16th ANS Topical Meeting on the Technology of Fusion Energy
Sept. 14-16, 2004, Madison, Wisconsin

# O-I-5.4
# Plan and Strategy for ITER Blanket Testing in Japan

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# Blanket Development Strategy in Japan

- The Fusion Council of Japan has established the long-term research and development program of the blanket in 1999.

- Japan Atomic Energy Research Institute (JAERI) will pursue solid breeder blanket cooled by high pressure and temperature water for ITER Test Blanket Module (TBM) and the fusion power demonstration plant.

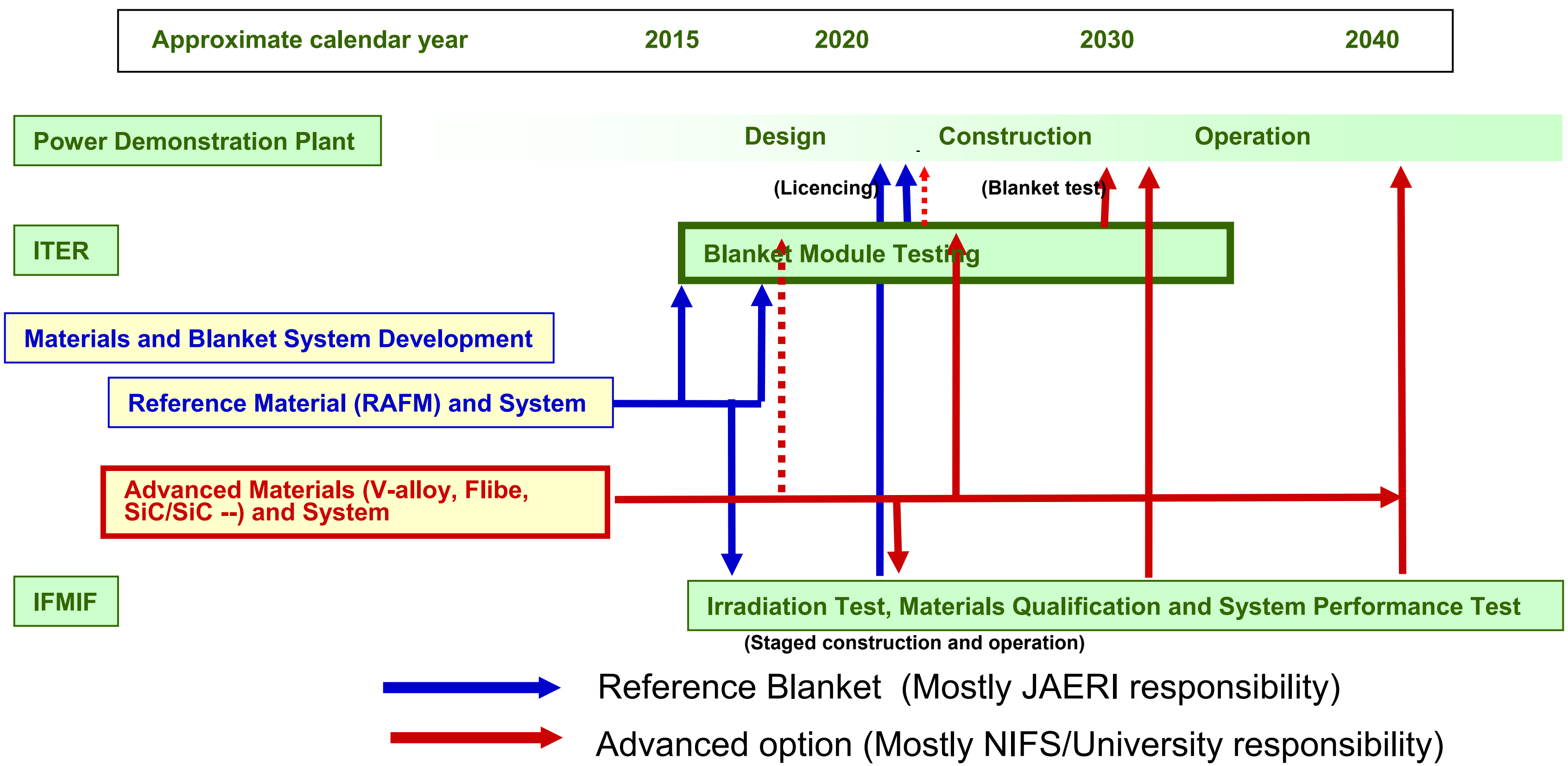
- Universities and National Institute for Fusion Science (NIFS) will develop advanced concepts: liquid breeder blankets and high temperature solid breeder blanket will be attempted for the module testing in ITER and the fusion power demonstration plant.

# Roadmap of Materials and Blanket Development in Japan

**Milestones to the fusion power demonstration plant**

(1) By ITER TBM testing, demonstrative data of blanket functions will be obtained in fusion environment.

(2) Together with the material irradiation data by IFMIF, the fusion power demonstration plant will be decided.



# Blanket Types Covered by Japan

Japan investigates the possibility of testing all types of blankets under Test Blanket Working Group (TBWG) framework with both of JAERI and universities/NIFS involvements.

**Primary Candidate**

- Solid breeder / RAFM structure / water cooled, He cooled (WSG-1 and WSG-3)

(Development is lead by JAERI)

**Advanced Blankets**

- Solid breeder / $SiC_f$/SiC / high temperature He-cooled (WSG-1)
  LiPb / $SiC_f$/SiC / high temperature He-cooled (WSG-2)
- Li self-cooled / V alloy (WSG-4)
- Molten salt self-cooled / RAFM (WSG-5)

(Development lead by universities and NIFS)

# TBWG Participation from Japan

| WSG | Blanket Type | Japan Member |
|---|---|---|
| WSG-1 | Helium-cooled Ceramic | A. HASEGAWA, Tohoku University<br>A. SHIMIZU, Kyushu University<br>M. ENOEDA, JAERI |
| WSG-2 | Helium-cooled LiPb | S. KONISHI, Kyoto University<br>A. KOHYAMA, Kyoto University |
| WSG-3 | Water-cooled Ceramic | A. KIMURA, Kyoto University<br>M. ENOEDA, JAERI (WSG-3 leader) |
| WSG-4 | Self-Cooled Lithium | T. MUROGA, National Institute of Fusion Sci.<br>S. TANAKA, the University of Tokyo |
| WSG-5 | Self-Cooled Molten Salt | S. TANAKA, the University of Tokyo<br>A. SAGARA, National Institute of Fusion Sci. |

# Solid Breeder Test Blanket Modules (WSG1, WSG3)

1. Japan is planning to test two types of Solid Breeder Test Blanket Modules (TBM), from the first day of ITER operation with Japanese own TBMs
- Water Cooled / $Li_2TiO_3$ / Be /RAFM (F82H)
- Helium Cooled /$Li_2TiO_3$ / Be / RAFM (F82H)

2. Organized R&Ds are being performed in JAERI, to realize ITER blanket module testing on time.

3. Elemental Technology R&D is almost completing. Now, Engineering R&D phase is expected for evaluation of TBM design relevancy and determination of detailed manufacturing specification of TBMs.

<u>R&D Achievement is presented in "O-II-5.1: Development of Solid Breeder Blanket at JAERI", in this meeting.</u>

# Plan of Module Testing for First 10 year Period of ITER

| Year | | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | 10 |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Mile Stone | | **First Plasma** | Full Field, Current & H/CD Power | | | Short Burn 200 MW | Q=10 500 MW | Q=10 500 MW 400 s | Non-inductive Current Drive | | |
| Operation | Baking & Conditioning<br>- Machine com-missioning<br>- Achieve good vacuum & wall condition | **H-Plasma**<br>- Machine commissioning with plasma<br>- Heating & CD experiments<br>- Reference scenarios with H | | | **D-Plasma (limited T)**<br>- Commissioning with neutron<br>- Reference with D<br>- Short DT burn | **Low Duty DT**<br>- Development of full DT high Q<br>- Development of non-inductive operation aimed at Q=5 | | | **High Duty DT**<br>- Improvement of inductive and non-inductive operation<br>- Demonstration of a high duty operation | | |
| Equivalent Number of Burn pulses (500 MW x 400 s) | | | | | 1 | 750 | 1000 | 1500 | 2500 | 3000 | 3000 |
| Cumulative Neutron Fluence at Test Port ($MWa/m^2$) | | | | | $9.9 \times 10^{-6}$ | 0.0074 | 0.017 | 0.032 | 0.057 | 0.087 | 0.12 |
| Installation & Commissioning | Basic Installation & Commissioning | | | For Activation Phase | | | For High Duty Operation | | | | Upgrade |
| Blanket Test Phase | | **System Checkout & Characterization**<br>**TBM W-1, TBM H-1**<br>**- Overall functions, instrumentation, flow control, heat loss, etc.**<br>**- Integrity to EM & thermal loads**<br>**- FW heat removal**<br>**- Ferritic steel effect on plasma control** | | **- Remote handlin g test during shut down** | **TBM W-2,3, TBM H-2,3**<br>**- Neutronics environment measurement**<br>**- Indication of tritium production & extraction**<br>**- Indication of heat generation** | | | | **Performance Test**<br>**TBM W-4, TBM H-4**<br>**- High grade heat generation and extraction**<br>**- Continuous tritium recovery**<br>**- Electricity generation with TBM w-3** | | |
| **Key Issues Of Testing** | | **System Checkout & Test Environment Measurement** | | | **Neutronics Measure.** | **TPR Test** | **Thermo-mechanical Test T and Heat Production (long term)** | | | | |

# Key Points of Japanese Solid Breeder TBM Designs (WSG-1 and WSG-3)

(1) Slit structure was adopted for TBM structure because of the requirements of endurance to electro magnetic force in VDE and internal over-pressure.

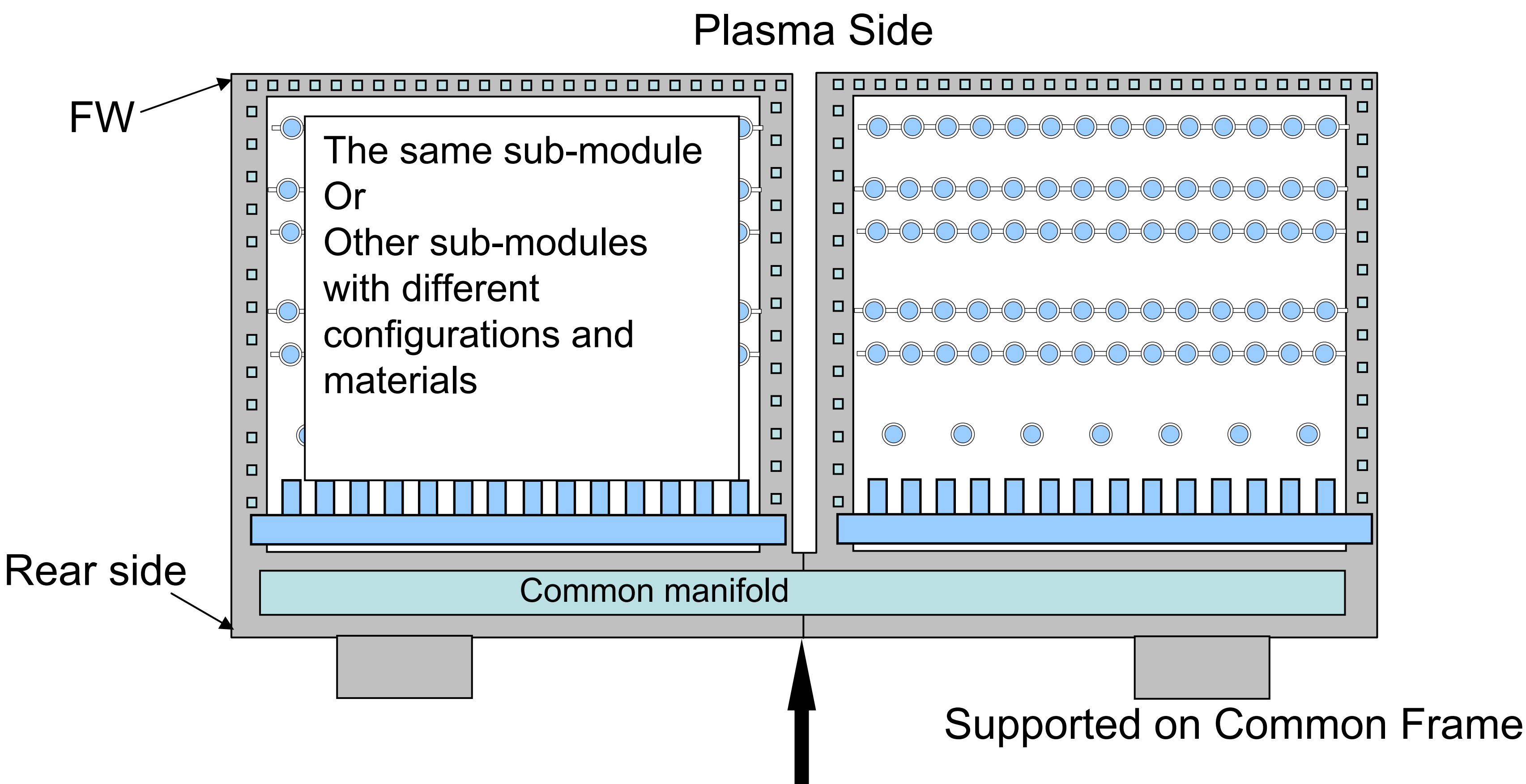
(2) Detailed structure was developed with preliminary consideration of fabrication procedure. Coolant flow routing, pressure relief routing and instrumentation cables routing are incorporated in the design.

(3) Analyses of TBM function and structural integrity

- Nuclear analyses for TBR, nuclear heating rate, induced activation
- Thermal-hydraulic analysis of coolant to confirm heat removal
- Thermo-mechanical analysis of FW for structural integrity
- Thermo-mechanical analysis of module box structure on endurance with internal over-pressurization
- Tritium inventory/release analyses for function confirmation and clarification of design input to tritium recovery system
- Safety analysis to show ITER safety requirement

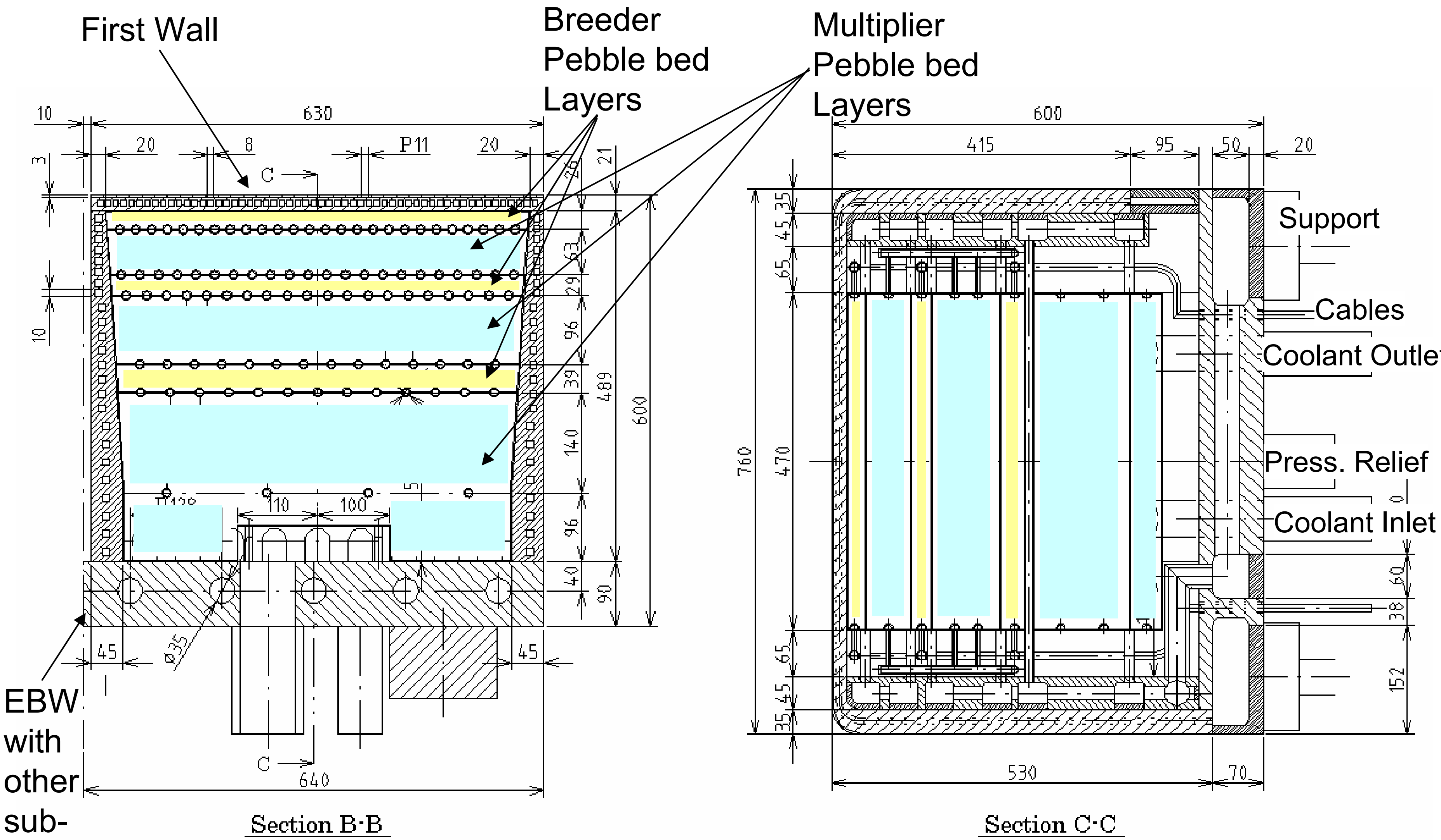
# Concept of He Cooled Solid Breeder TBM (WSG-1)

Slit structure was adopted. The TBM is divided into sub-modules. Sub-modules are welded to form integrated module structure.




Electron beam welding of two sub-modules. Then, common manifolds and rear wall equipments such as , keys and flexible supports will be fabricated.

# Structure of He Cooled Solid Breeder TBM (WSG-1)

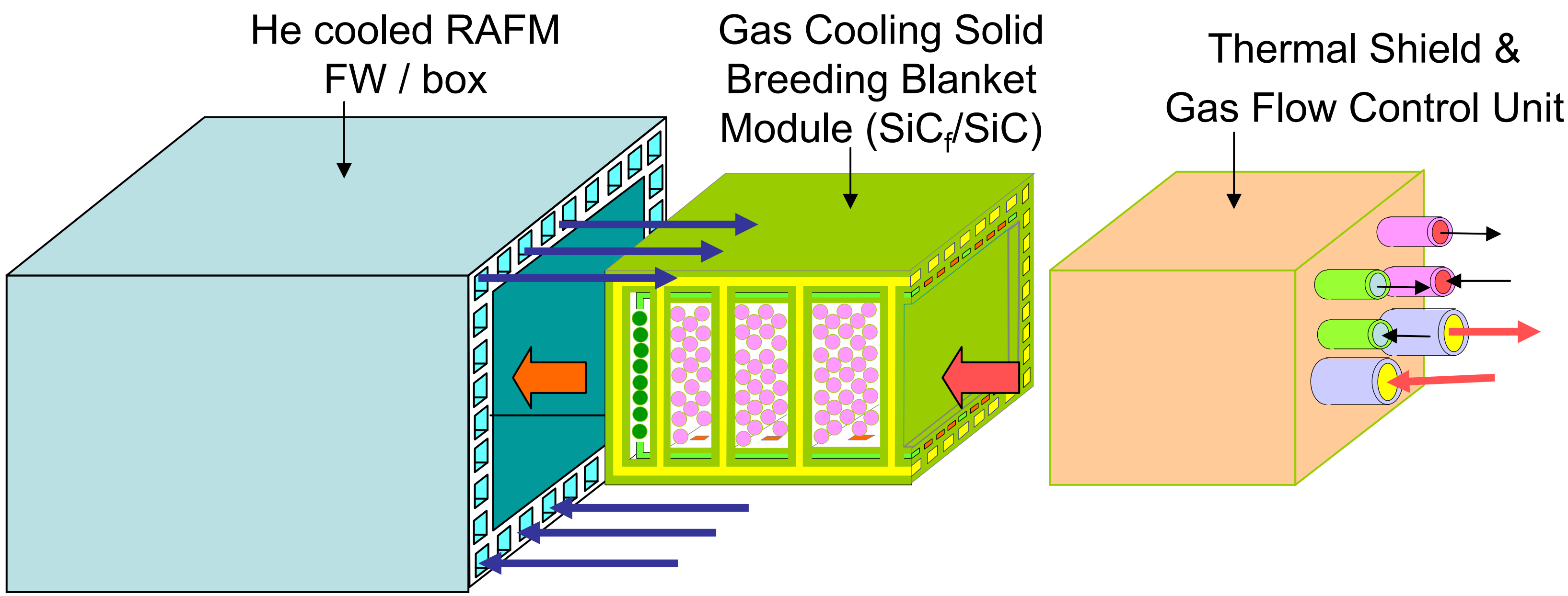


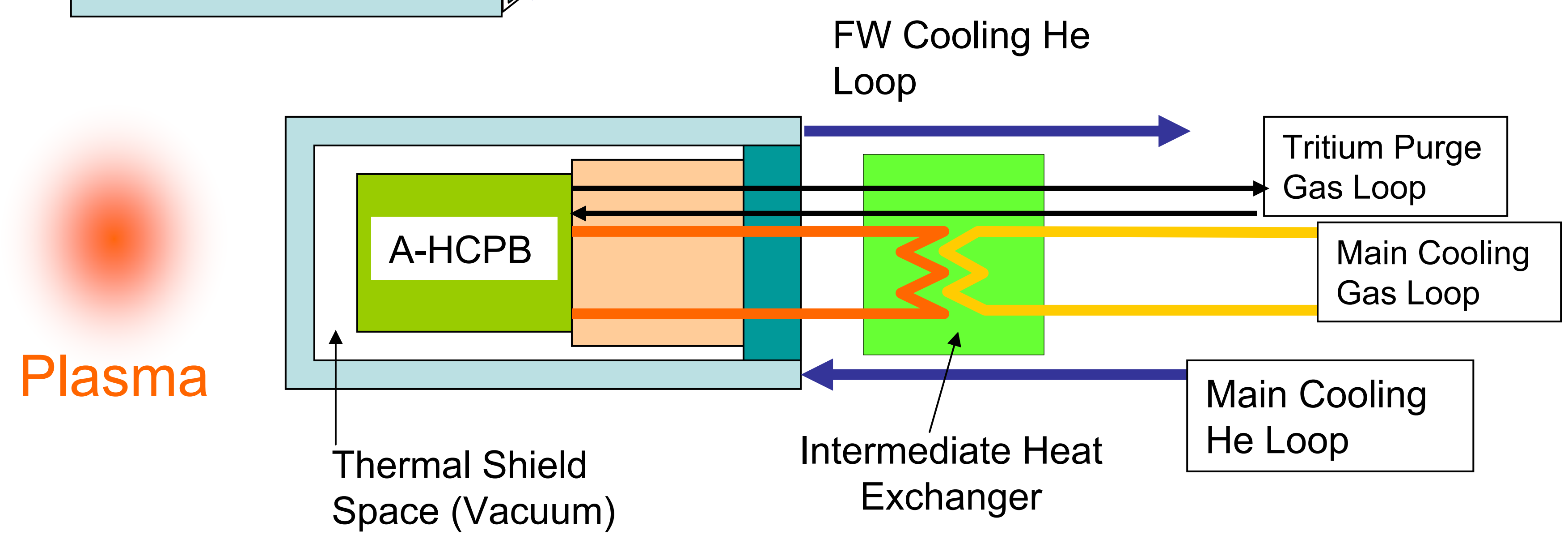

Horizontal Cross Section

Vertical Cross Section

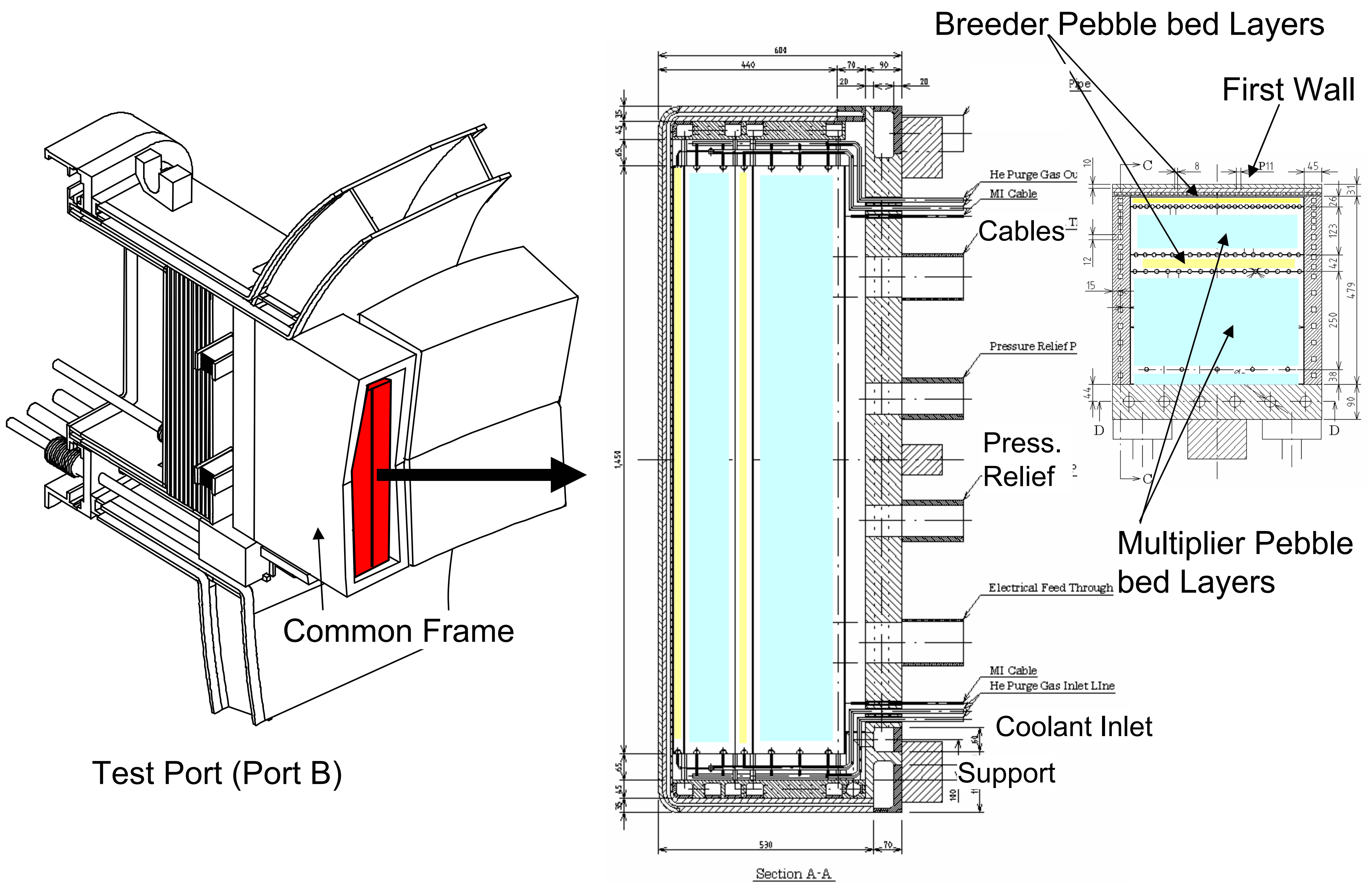
# Concept of ITER Test Blanket Module for Advanced Material System

[ Test Module of Advanced Helium Cooling Pebble Bed System using $SiC_f/SiC$ ]



# Configuration of Water Cooled Solid Breeder TBM (WSG-3)



# He Cooled LiPb – SiC$_f$/SiC Blanket (WSG-2)

## ITER TBWG/WSG2 and possible JA involvement

○SiC-He concepts not planned in TBWG

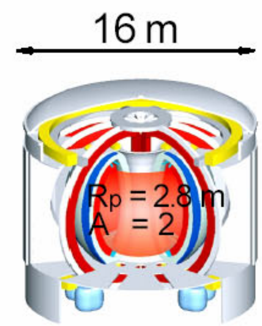
○LiPb-He cool – SiC concepts attracts interests of reactor designers

－EU：Tauro,PPCS-c,d

－US:ARIES-AT,ST

－JA:Vector

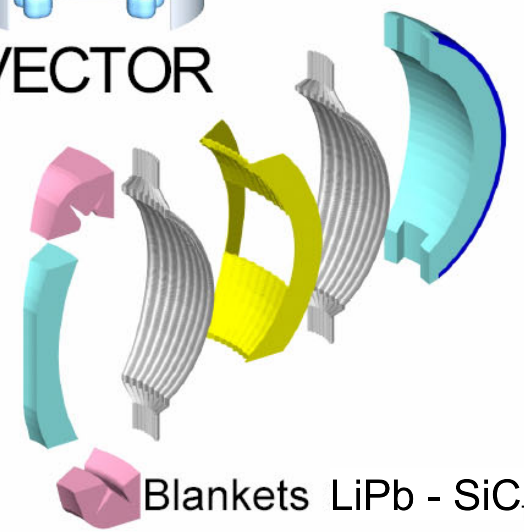
→HCLL concept can improve with SiC inserts test




VECTOR

○Possible area of Japanese contribution

- SiC Compatibility with LiPb
- SiC inserts development
- tritium systems design and evaluation



Blankets LiPb - SiC$_f$/SiC

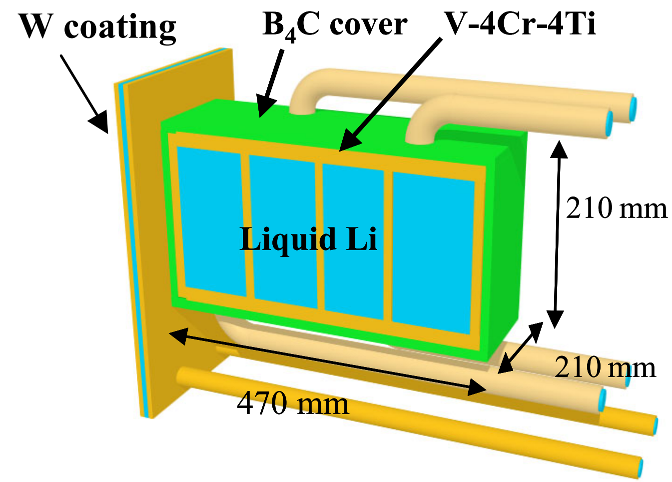
# WSG-4 (self-cooled Li/V blanket) -- Strategy and Plan

**Support Russian DDD (Li/Be/V module)**

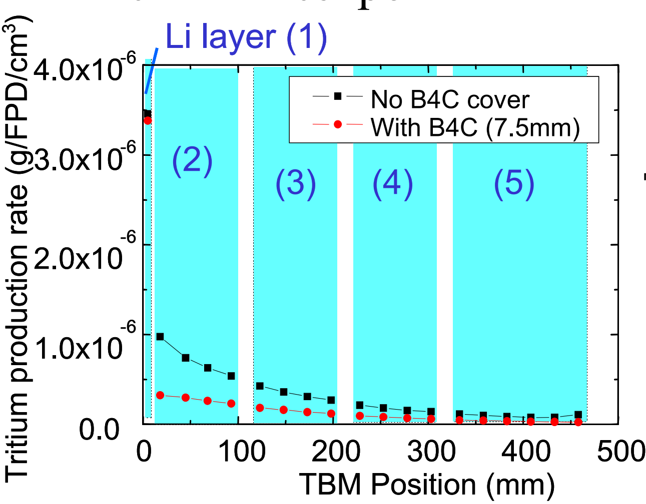
- Vanadium alloy database
- MHD insulator coating development

**Design Li/V (no Be) module as an alternative concep**

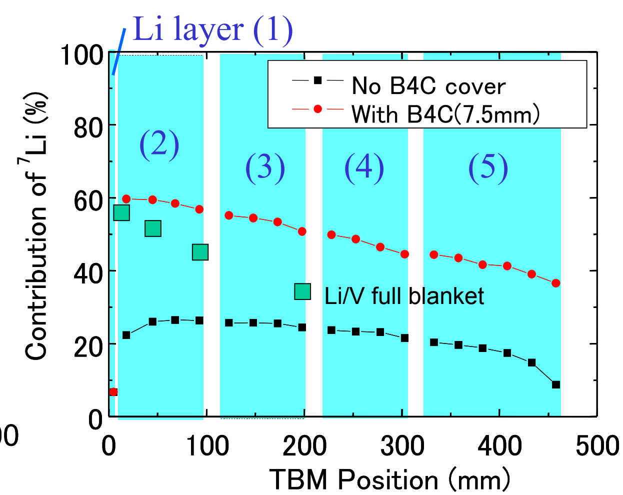
- Demonstrate T self sufficiency in Li/V blanket
- Thick structure with slow Li flow for verifying neutror transport calculation
- Test volume limited to observe ITER Li volume limit
- $B_4C$ shield for simulating Li/V full blanket condition in the ITER test port



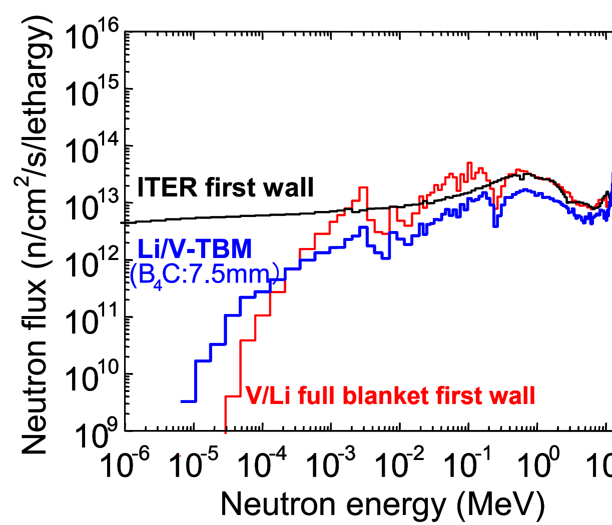

Li/V TBM Design




Tritium Production Rate




Contribution of $^7Li(n,n\alpha)T$ (enhanced by $B_4C$ cover)




Neutron Spectra

# WSG-4 (self-cooled Li/V blanket) -- Supporting Researches

## Blanket design

Tritium self-sufficiency for Li/V blanket was demonstrated by neutronics calculation in Tokamak and Helical systems → (**Poster PI-21,24**)

## Development of vanadium alloys

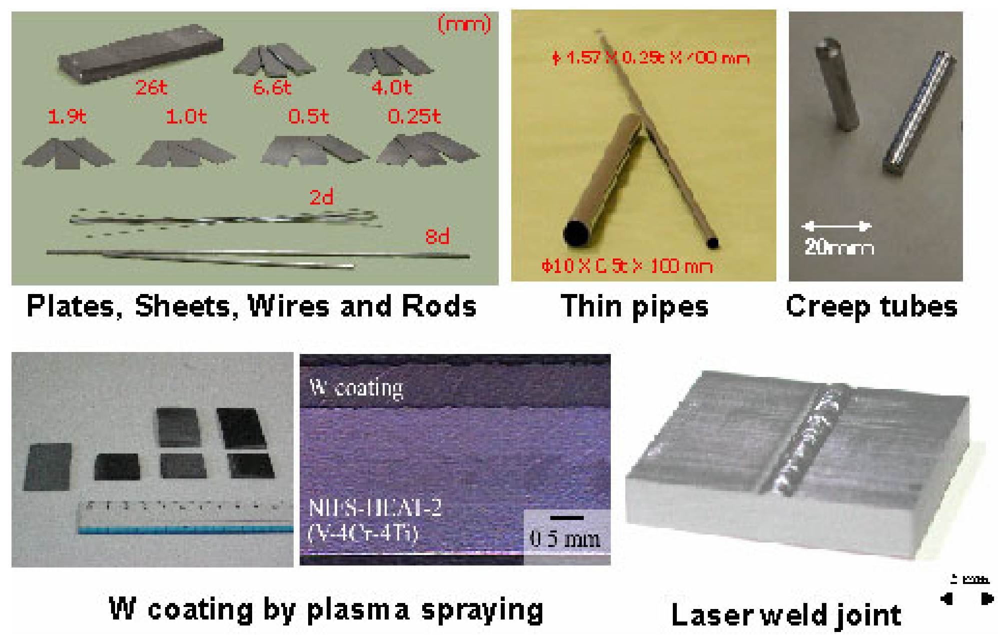
Fabrication technology was highly enhanced by recent researches
Feasibility of W coating on V-4Cr-4Ti was demonstrated → (**Poster PI-39**)
Manufacturing V-alloy test module with high quality is feasible

## MHD insulator coating

(1) PVD coating of $Er_2O_3$, $Y_2O_3$, (2) Two layer coating with $Er_2O_3$ and V-alloy, and (3) in-situ $Er_2O_3$ coating are under development

## T recovery

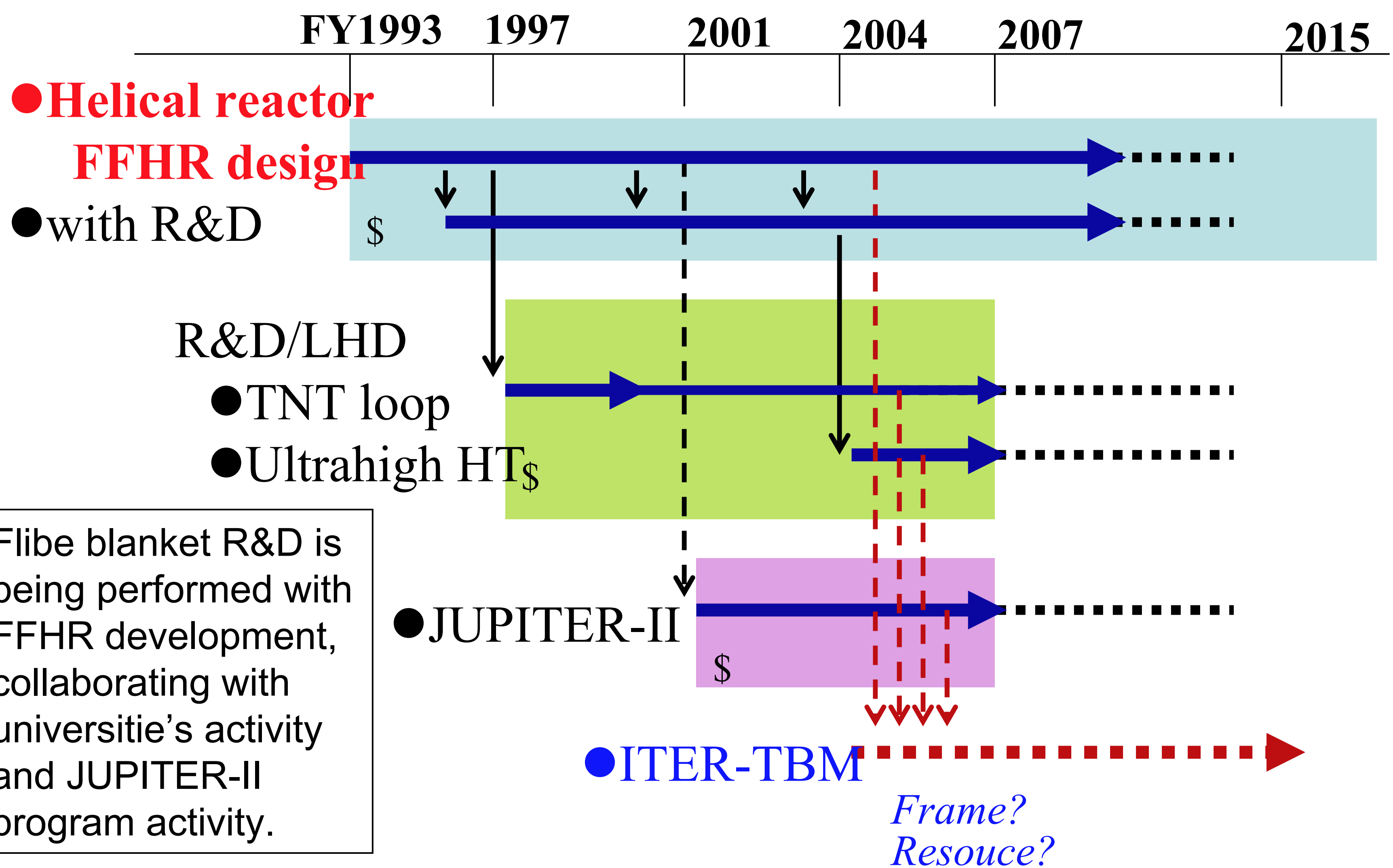
Feasibility of gettering T by Y was demonstrated, which can be applied to IFMIF and ITER-TBM
(Fukada, IFMIF-KEP)




V-4Cr-4Ti alloy products

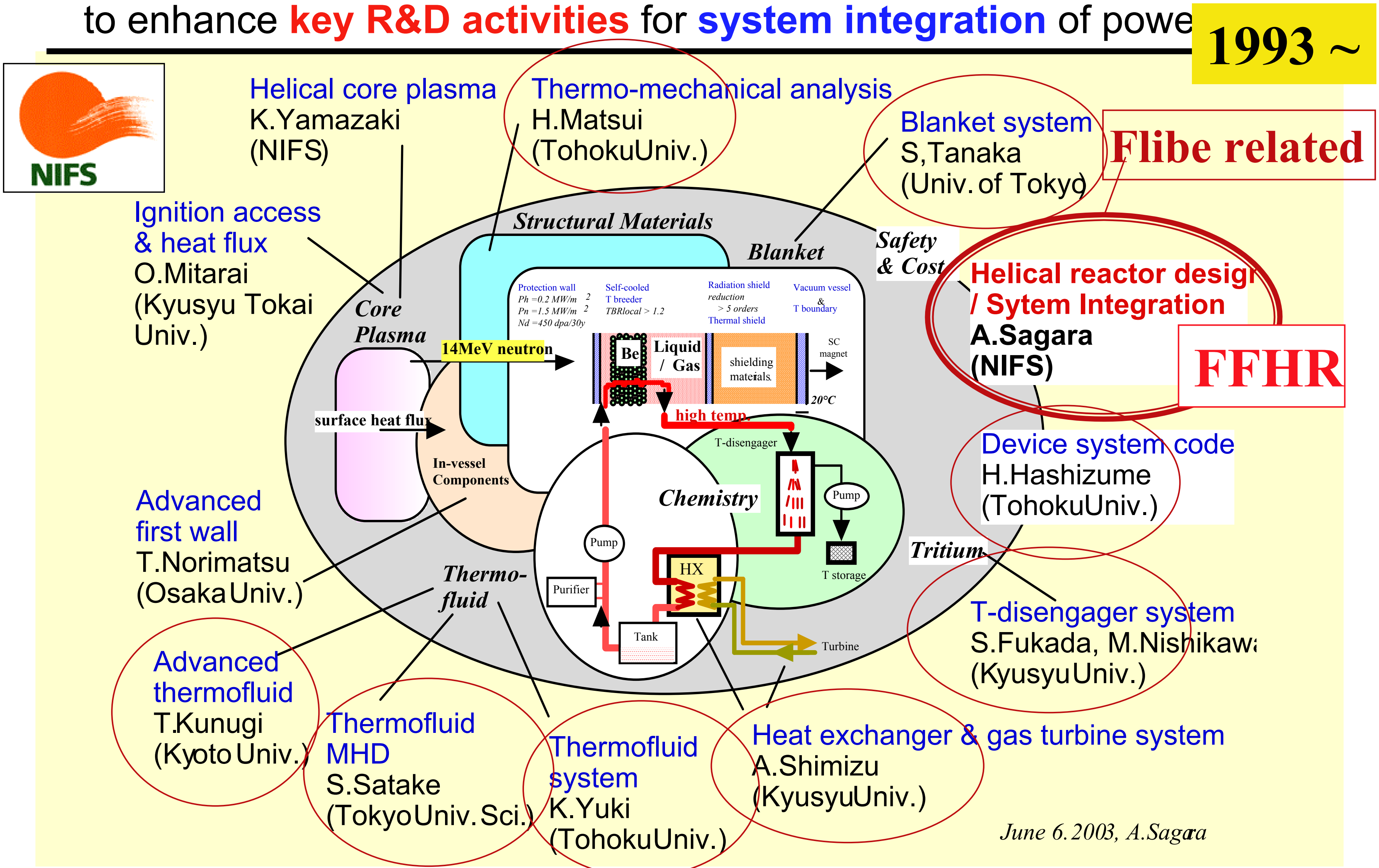
# Present R & D activities on Flibe blanket in Japan

*Presented by A.Sagara(NIFS) , Feb.'04*




Flibe blanket R&D is being performed with FFHR development, collaborating with universitie's activity and JUPITER-II program activity.

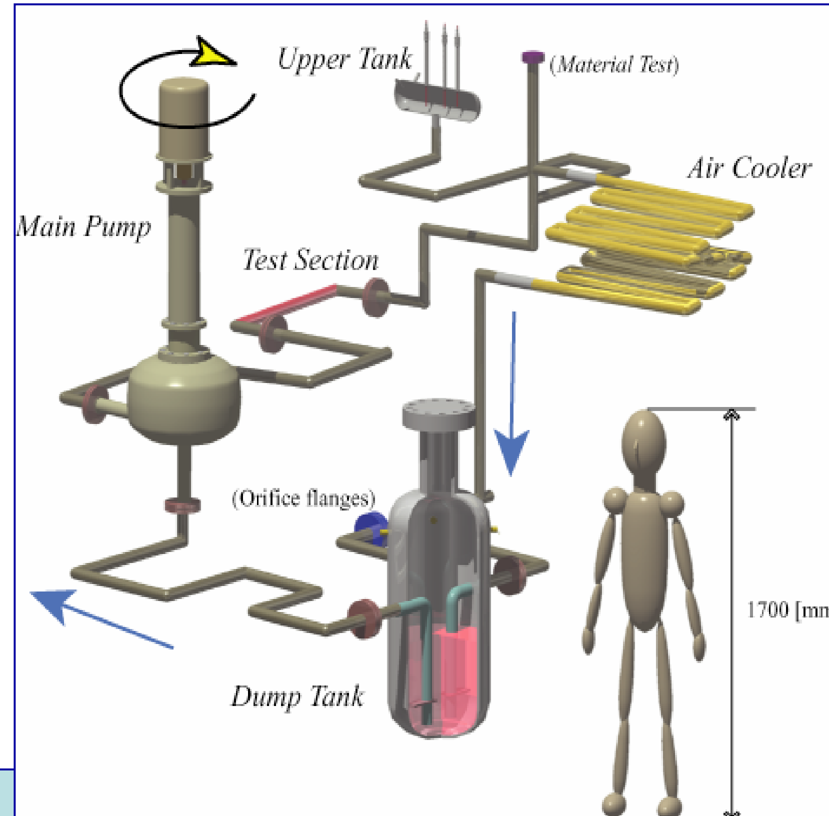
● **Reactor design collaborations in NIFS** are on going with international activities to make clear **key engineering issues** and to enhance **key R&D activities** for **system integration** of powe

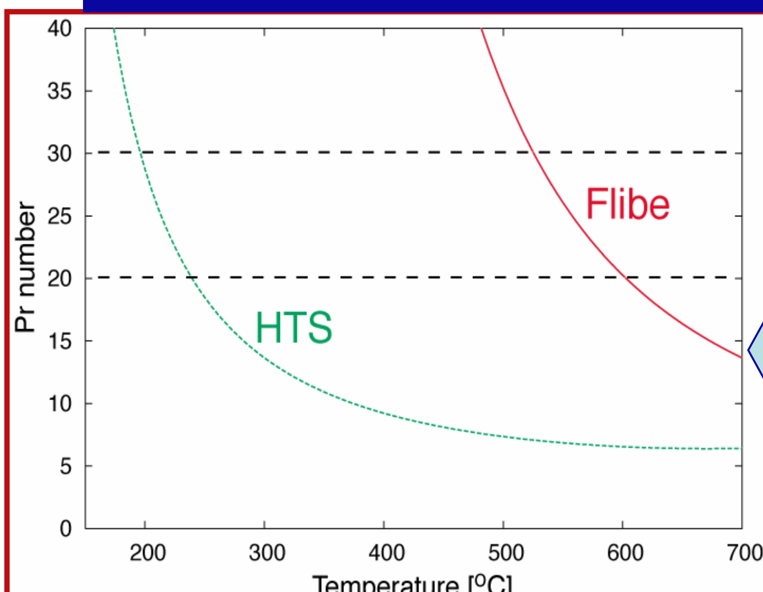


# Tohoku-NIFS Thermofluid Loop for molten salt (1997~) has accumulated high Temperature Device Technology

## TNT loop .

- u = 8 ~ 20 L/min
- T < 600° C
- V ~ 0.1m$^3$
- P < 0.7 MPa.
- operation ~ 30 kW
- air cooler < 80 kW
- material = ss316




Now HTS (heat transfer salt) is used as a simulant for Flibe

## HTS

53% $KNO_3$
40% $NaNO_2$
7% $NaNO_3$
$T_m$ =142° C

# Conclusions

(1) Blanket Module Testing in ITER, and the material irradiation tests by IFMIF are most important milestones to the fusion power demonstration plant.

(3) Solid Breeder Blanket is the basic option and first candidate. R&Ds are being performed intensively and systematically.

(4) For advanced options of blankets, Blanket Module Testing in ITER is strongly interested in a certain stage, if possible and feasible from the first day of ITER. R&Ds are showing steady progress.

(5) Japan investigates the possibility of testing all types of blankets under Test Blanket Working Group (TBWG) framework with both of JAERI and universities/NIFS involvements.