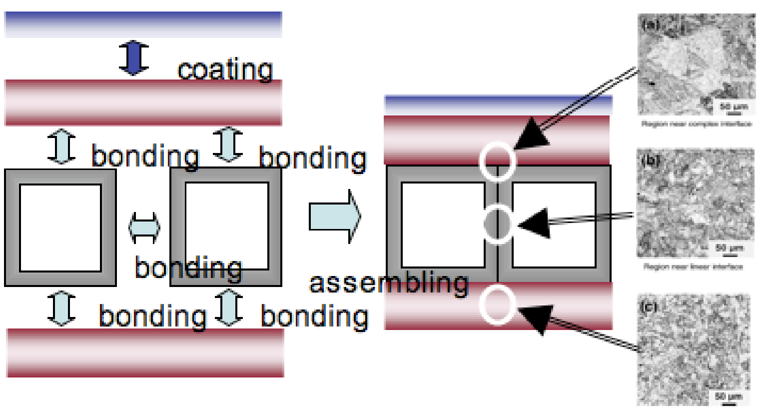
Joint & Weld

- Complex structure
- Multi-structure
- Many methods
- Coating

Irradiation effects

Tritium behavior



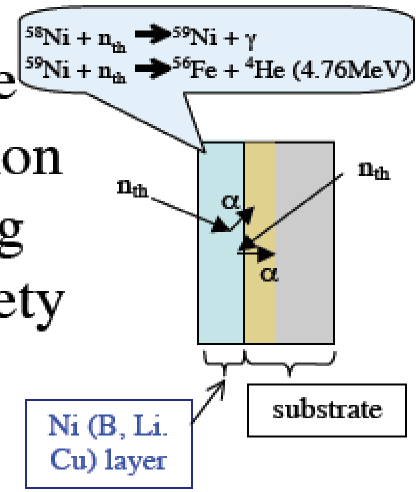


- Hot Isostatic Pressure Welding (HIP)
- Diffusion Welding (DW)
- E-Beam Welding (EBW)
- Laser-Welding (LW)
- TIG welding
- Plasma Welding (PW)
- Friction Stir Welding (FSW)
- PVD coating
- CVD coating
- Plasma-spray coating

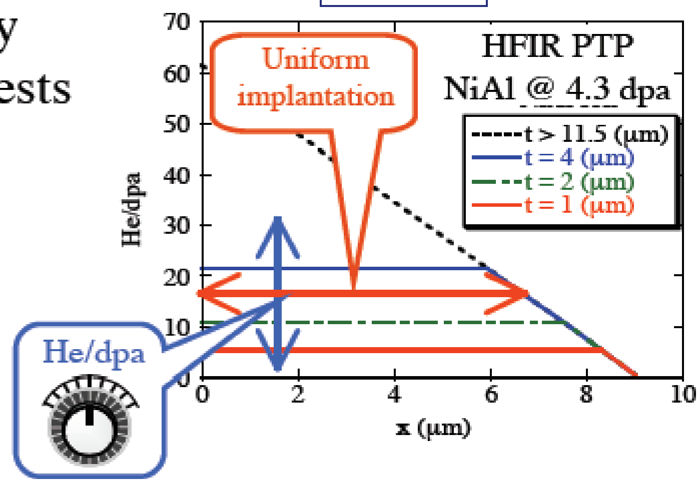
Task 2-2 Studies, cont.:

In Situ He Implanter Experiments

- (n,α) in mixed spectrum reactors to produce controlled He/dpa at fusion relevant condition
- Avoids most confounding factors vs. doping
- Apply to any material - e.g., SiC and a variety of specimens.
- Uniform implantation typically > 6 μm
  => sufficient for microstructure study
  => micro to nano scale mechanical tests



capsule	JP26			JP27			JP28/29		
Tirr (ºC)	300	400	500	300	400	500	300	400	500
dpa	4.3	4.3	10	11.4	11.4	24.5	21	45	50
He/dpa	5	5							
	11	11	9	10	10	13	10	11	11
	21	21	19	20	20	26	21	22	22
			38	50	50	65	52	55	55
He (appm)	23	23							
	45	45	94	115	115	320	200	500	550
	91	91	188	230	230	640	410	1000	1100
			380	570	570	1600	1030	2500	2770



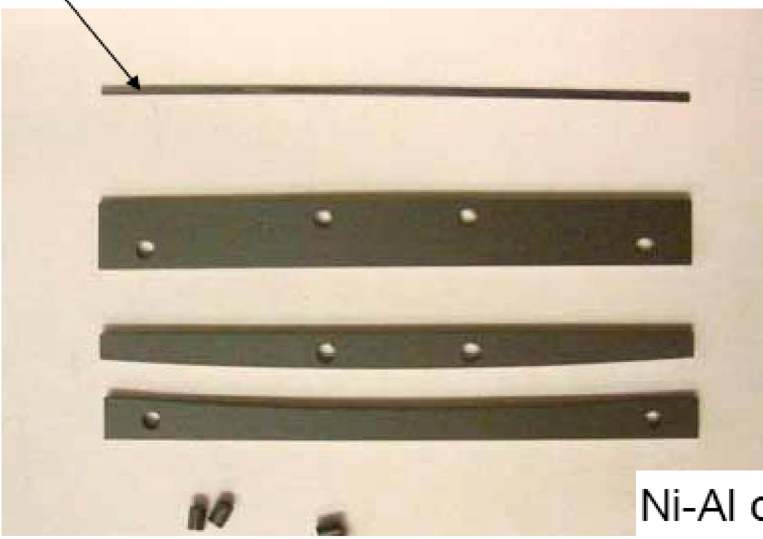
Task 2-3 Studies: Dynamic Deformation

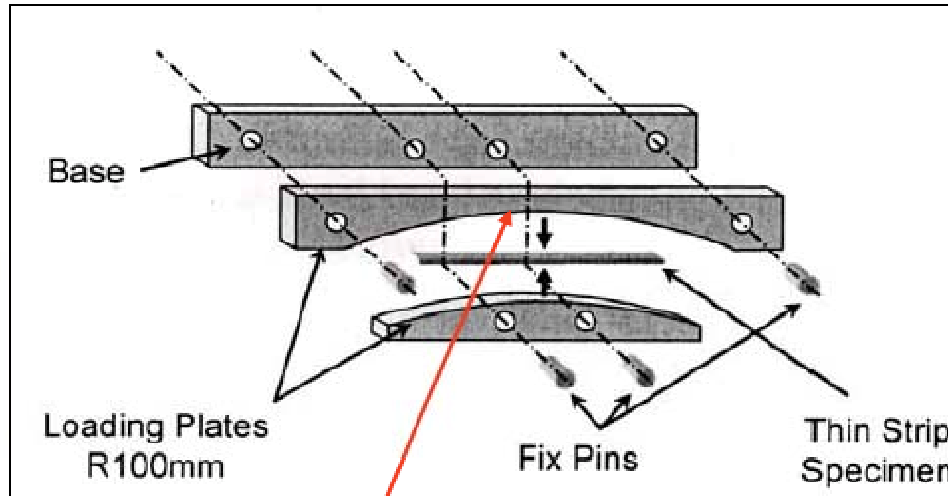
Areas of interest (still resolving what to achieve)

- Fundamental Studies of Advanced SiC/SiC Composites
  - — Post Irradiation Experiments of 18J SiC tests (800,1100,1300C, ～5dpa)
    Mechanical properties of advanced composite, compatibility studies of SiC-CB systems
  - — Irradiation Experiments of Rabbit (200～1200C, 0.1～10dpa)
    Tensile strength, Bend strength of Composites etc.
- Dynamics Deformation Behavior under Irradiation
  - — Stress relaxation method
  - — Creep Mechanism of SiC and SiC/SiC
- Synergistic Effects of Helium and Tritium on Dynamic Deformation Behavior of SiC/SiC Composites
  - — In-situ He and T Injection during Creep Experiment
- Tritium Inventory Evaluation of SiC
  - — Post-irradiation Measurement by TDS
  - — T Distribution Measurement in Creep Tested Specimen by IP
- Modeling of Dynamic Deformation and Synergistic Effects
  - — Mechanistic modeling
  - — Life Time Evaluation of SiC/SiC Composites

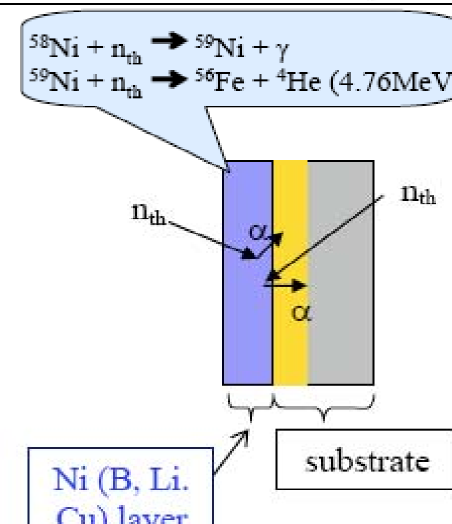
Task 2-3 Studies, cont.:

BSR Specimen（CVD-SiC）
$0.8^{T} \times 2.0^{W} \times 50^{L}$mm





Ni-Al coating as He injector

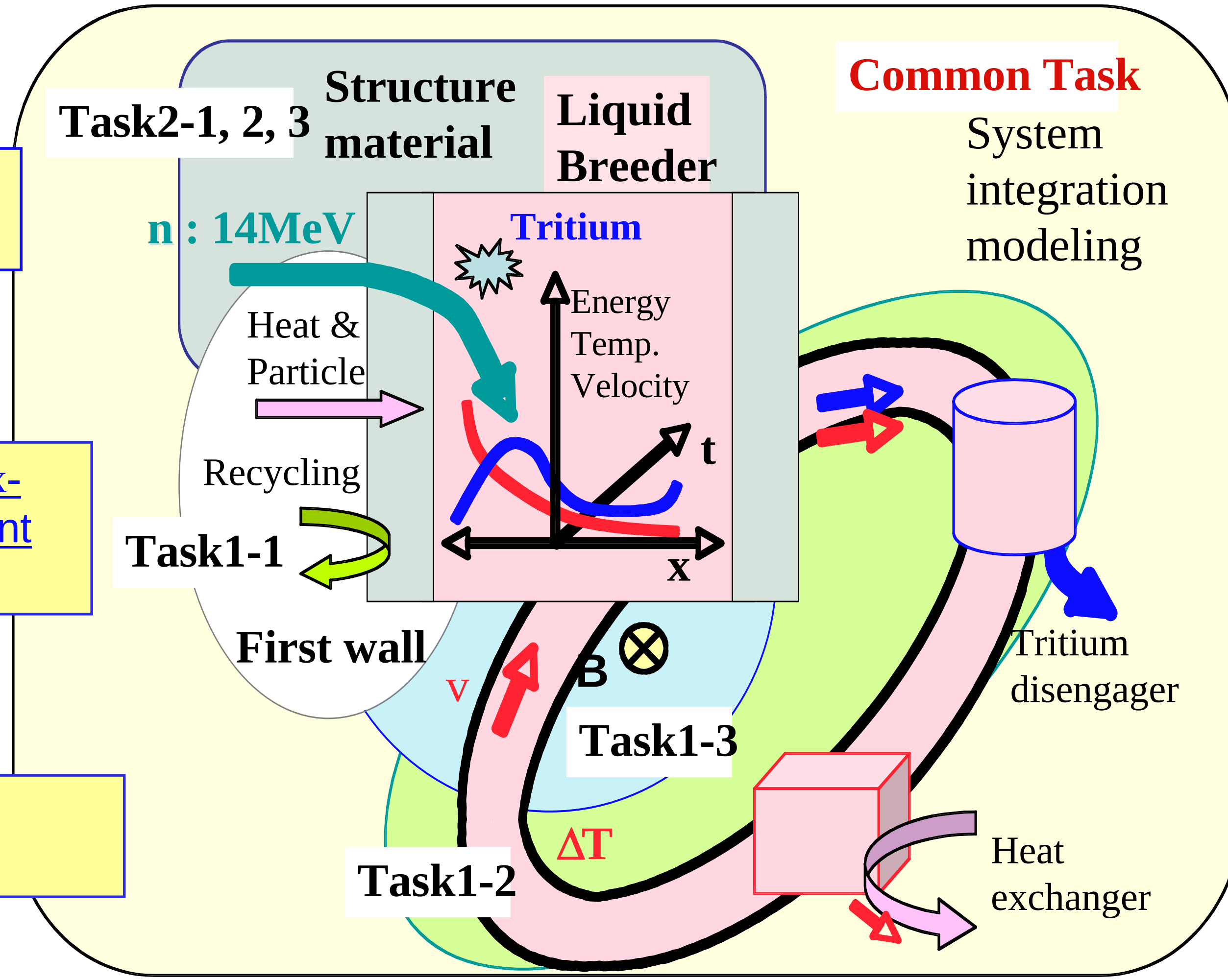


Task 3 Studies: MFE/IFE system integration modeling

Tritium and thermofuid system designs

A burning core plasma and the ex-vessel environments are important boundary conditions.

Approach: cross-connect TITAN tasks



Fiscal Year	2007	2008	2009	2010	2011	2012

Interim report

Task	Subtask	Activities
Task1 tritium and mass transfer blanket	1-1 Tritium and mass transfer in first wall	Equipping PISCES First wall test Deuterium-Materials test Combined conditions Equipping TPE First wall test Tritium-Materials Test First wall/blanket test
	1-2 Tritium behavior in blanket systems	Tritium solubility measurement Tritium permeation test Tritium recovery test Flow test
	1-3 Flow control and thermofluid modeling	Modification of test section MHD thermofluid test System construction MHD pressure drop test for liquid breeders
Task2 irradiation n-tritium synergism	tritium synergism	Design, Fabrication, Irradiation, Disassembling and PIE
	2-2 Joining and coating integrity	Design, Fabrication, Irradiation, Dis assembling and PIE
	2-3 Dynamic deformation	Design, Fabrication, Irradiation, Disassembling and PIE
Common Task System integration modeling		PMI, T/mass transfer、Thermofluid、 Integration, System analysis for MFE/IFE

Conclusion

- Program address several technical items for longer-term fusion reactor power handling technologies
- Fits within scope of US FNT
- May provide some preliminary data to arguments made in ARIES-Pathways study
- Will have more to report when the funding actually starts!