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RECENT ACCOMPLISHEMNTS AND FUTURE PROSPECTS OF MATERIALS R & D IN JAPAN

Institute of Advanced Energy
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AGENDA

1: Japanese Fusion Materials R & D Activities and Strategy

2: Research Emphasis and Status

- a: Reduced Activation Ferritic/Martensitic Steels
- b: Vanadium Alloys
- c: SiC/SiC Composite Materials
- d: Refractory Alloys
- e: Modeling and Mechanistic Studies

3: Key Issues for Materials Development

4: Summary

Materials R & D Strategy (1)

- Mission-oriented and time-driven to meet ITER and DEMO schedules
- Long-range fundamental studies and development of advanced materials are also promoted
  - Long lead time necessary for their development
- Planned to proceed with IFMIF and ITER in a coordinated way
  - IFMIF as a crucial facility for materials qualification
  - ITER blanket module tests as an important milestone for technology integration

Materials R & D Strategy (2)

- Categorized into reference and advanced materials
  - **Reference materials** for timely realization of DEMO and first generation power plant:
    Reduced Activation Ferritic Martensitic Steels
    - Having the most matured industrial infrastructure and database
    - Use of Oxide Dispersion Strengthened (ODS) RAF(M) considered as a method to increase the operation temperature
  - **Advanced materials** for increasing attractiveness of the power plant (COE, environmental benignity):
    V alloys and SiC/SiC Composites
    - Having the very rapid progress under the very extensive R & D efforts

Materials R & D Strategy (3)

JUPITER-II: Japan/US collaborative program (FY2001-2006)

-Materials integration utilizing reactor irradiation and related basic research for advanced blankets-

Emphasizing the importance of strong tie of "Materials" with "Blanket Engineering".

- concentrating

- ***1: Self-cooled Liquid Blanket with V (RAF, SiC/SiC)***
- ***2: He Cooled Solid Blanket with SiC/SiC***
- ***3: Modeling and Mechanistic Studies***

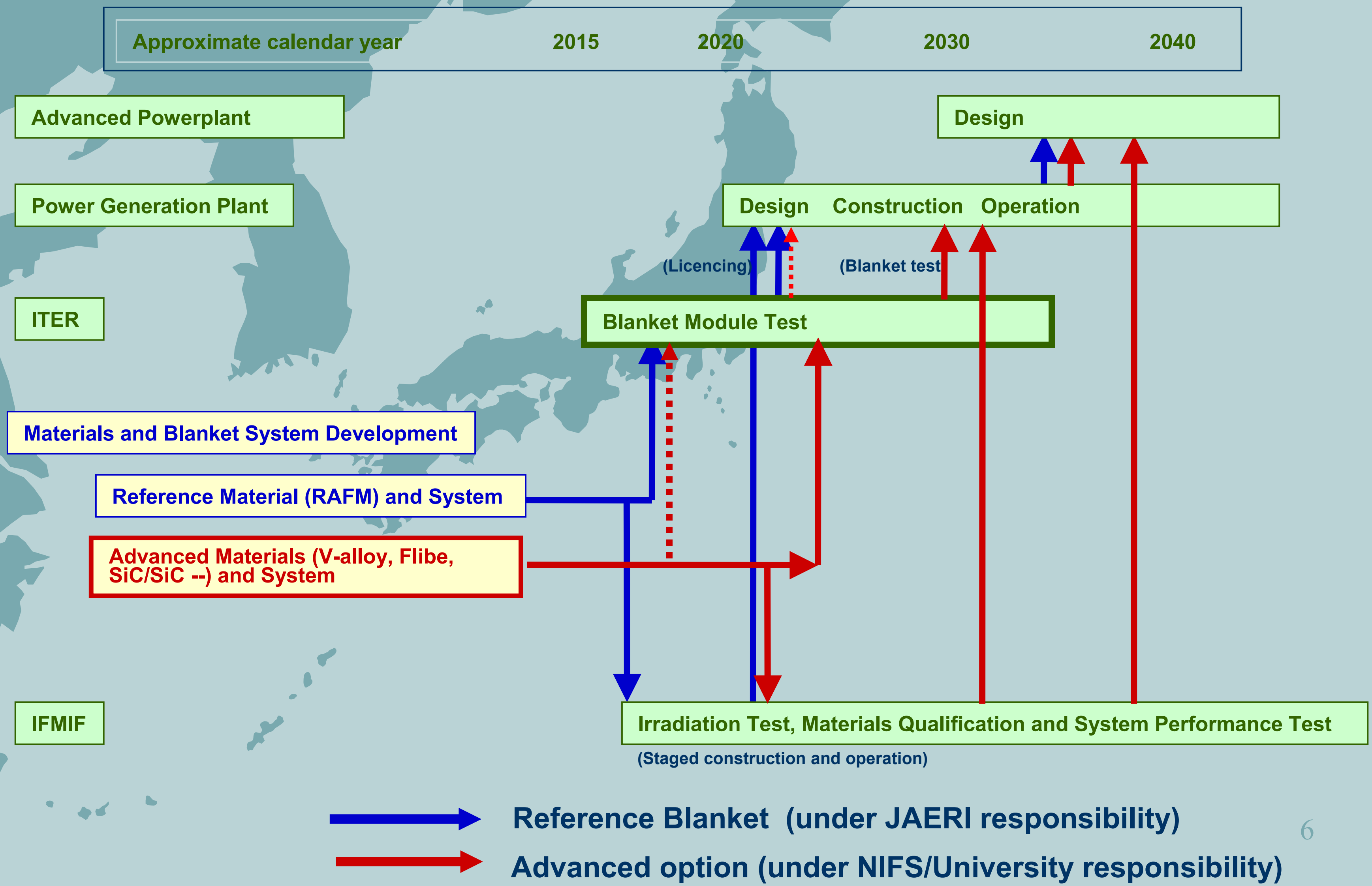
JAERI/ORNL Phase IV and V: (FY1999-2003, 2004-2008)

Emphasizing the importance to establish Clear understandings and Basis for blanket design from "Neutron Radiation Effects up to High DPA"

- concentrating

- ***1: water cooling ceramic breeder/Be blanket with RAF***
- ***2: He cooling solid breeding blanket with RAF***

Roadmap for Materials and Blanket Development in Japan



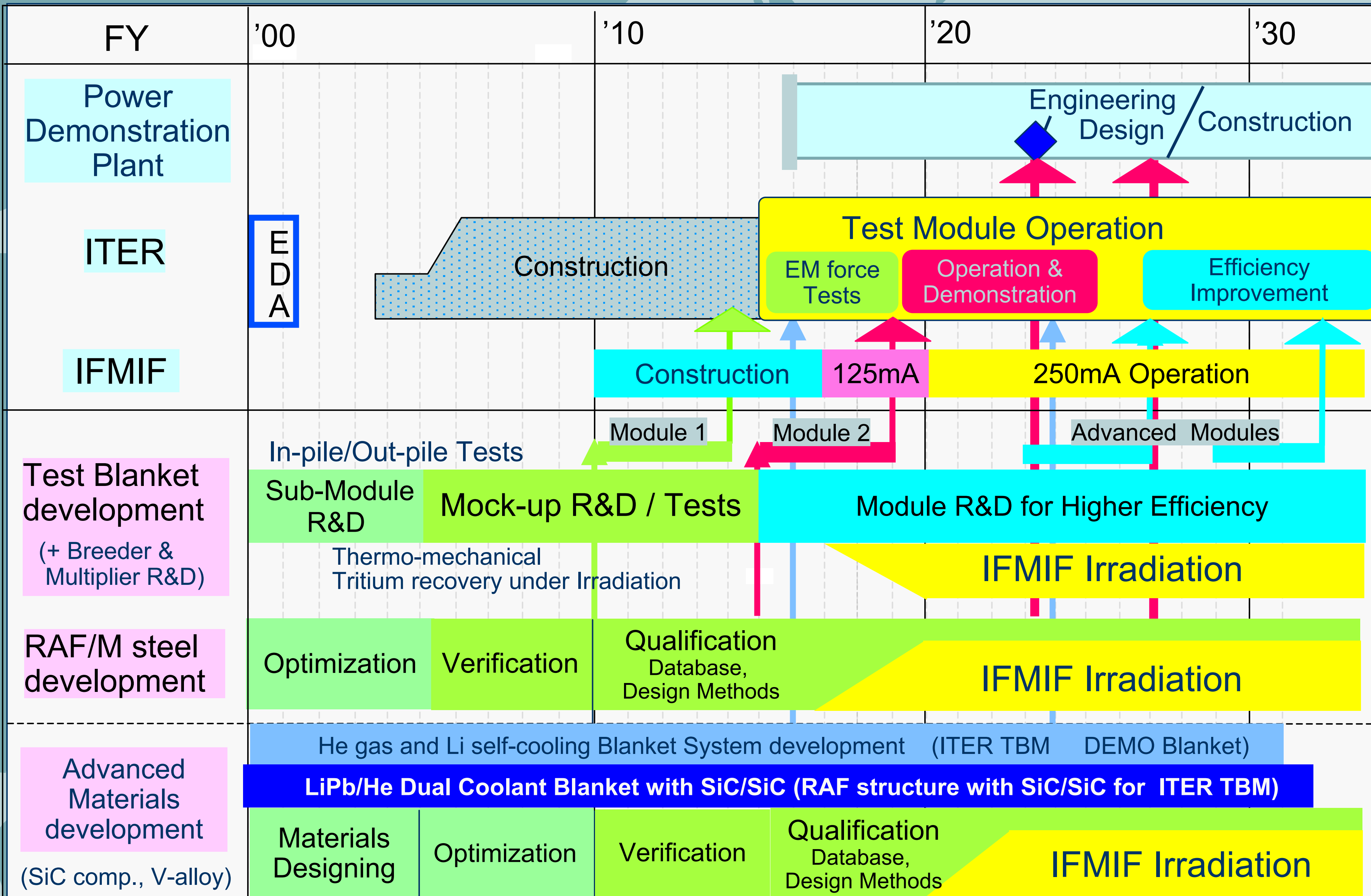
Advanced Blanket Concepts of Concern:

1. Liquid Lithium(or FLiBe)/Vanadium-Alloy
2. He cooled Solid Breeder/SiC/SiC

A: LiPb/He Dual Coolant with SiC/SiC (+RAF)
B: LiPb as Breeding Material and Coolant

1 and 2 are being conducted in Universities and NIFS
as the activities of Fusion Network and JUPITER-II
A and B are new activities in Universities
R & D strategy is under extensive discussion
including the planning as a
candidate for the post JUPITER-II (2007-)

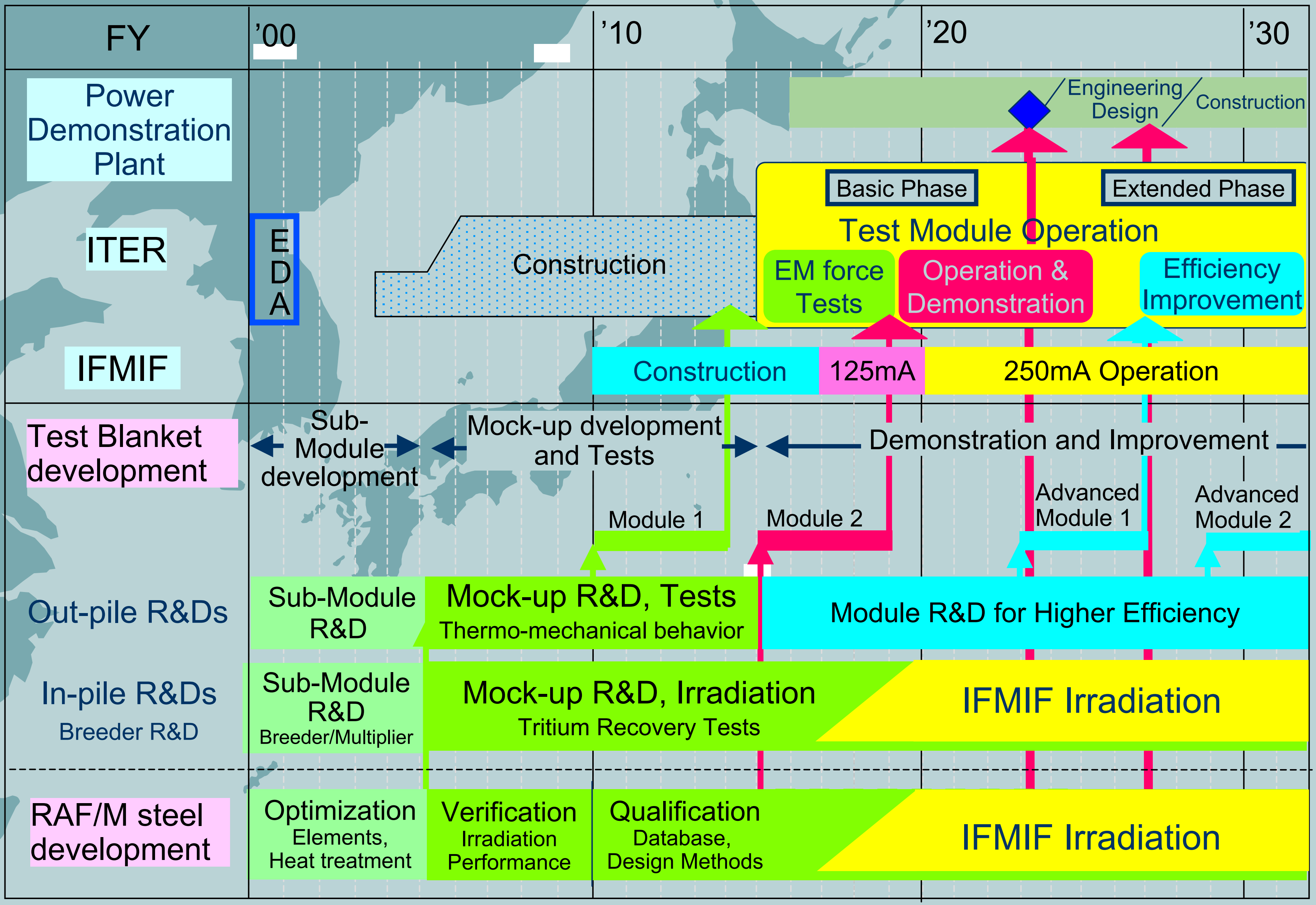
Blankets and Structural Materials Development



Research Emphasis and Status (1)

Reduced Activation Ferritic/Martensitic Steels

Blanket and RAF/M steel Development



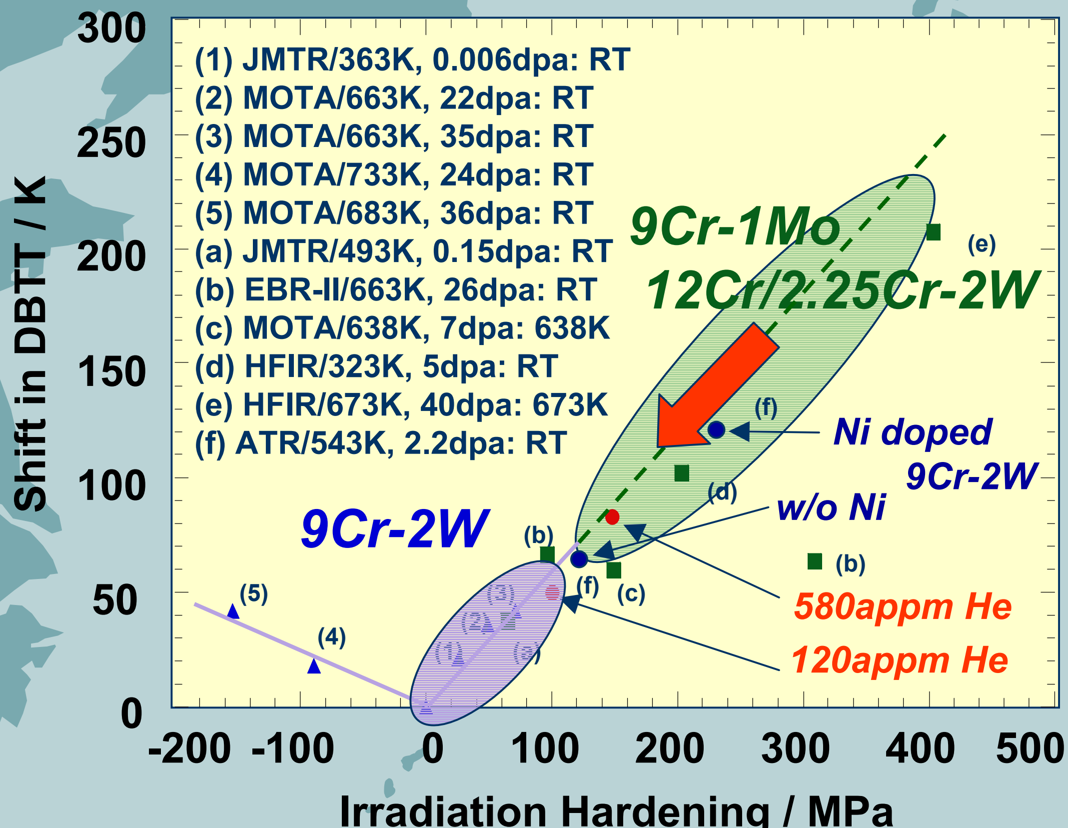
8/26/04SJ

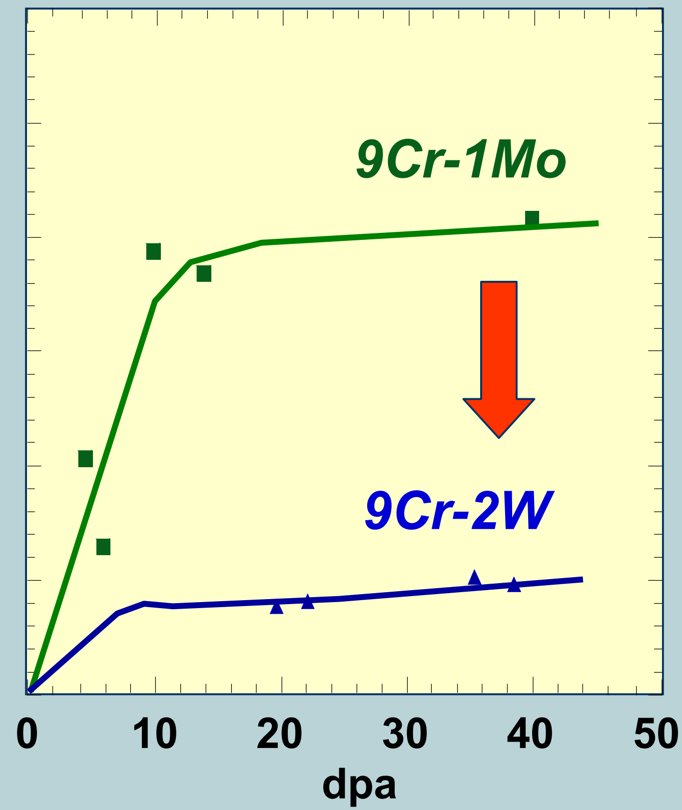
Research Status and Emphasis - RAFM

- Large heats of F82H (8Cr-2W) and JLF-1 (9Cr-2W) were produced in Japan as reference heats and characterized by IEA test program
    - Data obtained on various physical and mechanical properties
    - Well characterized and recommended heat treatment and welding procedures have been established
    - Irradiation data are being obtained for matrix and weld joints
- The resulting data are compiled in relational database
    - With emphasis on the traceability of the results
- Large heat (9 ton) of JLF-1 was melted and thick plate (38mm, 25mm) processing is on-going
    - Production of 20 ton heat of JPCS (F82H/JLF-1) is under negotiation for the early production in FY2005

Optimization of Total Performance for Fusion

● Suppression of irradiation embrittlement



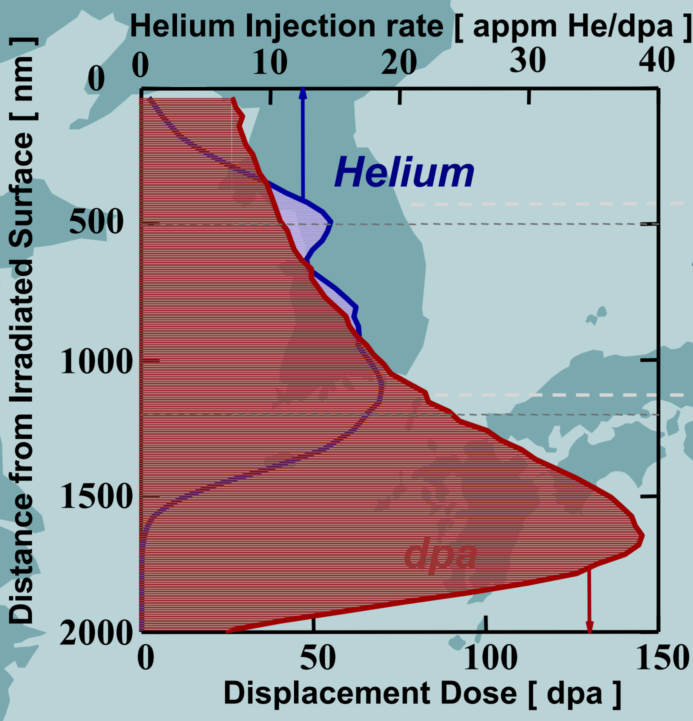


Successful substitution of steel elements
Optimization of Cr content (8–9%Cr) with 2W
Optimization of whole processing

Mechanistic Studies of Microstructure Evolution Under Fusion Environment

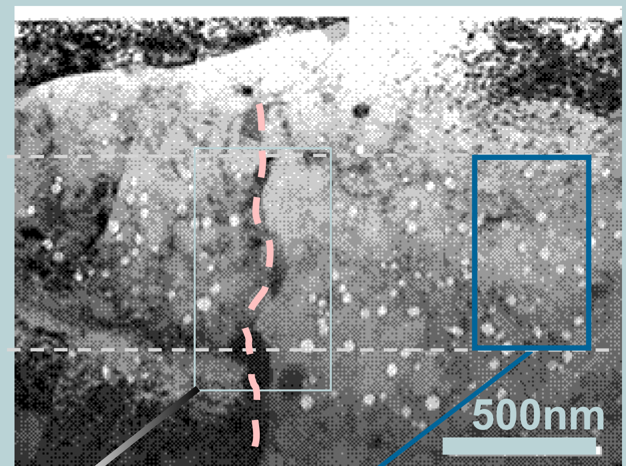
Simulation Radiation Damage Study

Dual-Ion Irradiation : at 743 K with 6.4 MeV Fe3+ + Energy-Degraded 1.0 MeV He+

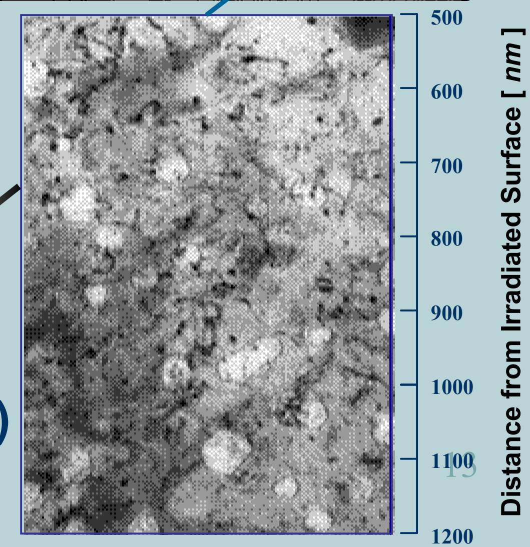


Depth Profile
6.4MeV Fe^{3+}, Degraded 1.0MeV He^{+}

Dual Ion-Beam Irradiated Regions



Void-depleted zones along lath boundaries



Cavity and dislocation image

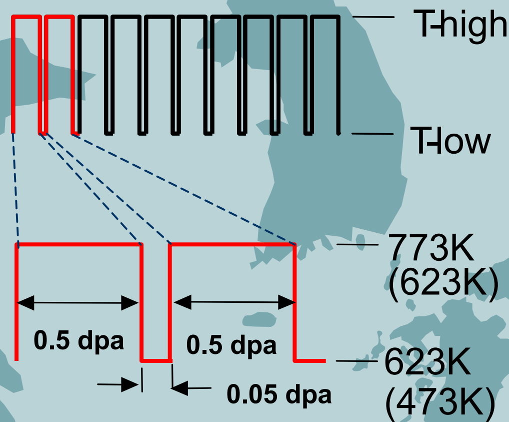
Swelling : 3.2 % (60 dpa @850 nm)

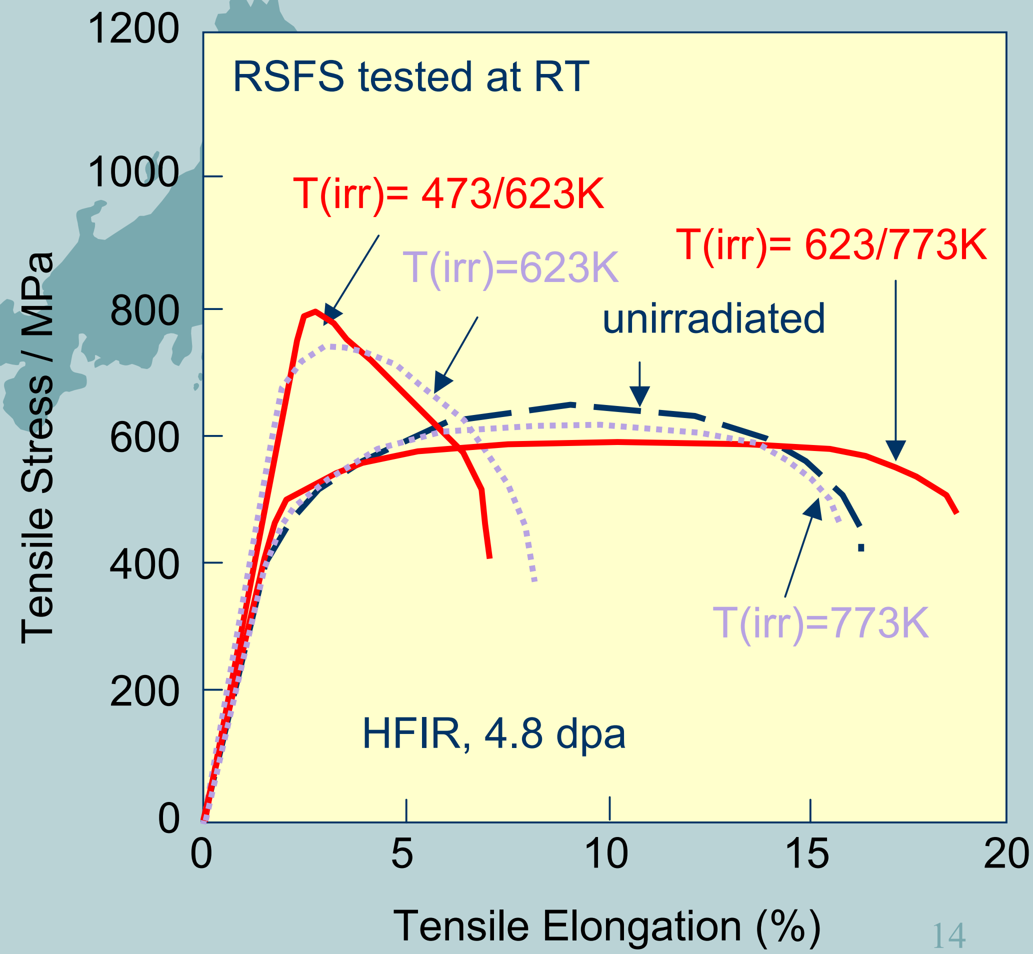
g = [110] Z ≃ [$\bar{1}$11]

Mechanistic Studies of Microstructure Evolution Under Fusion Environment

- **Insensitive to varying temperature irradiation**

Varying Temperature Irradiation
(HFIR: 0.5 x10 cycles)

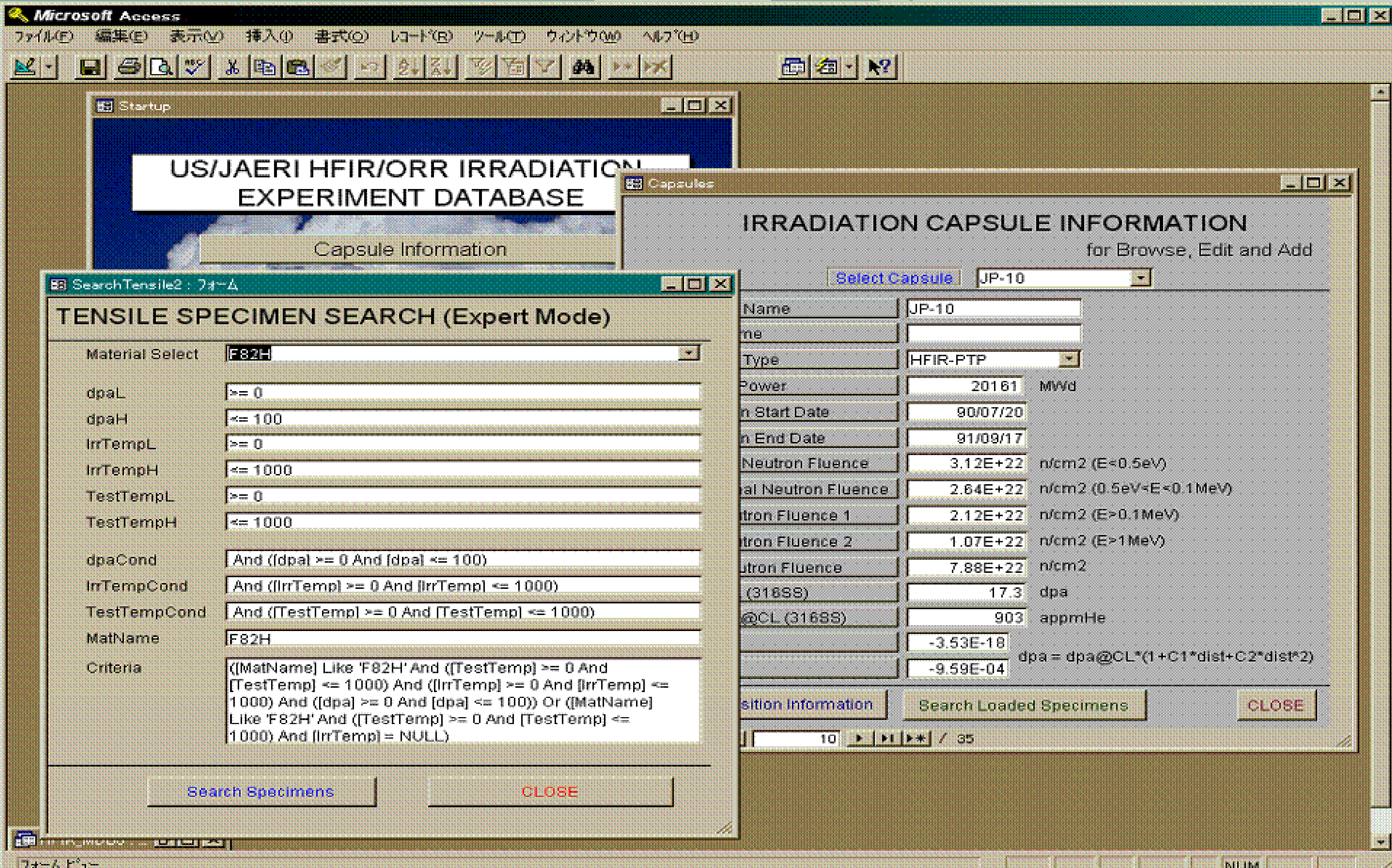




Tensile properties are similar between constant and varying temperature.

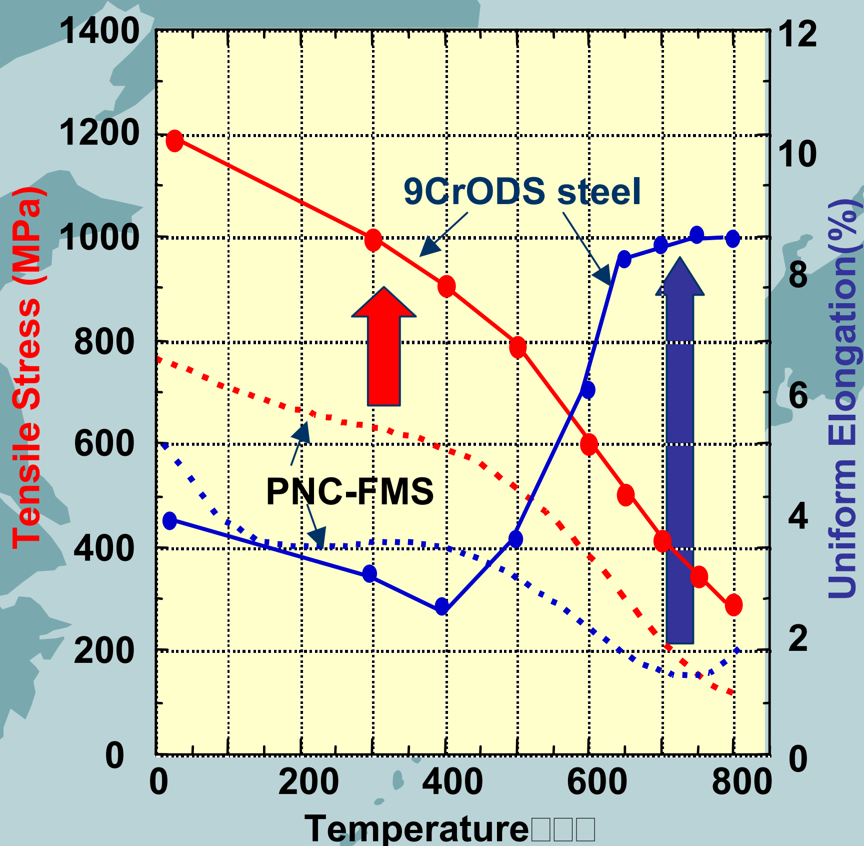
RAF Relational Database

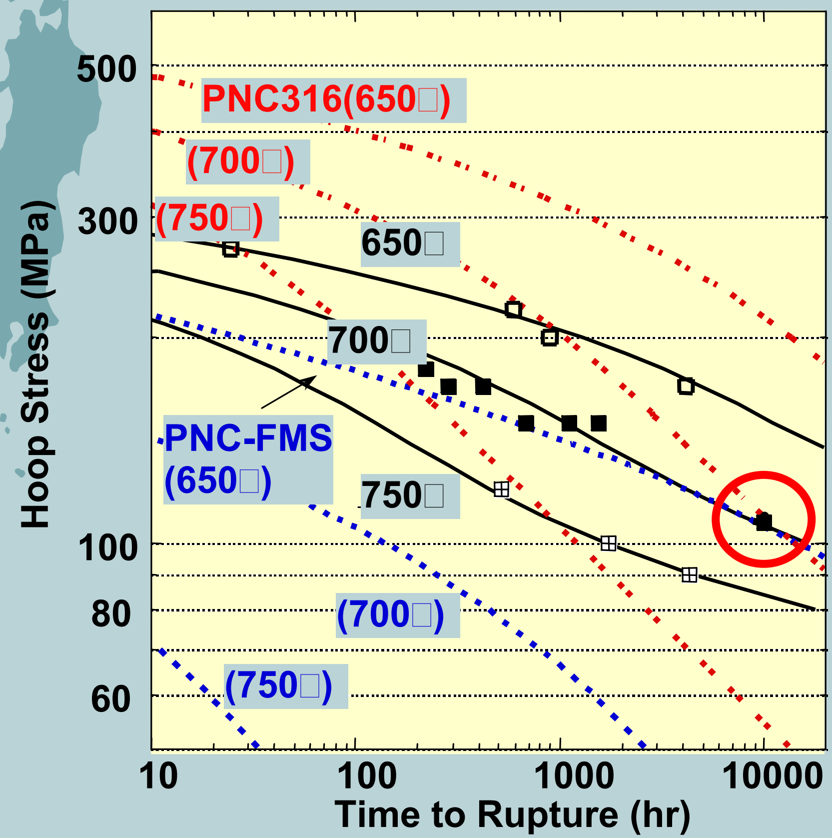
- On going under international collaboration -



ODS RAFS R & D for Fusion

- **Big improvement of high temp. properties**



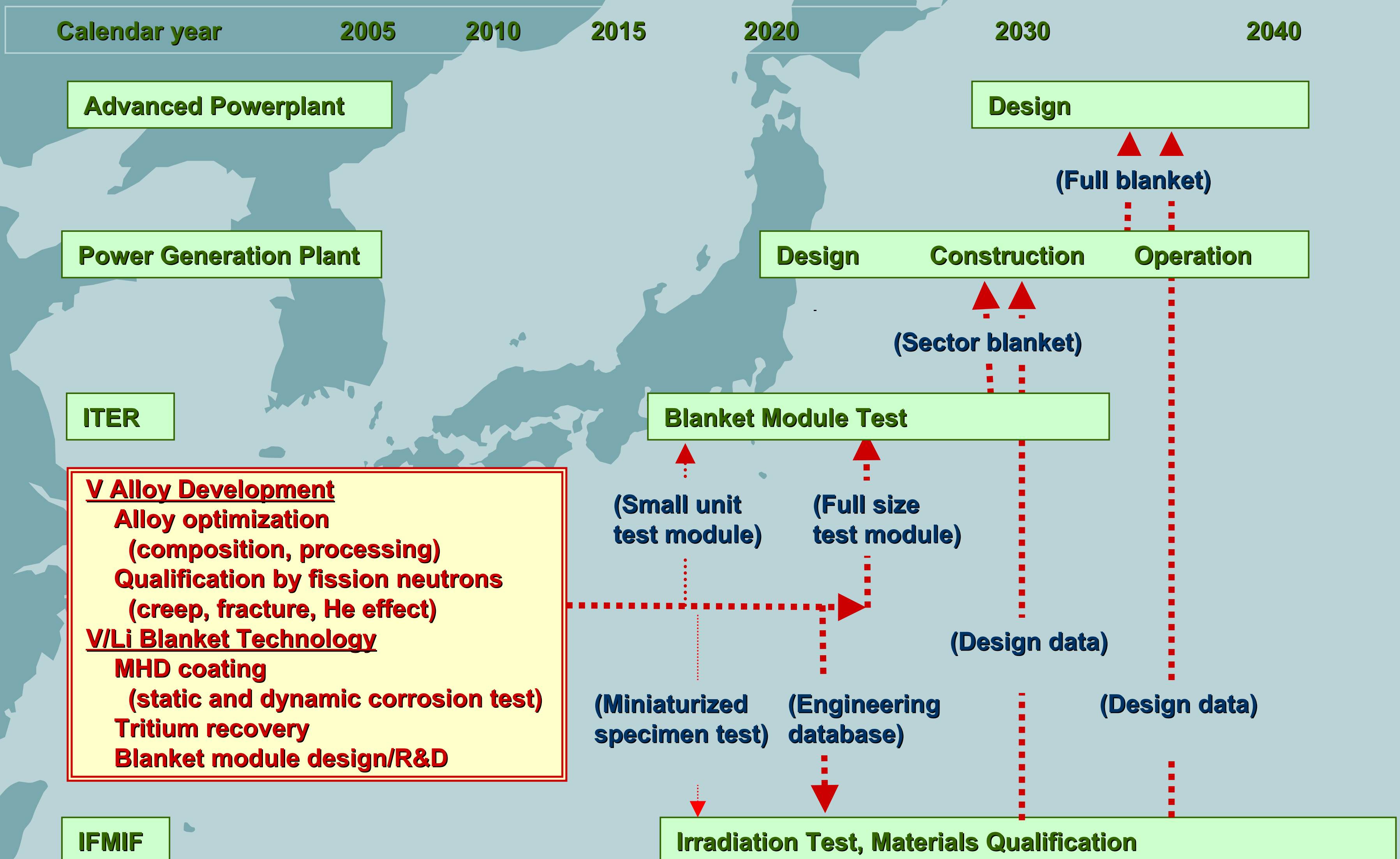


Time to rupture: 100 times longer than RAFS

Research Emphasis and Status (2)

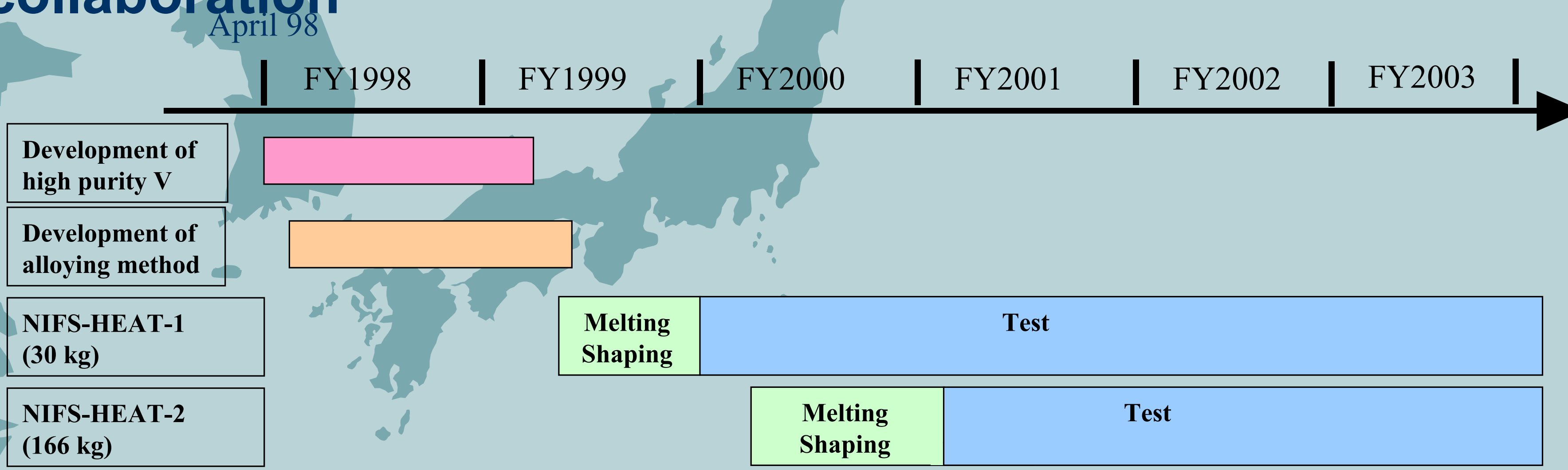
Vanadium and Its Alloys

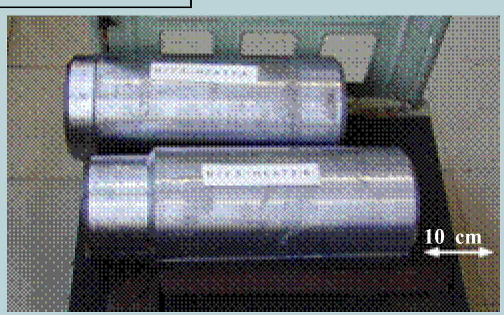
Roadmap for Vanadium Alloy Development



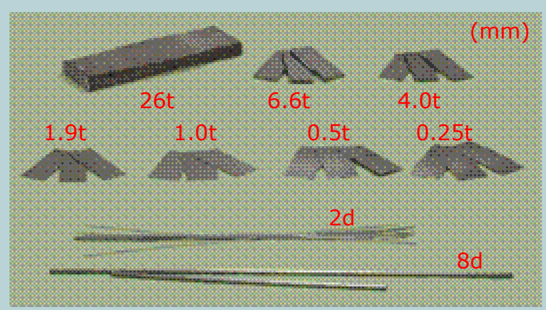
Production and Characterization of High Purity V-4Cr-4Ti (NIFS-HEATs)

- NIFS-HEAT-1 has been tested by Japanese Universities
- NIFS-HEAT-2 was distributed for international collaboration

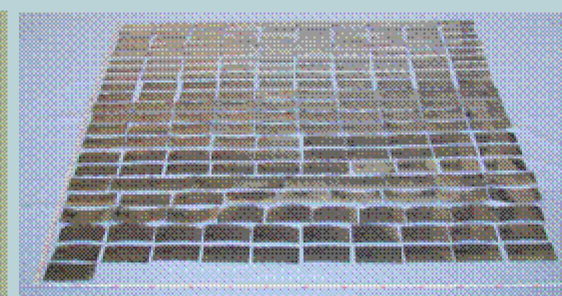




NIFS-HEAT-2 ingots

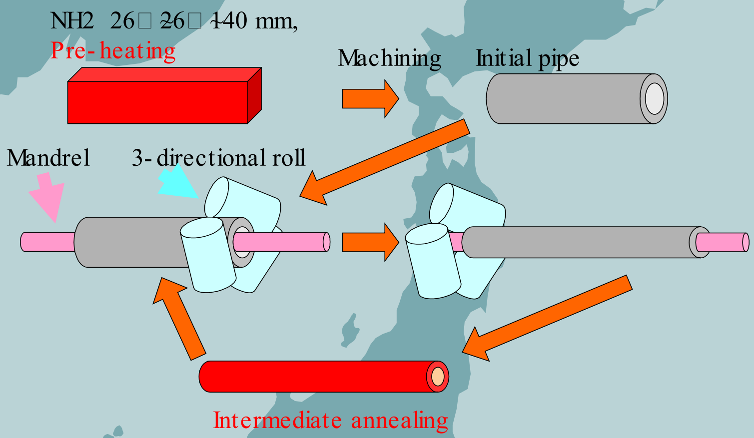


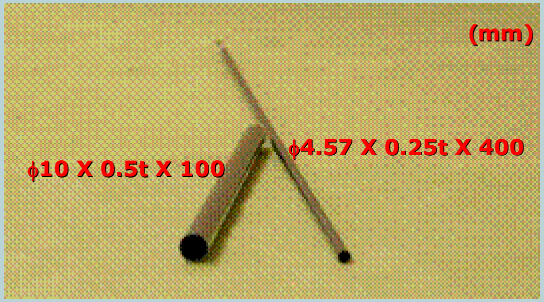
Products

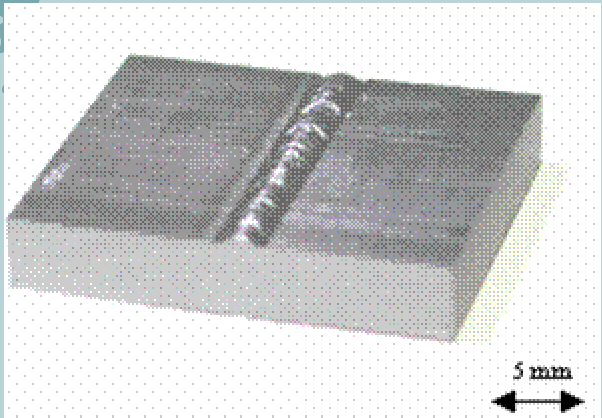

Specimens distributed

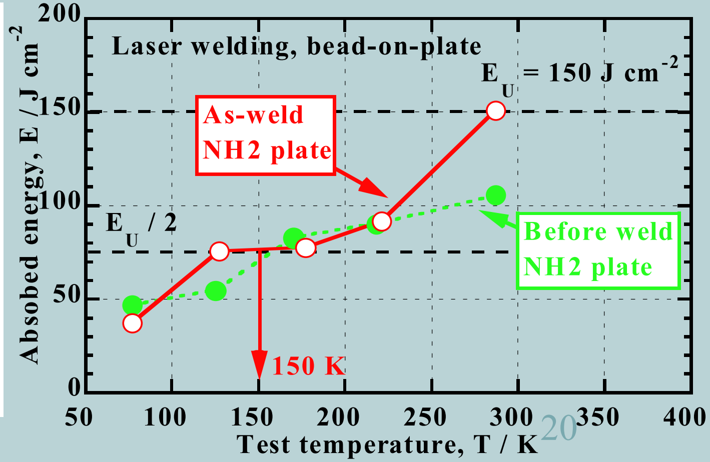
Fabrication Technology Tubing, welding

- Based on the control of impurities and precipitates, plates, sheets, wires and tubes were successfully fabricated
- Good properties of laser-weldment by atmospheric control and PWHT

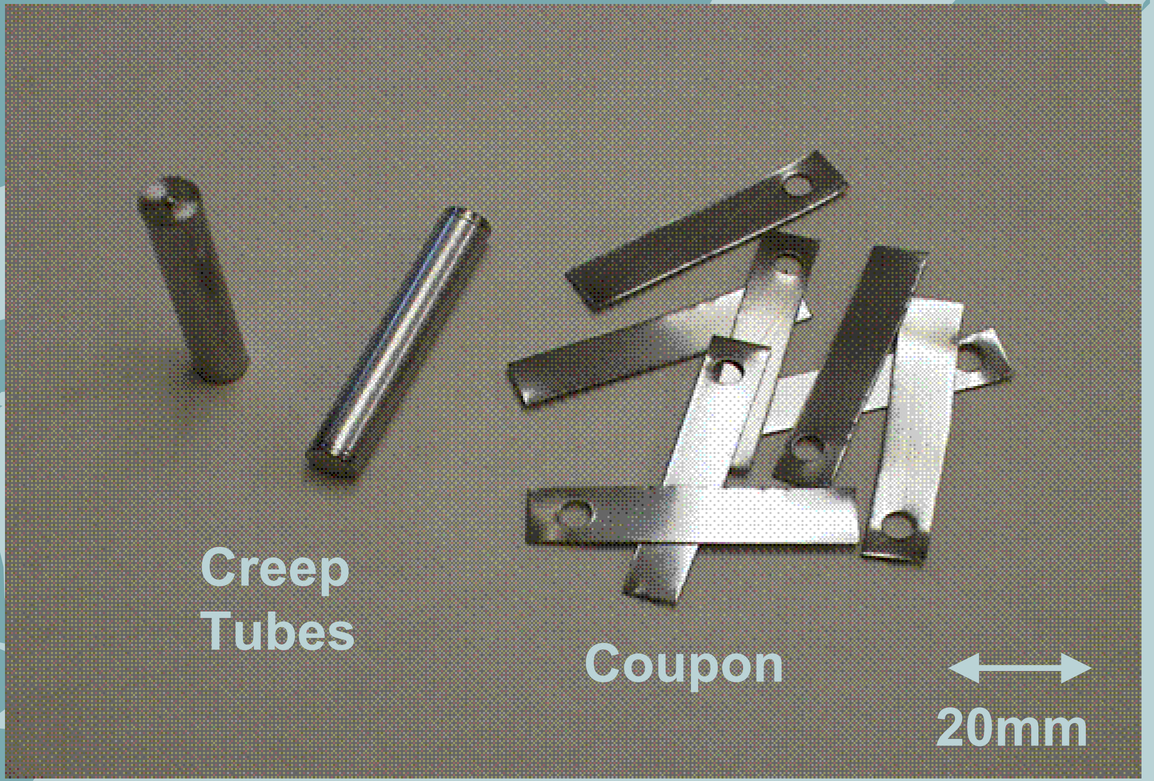


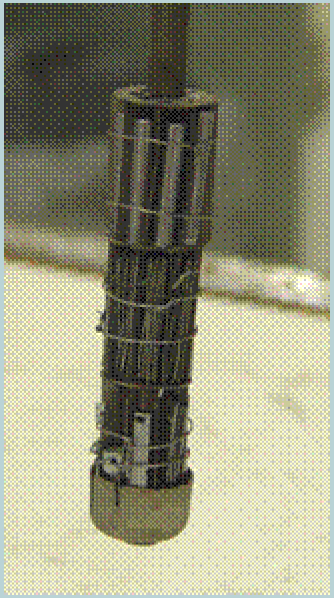






Thermal and Irradiation Creep Tests

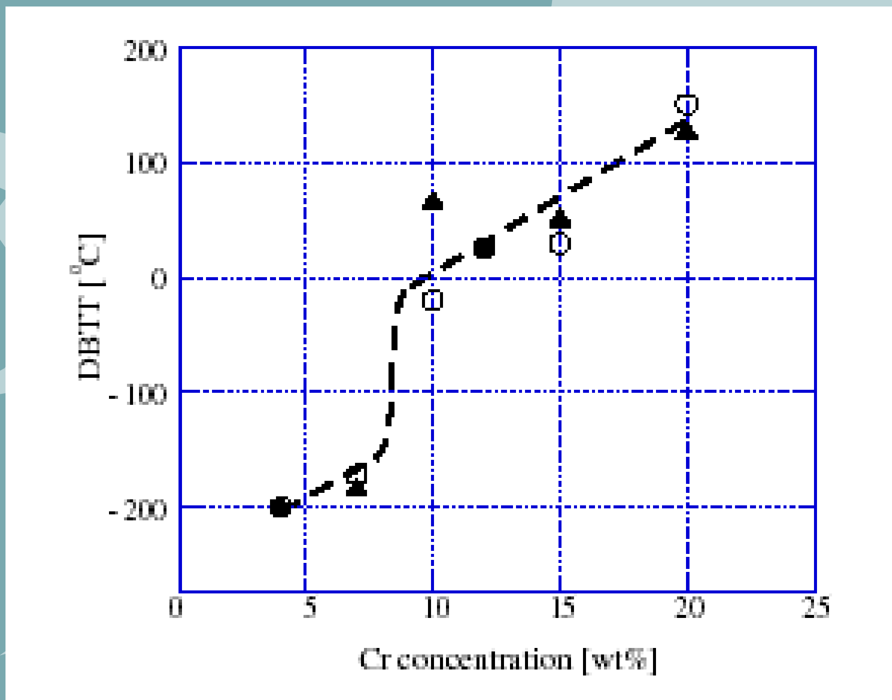




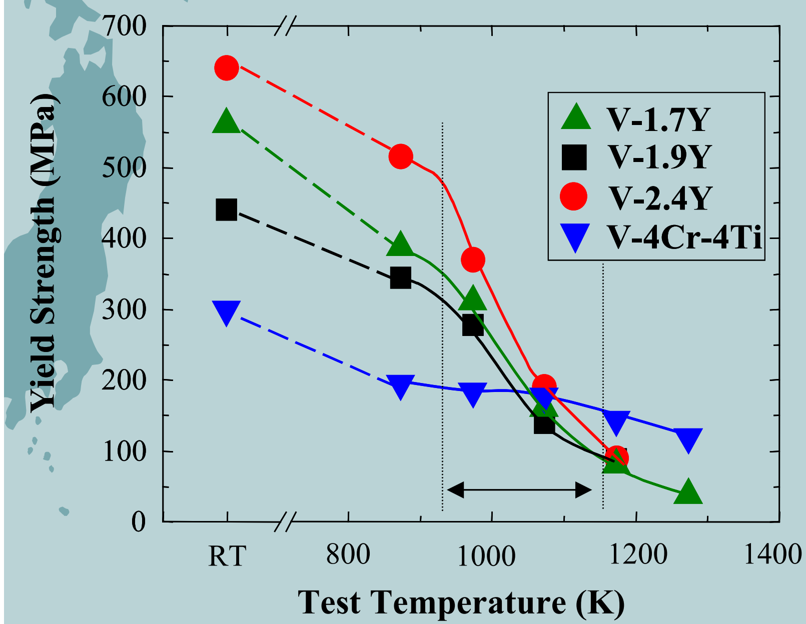
Specimen holder for thermal creep tests

- Thermal creep tests in Li is in progress
- Irradiation tests in Li is going on in HFIR

Improvement of Vanadium Alloys



V-xCr-4Ti alloy



Mechanically Alloyed V-xY

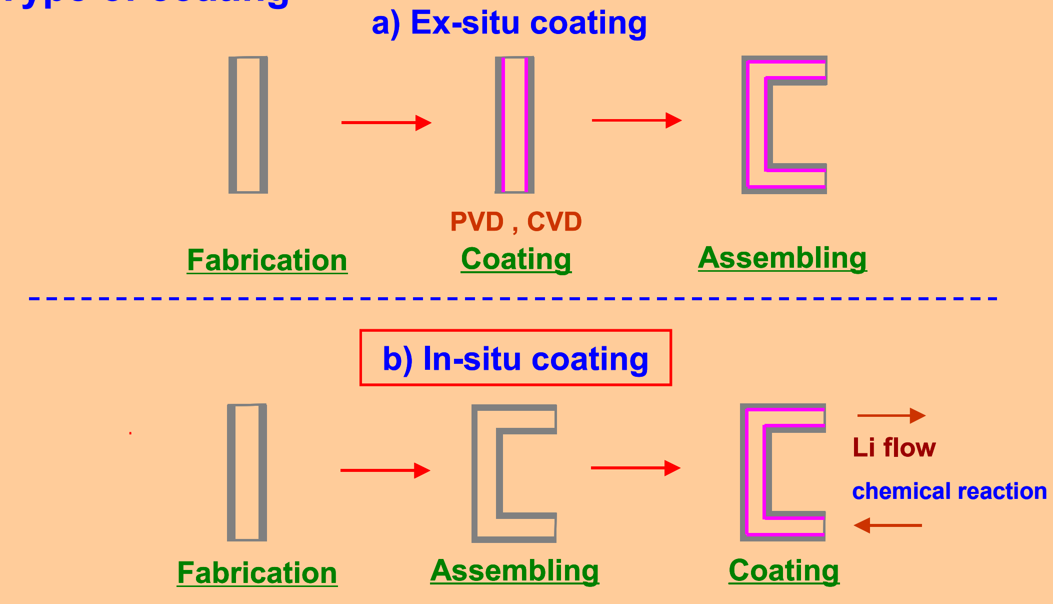
- Cr level may be increased to ~7 wt% without degrading fracture properties

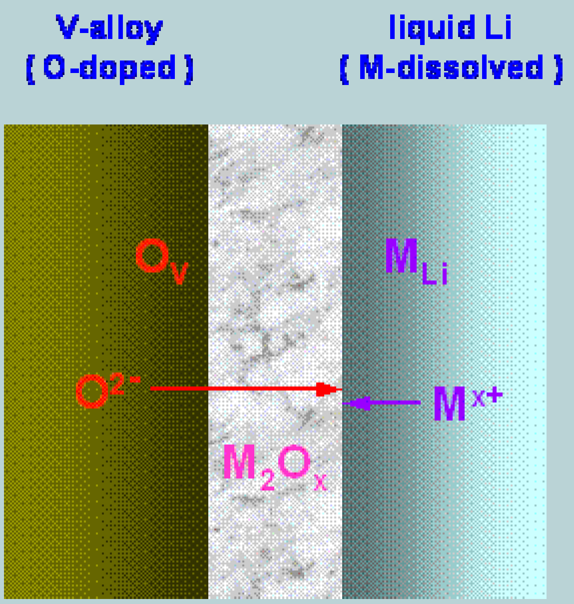
- Higher strength at <800C relative to V-4Cr-4Ti

In-situ Er_2O_3 Coating on V alloy

- Er_2O_3 was identified as promising candidate MHD insulator coating material (JUPITER-II)
- In addition to PVD coating, in-situ method has been developed.
- Advantages of in-situ coating
  - Coating on complex component
  - Healing without disassembling coating

Type of coating



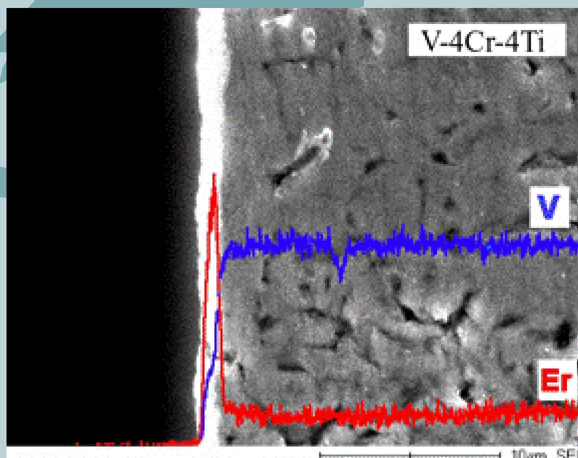


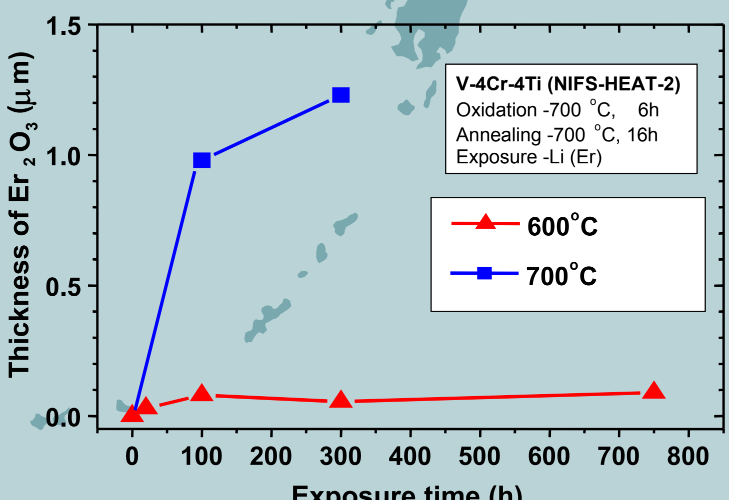
Generic model of in-situ method

In-situ Formation of Er_2O_3 Layer

Er_2O_3 coating formed successfully on V-4Cr-4Ti

SEM and EDS of cross section





oxidized	700°C, 6h
annealed	700°C, 16h
exposed in Li (Er)	600°C, 300h

XPS and ERD confirmed the Er_2O_3 phase

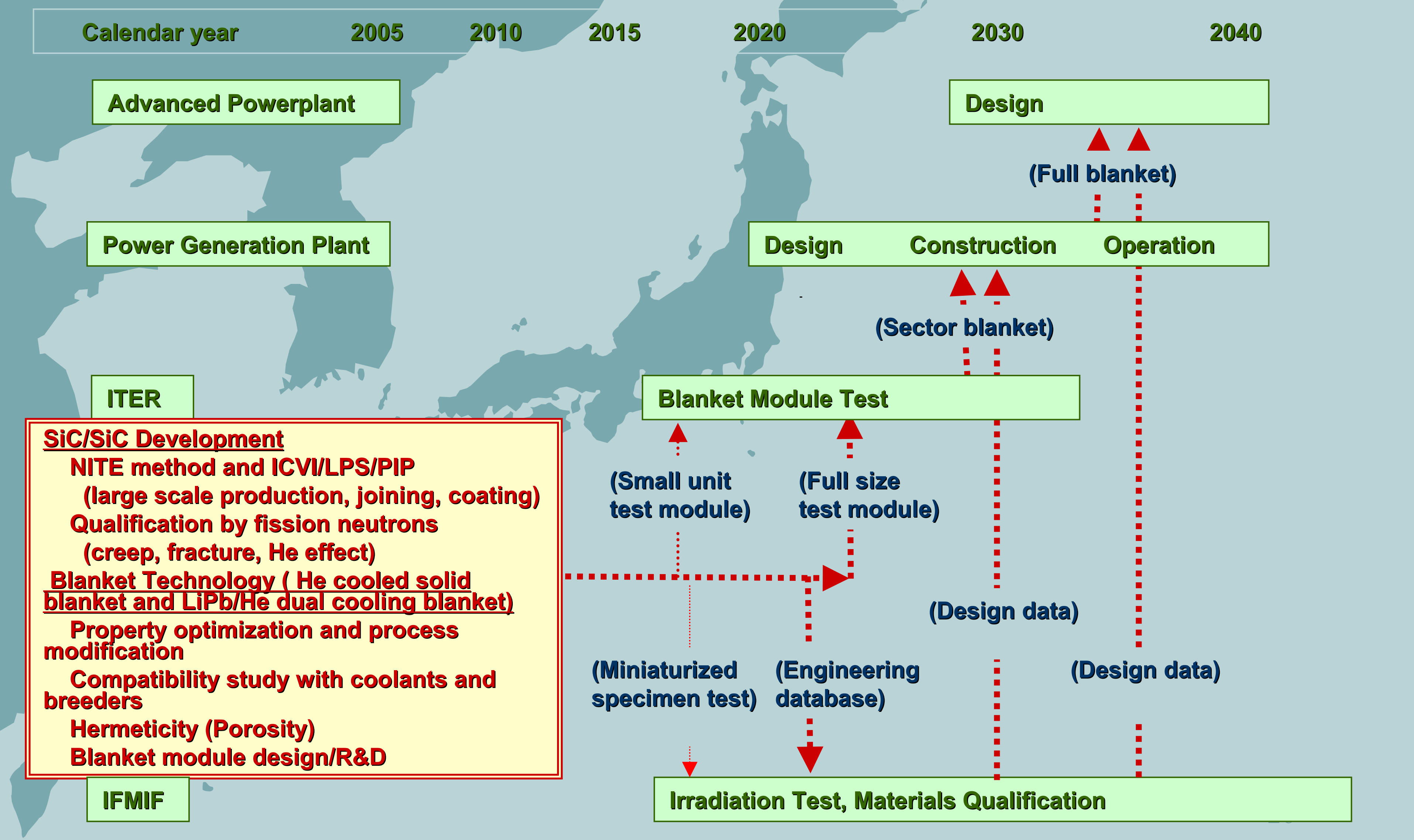
Er_2O_3 layer grow and saturate

Stable to 750h at 600C

Research Emphasis and Status (3)

SiC/SiC Composite Materials

Roadmap for SiC/SiC Composite Development



What Types of Blanket System with SiC/SiC

1: He Cooled Solid Blanket

a) The SiC/SiC has to be compatible with He, Solid breeder and Be at high temperature

b) SiC/SiC should keep **high thermal conductivity** and low tritium retention. “sealing layers” is an option

   a) the degradation of thermal conductivity under neutron irradiation is critical from heat flow point of view

c) The degradation of mechanical properties under high fluence neutron irradiation at elevated temperature is an issue.

2: He/Li-Pb Dual Cooled Liquid Blanket

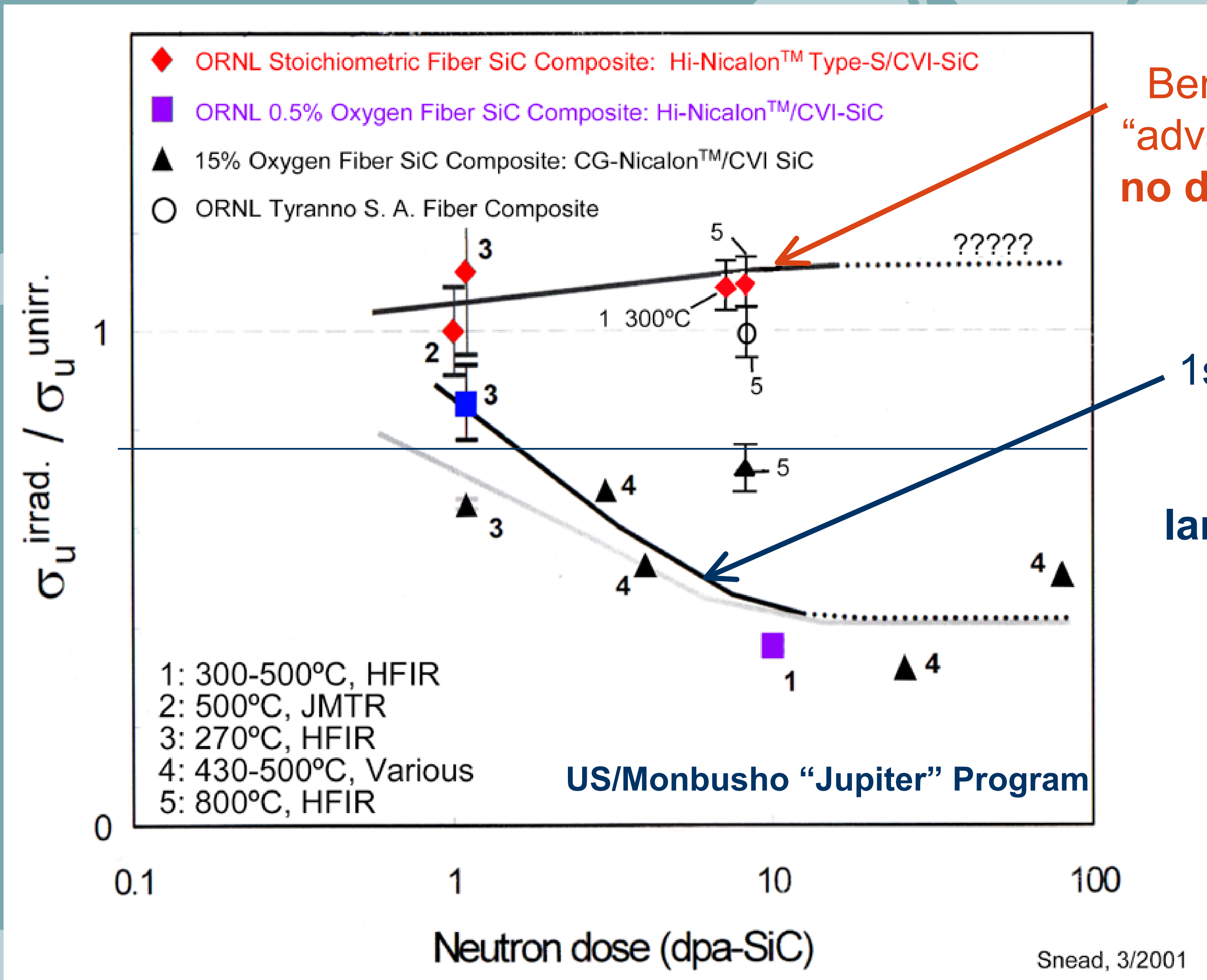
a) The SiC/SiC inserts have to be compatible with Pb-17Li and He

b) SiC/SiC should keep **low electrical conductivity** and low tritium retention. “sealing layers” is an option

   a) the change in conductivity is critical from pressure drop point of view and flow balance

c) The degradation of mechanical properties under high fluence neutron irradiation at elevated temperature is an issue

We Now Have First Radiation-Resistant SiC Composite



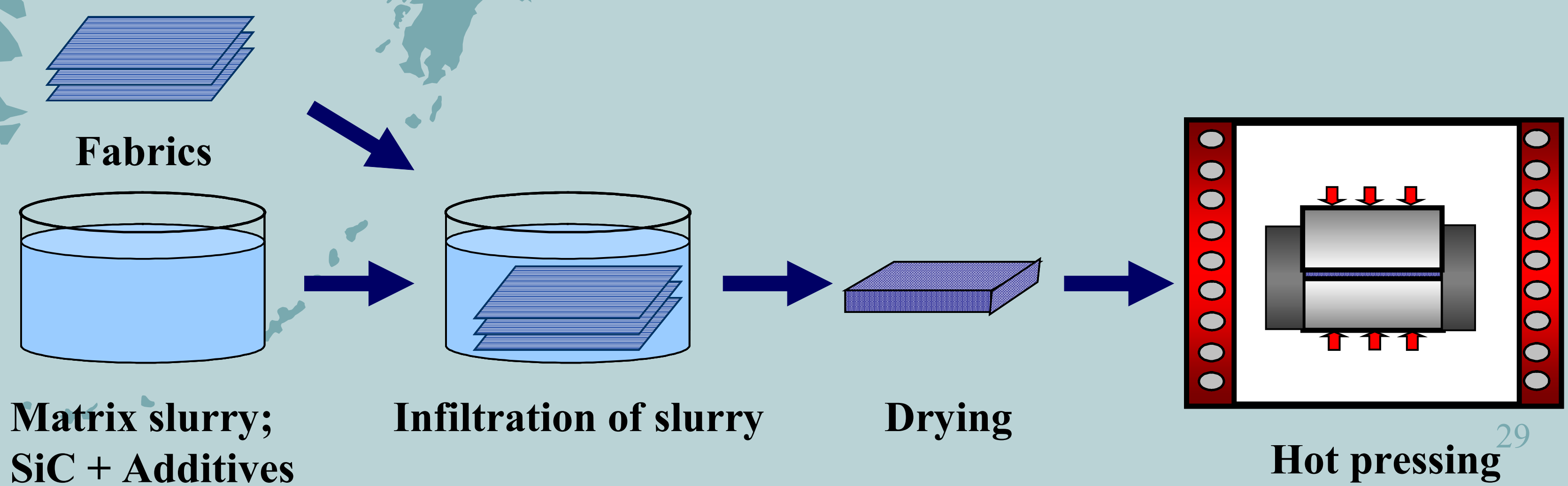
Bend strength of irradiated "advanced" composites show **no degradation** up to 10 dpa

1st- and 2nd generation irradiated SiC/SiC composites show **large strength loss** after doses >1 dpa

A Solution to the severe requirements for Nuclear Application is NITE Process

NITE: Nano-Infiltration Transient Eutectic Phase Process

- High Density and Excellent Hermeticity
- High thermal conductivity
- Chemical stability
- Flexibility in size and shape
    - Applicability of existing near net-shaping techniques
- Low production cost



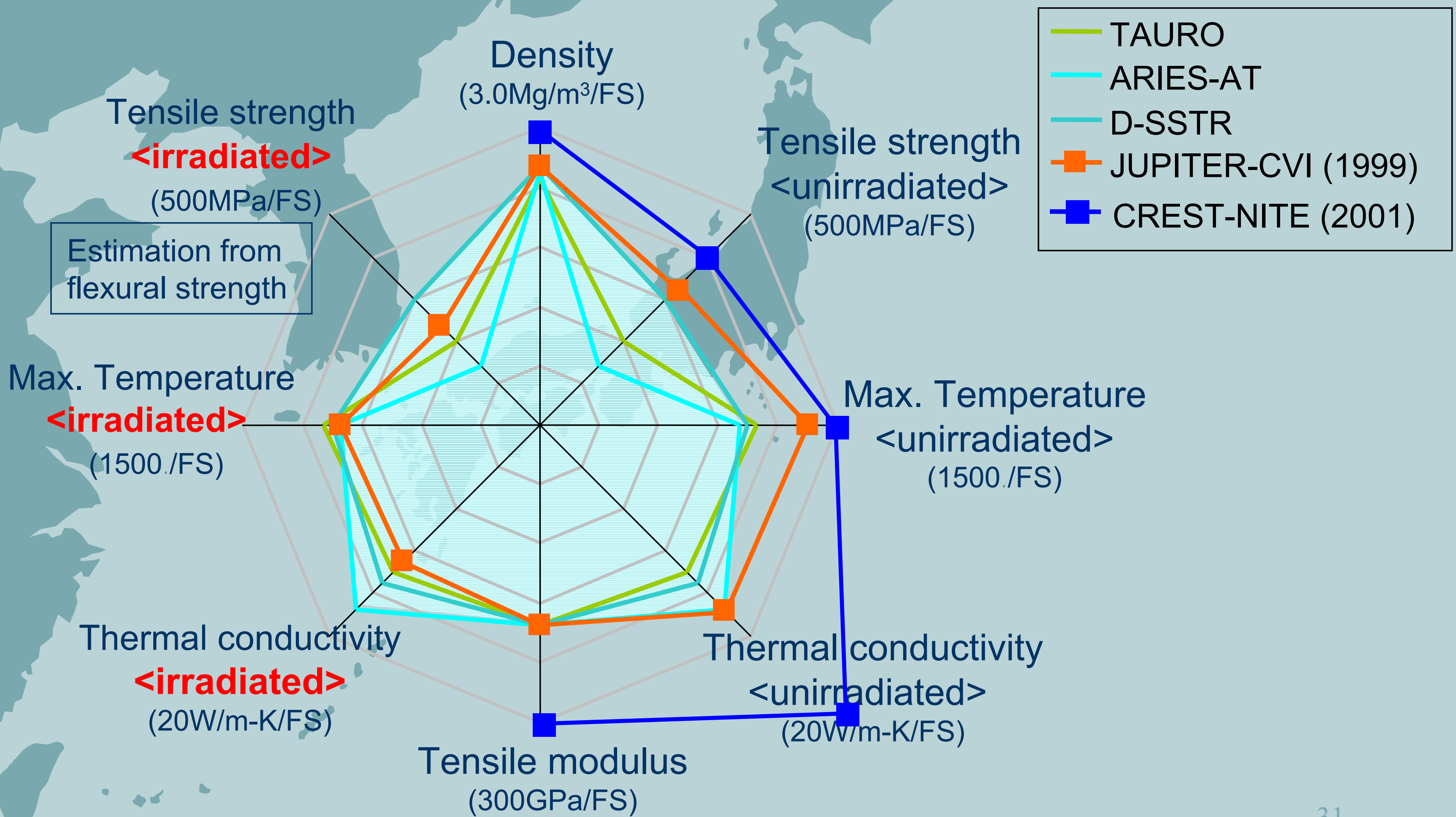
Why NITE Process is the solution ?

- The Combinations of :
  - •Near stoichiometry SiC fiber:
    Heat resistance in inert environment ~1800ºC
  - •Nano Powder:
    Large specific surface area
    Flexibility in chemistry adjustment and surface control
  - •Protective interphase (fiber coating):
    CVD-carbon, Phenolic resin-derived carbon, CVD-BN
  - •Nano powder-based matrix slurry:
    Infiltration to intra-fiber bundle openings
    Minimize binding agents to reduce detrimental impurities

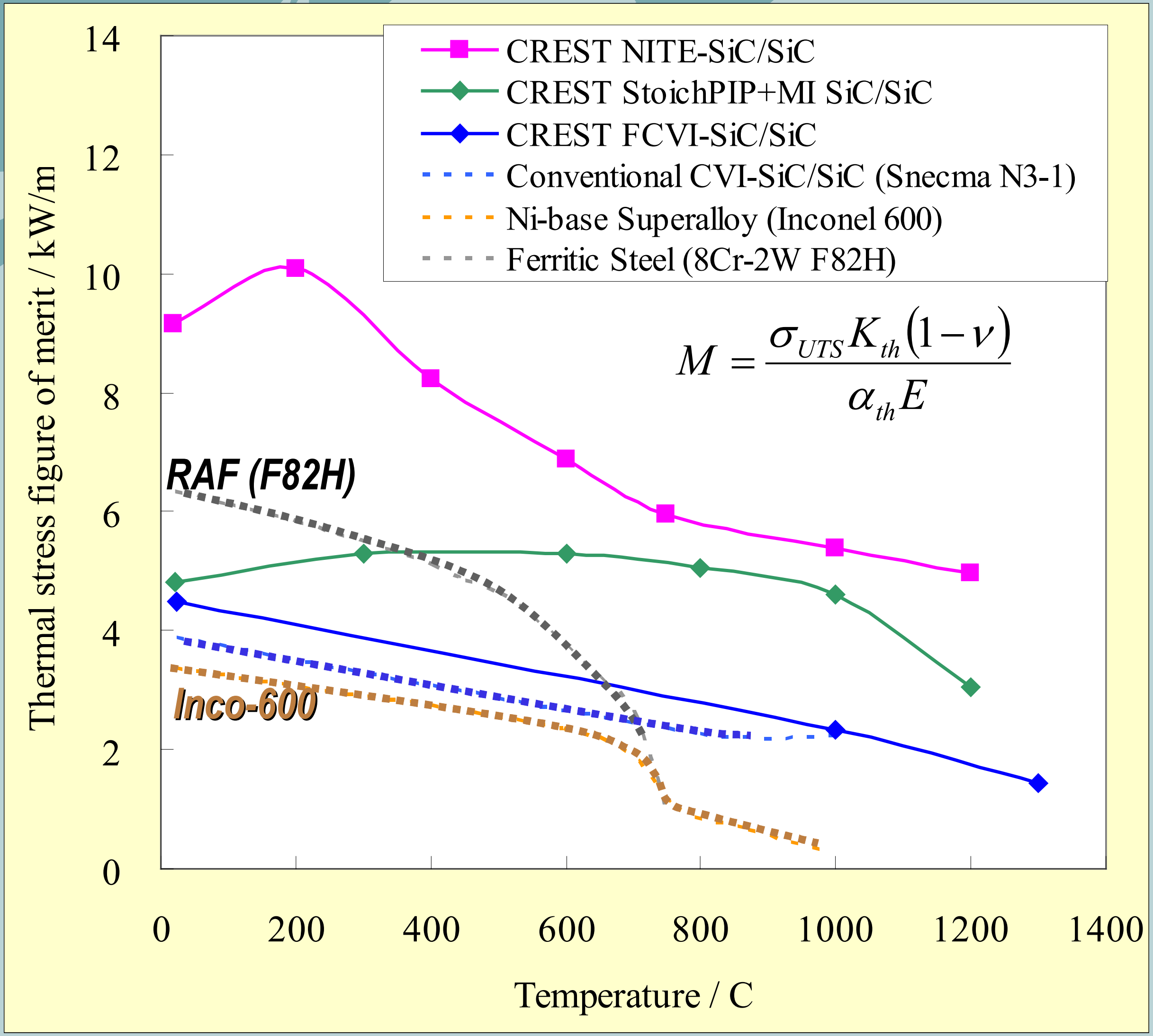
Make it possible to produce SiC/SiC

High purity and highly crystallized fiber and matrix with sound interface and optimized microstructure

Current Status of NITE Composites



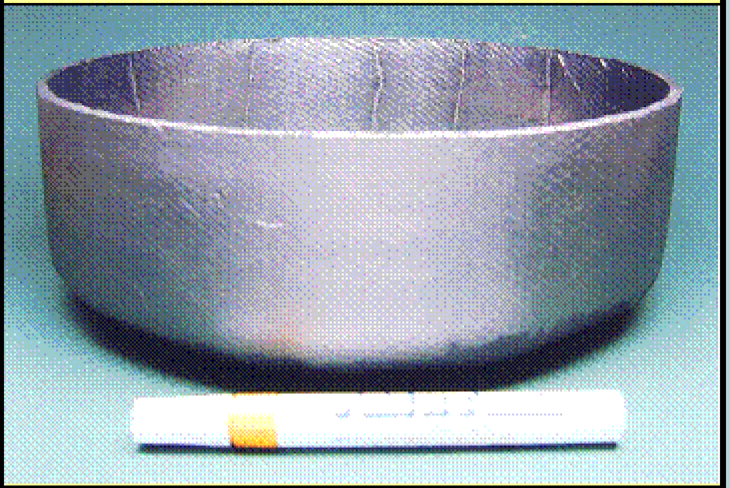
Thermal/Mechanical Properties of NITE Composites



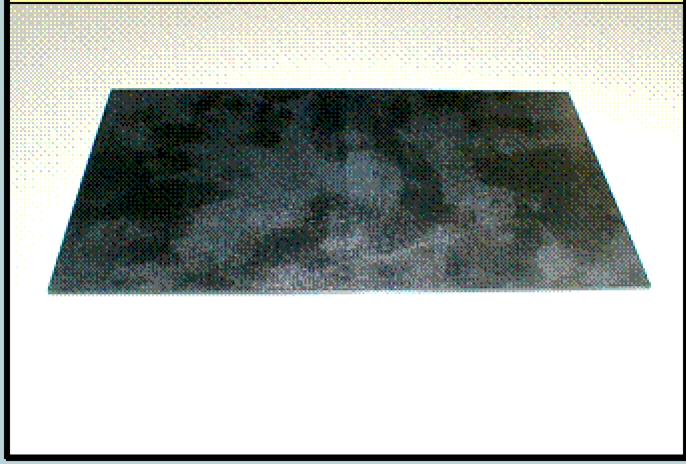
Shape Flexibility of NITE Composites

Thick Block: 97 x 97 x 70mm

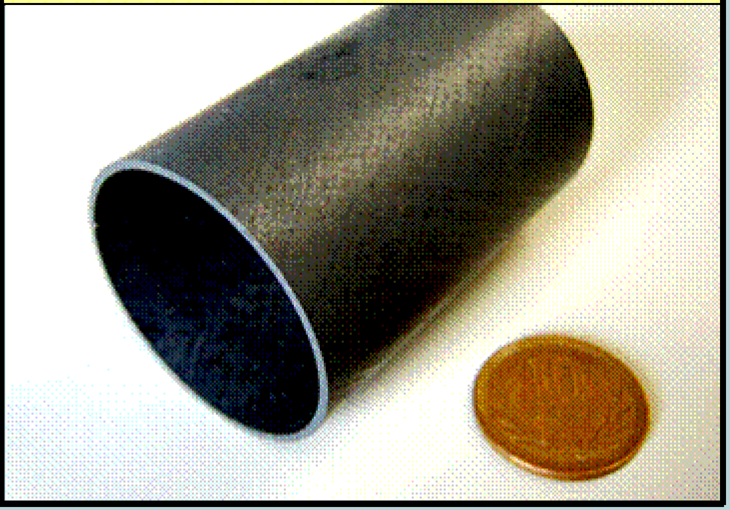
Combustor Liner



Thin Plate: 195 x 195 x 2.0mm

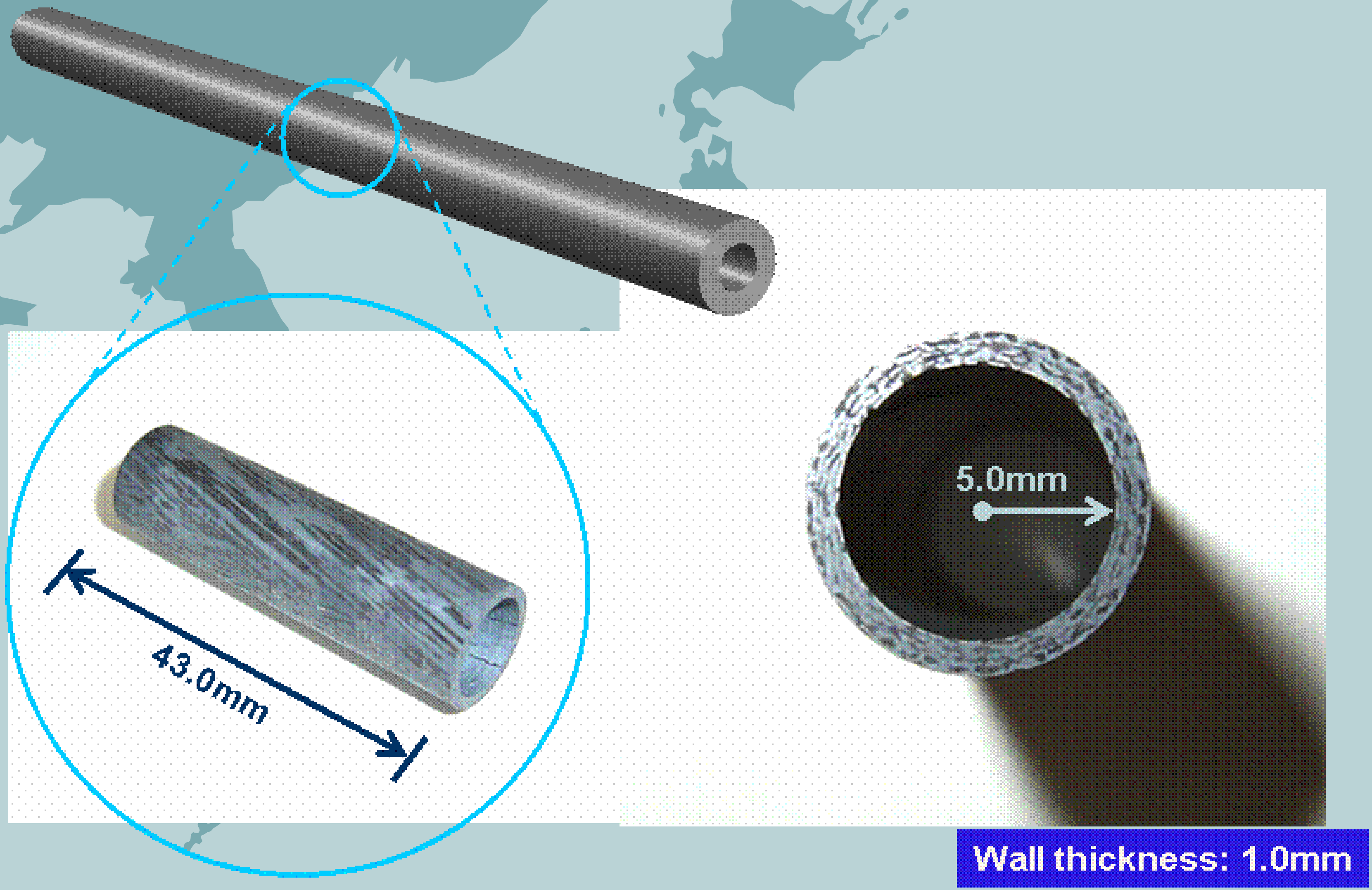


Tube: Φ31-33 x 60.0mm



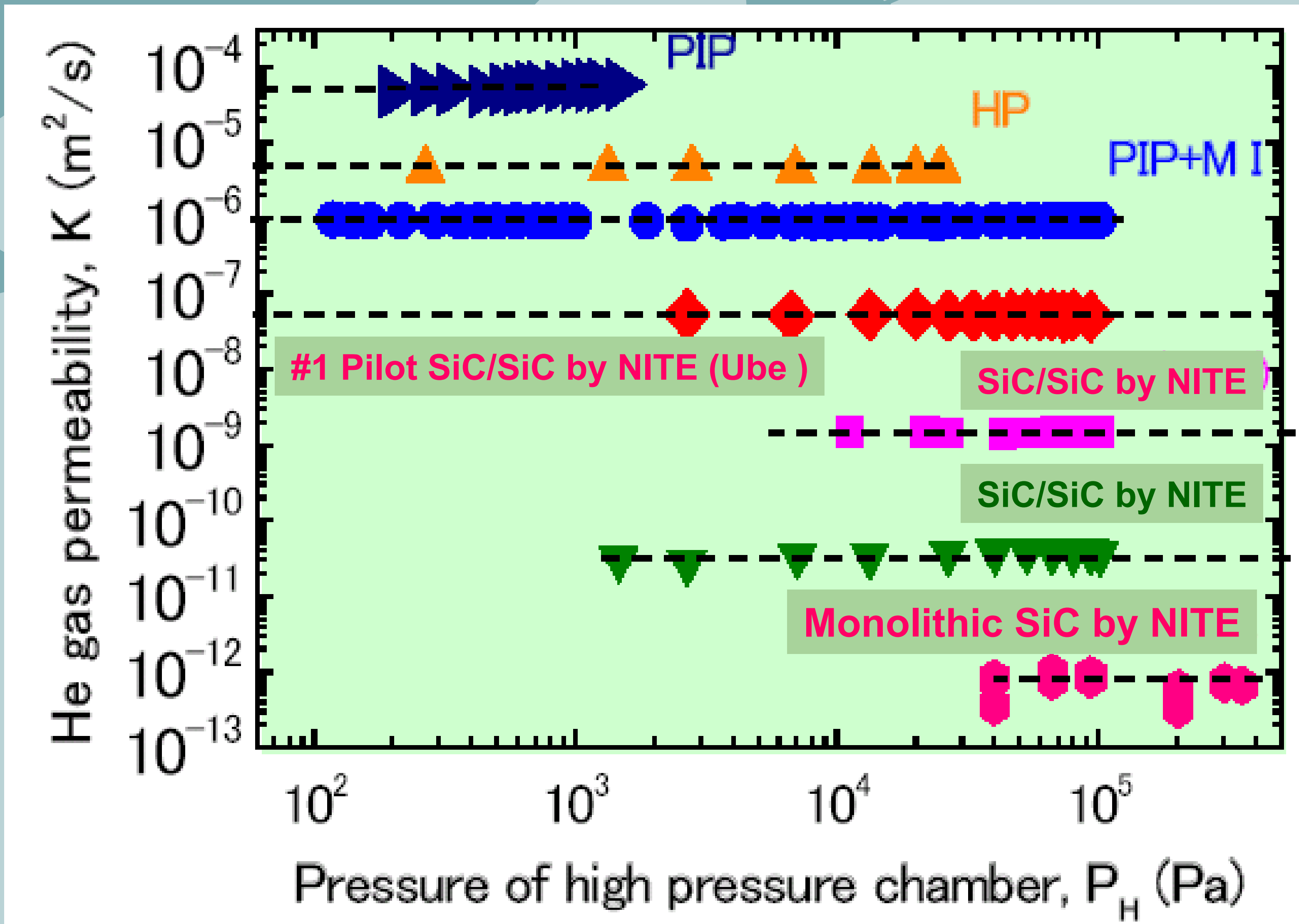
2D SiC/SiC Fuel Pin for GFR by NITE Process

- The first trial at Ube Co. Ltd., 9/12/03 -



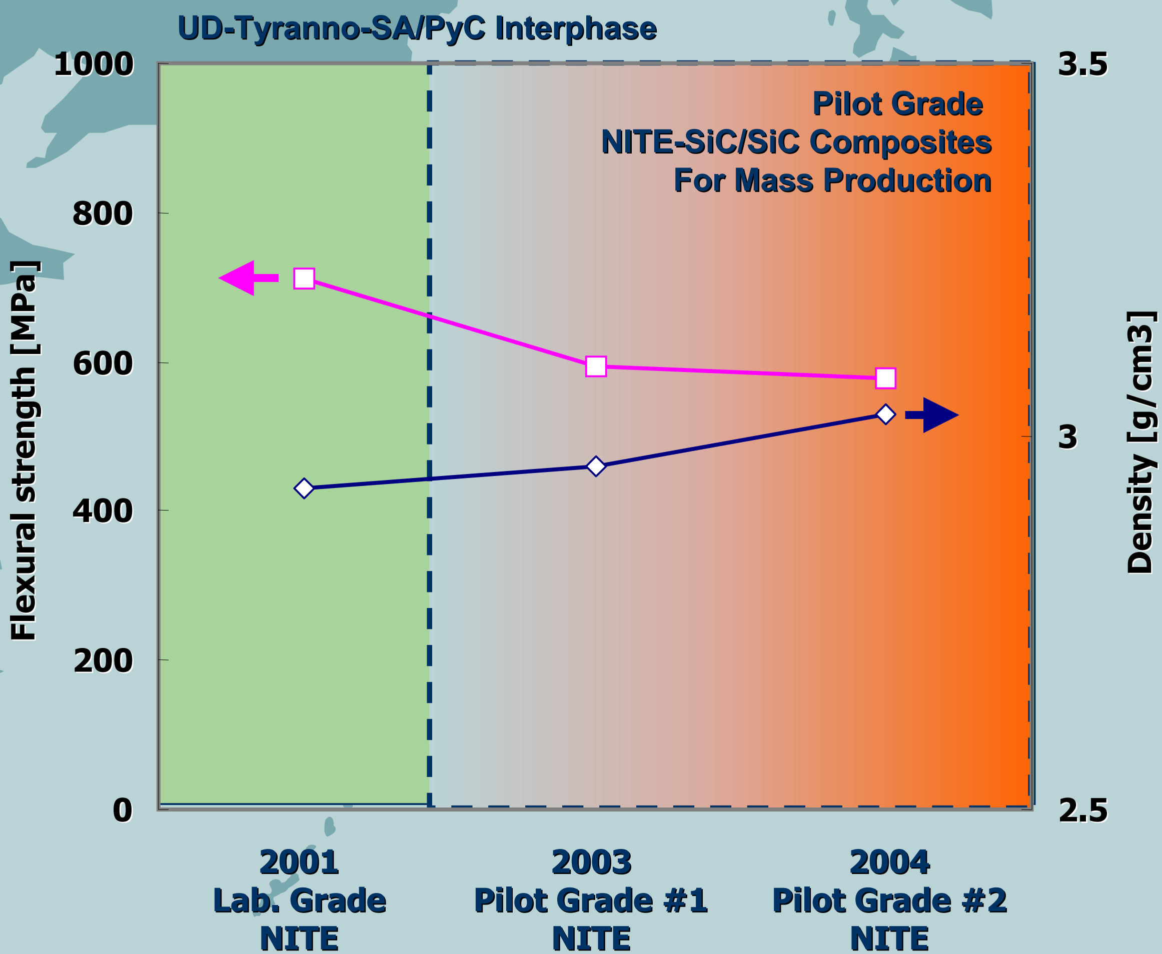
Goal: 3m(length) x 10mm(inner diameter) x 1mm(wall thickness)

Permeability of NITE Composites



Ref.): E. Hayashishita et al., to be presented at ICFRM-11

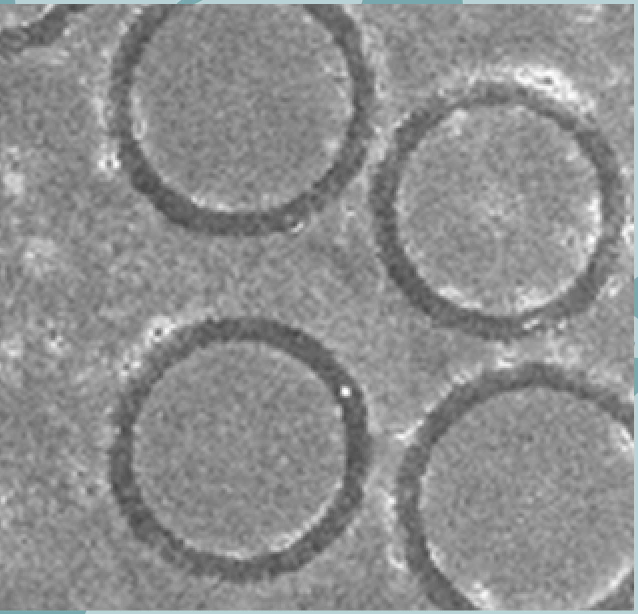
Progress in Large Scale Fabrication of NITE Composites at Ube Industries Co.



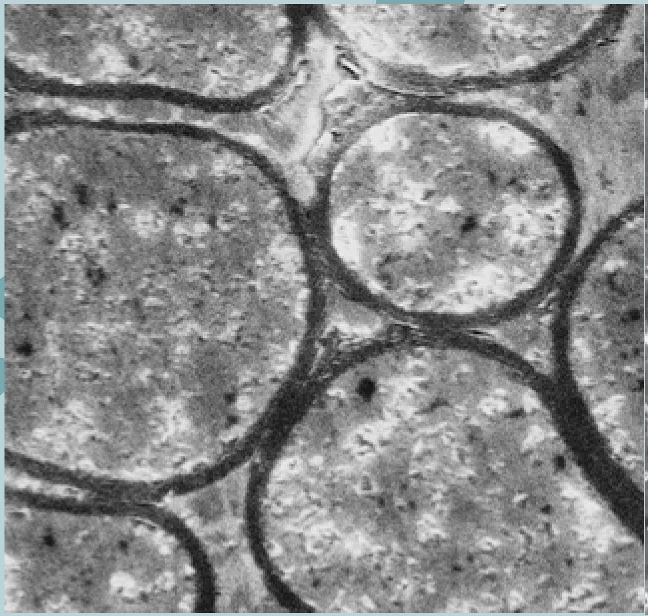
3 point Flexural Test
Support span : 18mm
Crosshead speed : 0.5mm/min
Specimen dimension :
4W x 25L x 2t mm

Progress in Large Scale Fabrication of NITE Composites at Ube Industries Co.

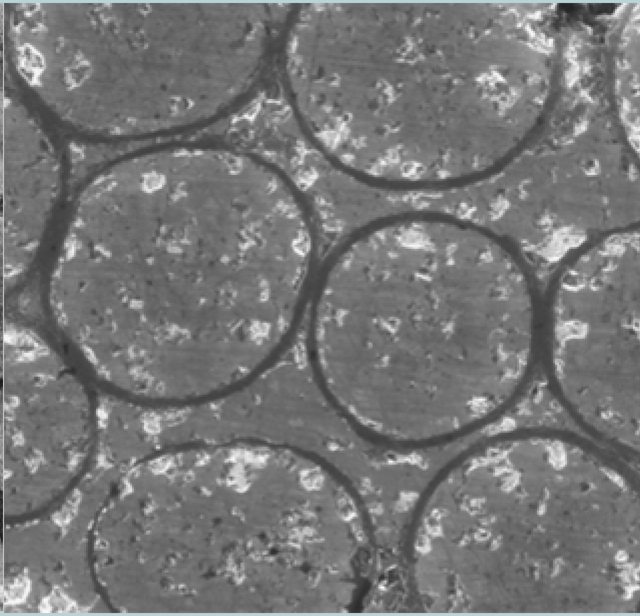
2001 Lab. Grade
NITE-SiC/SiC Composites



2003 Pilot Grade #1
NITE-SiC/SiC Composites

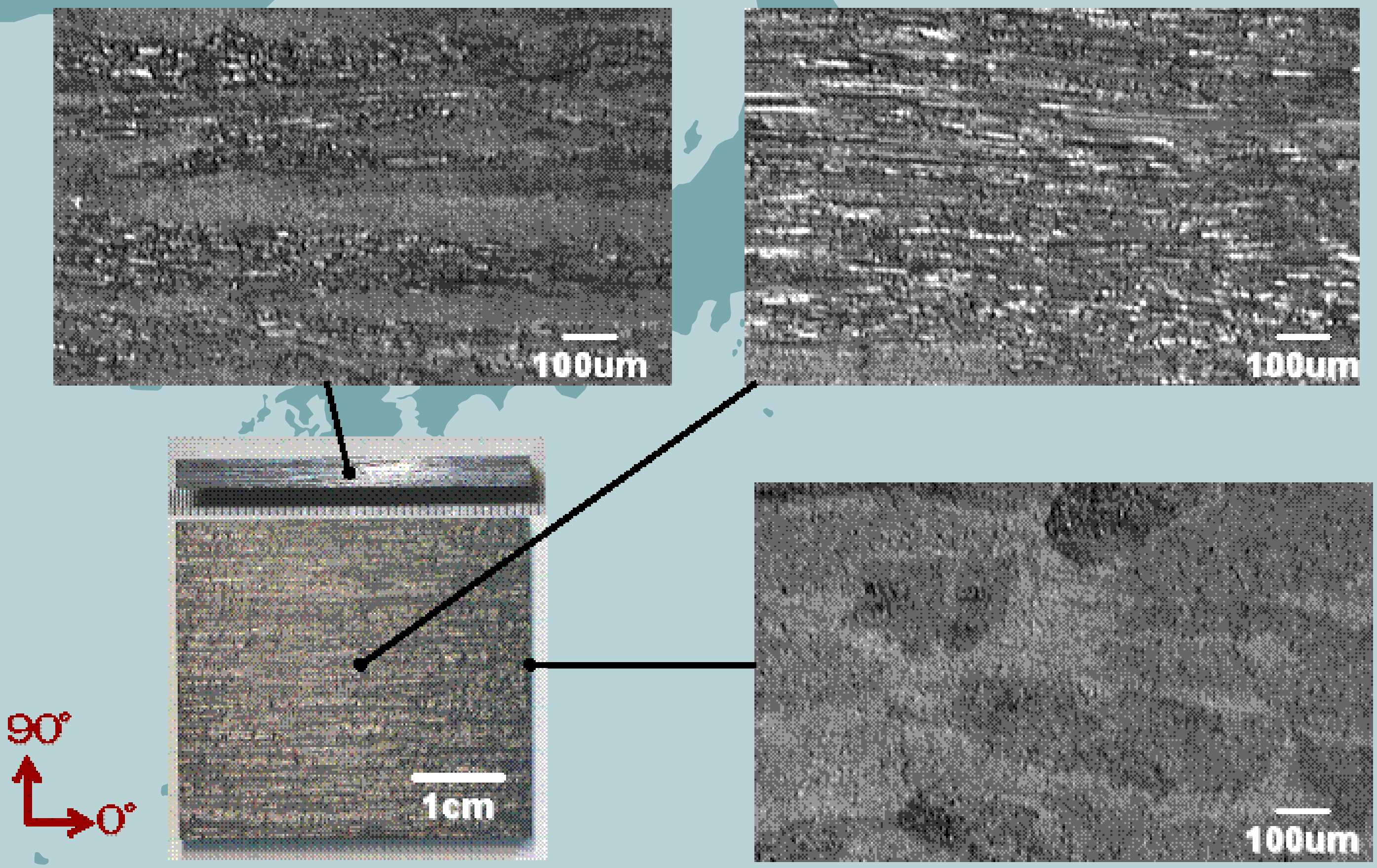


2004 Pilot Grade #2
NITE-SiC/SiC Composites



New NITE-SiC/SiC: Pilot Grade #2-1

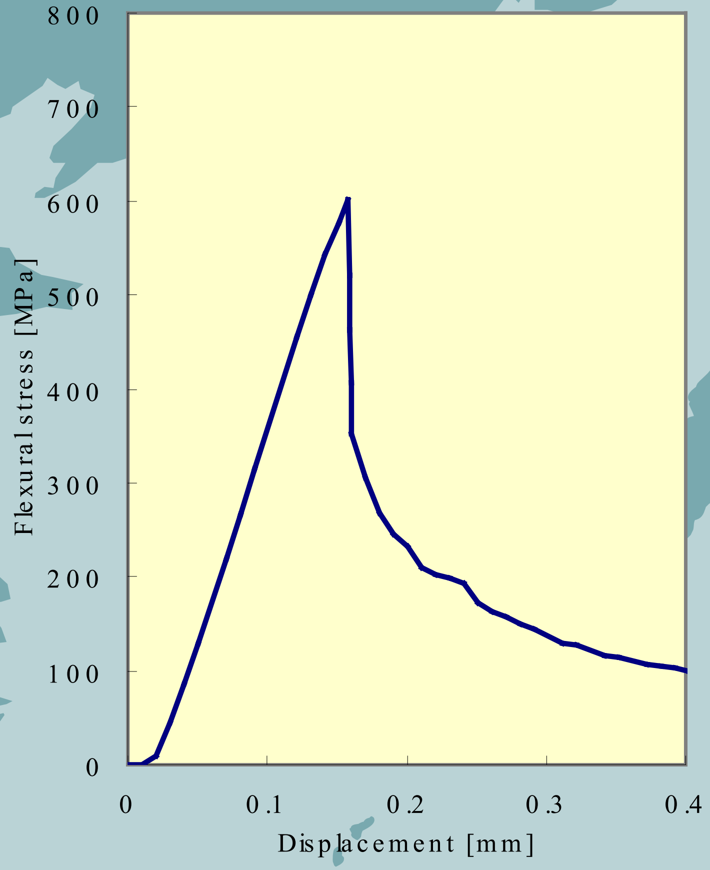
- Specimen Size: 40Wx45Lx4Tmm (27 UD-prepreg sheets stacking)
- Fiber: Tyrano SAK (7.5μm.800fiber/bundle)
- Fiber Coating.Pyro-Graphite.0.5μmt
- Vf: 45vol%
- Fabrication Process: NITE method (1800., 1hr, 200kg/cm^2., in Ar)

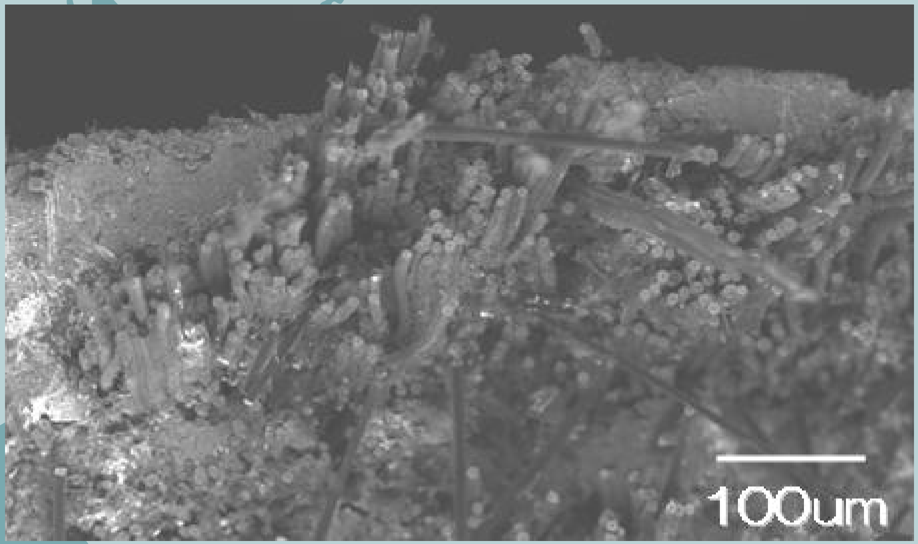


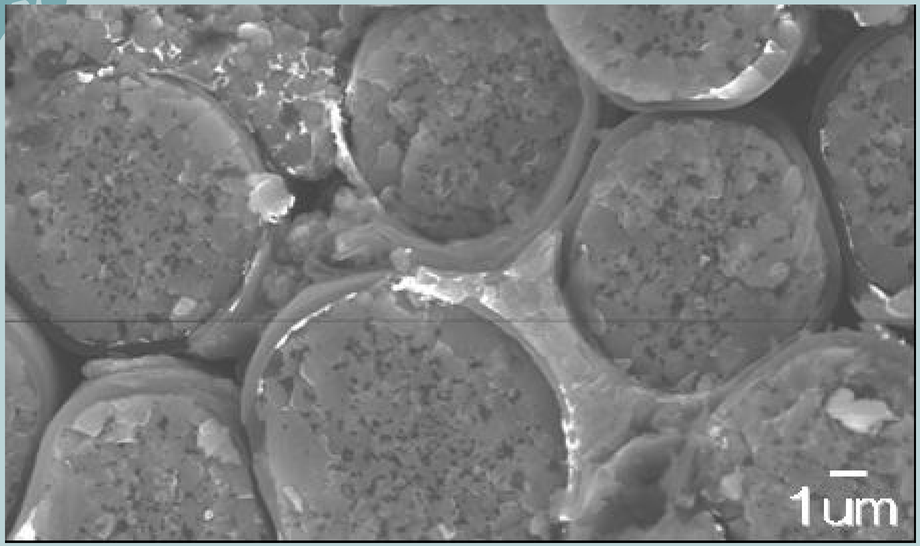
Dimension : 36.48W×40.5L×3.34T mm
Weight : 14.98 g / Bulk Density : 3.03 g/cm^3

Three Point Bending Test Results (Pilot Grade #2-1)

- Specimen Size: 4.2Wx25.2Lx1.7t mm
- Span.18 mm
- Cross Head Speed: 0.5 mm/min

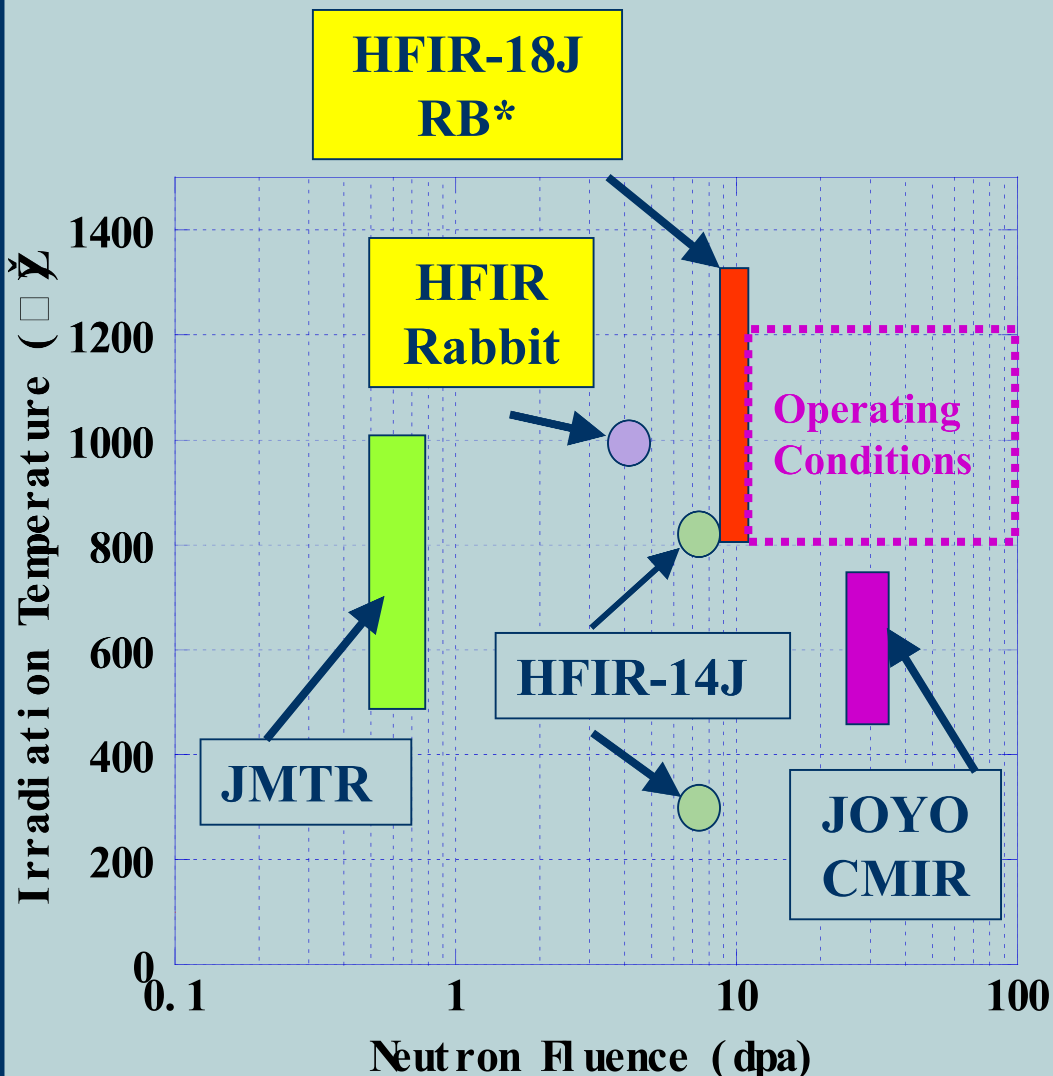






- Three Point Bend Strength: 601 MPa
- Elastic Modulus.150 GPa
- Fracture Mode.
  Pseudo Elasticity with Fiber Pull Outs

Neutron Irradiation Research Status



Issues to be studied

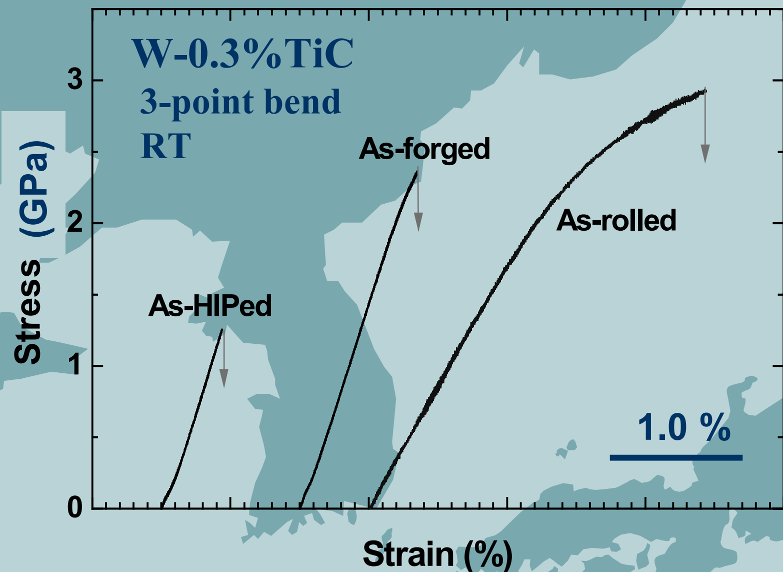
- Max. Temperature Limit under Irradiation
- Compatibility of solid breeding materials and SiC or SiC/SiC
- Irradiation behavior of joining or coating

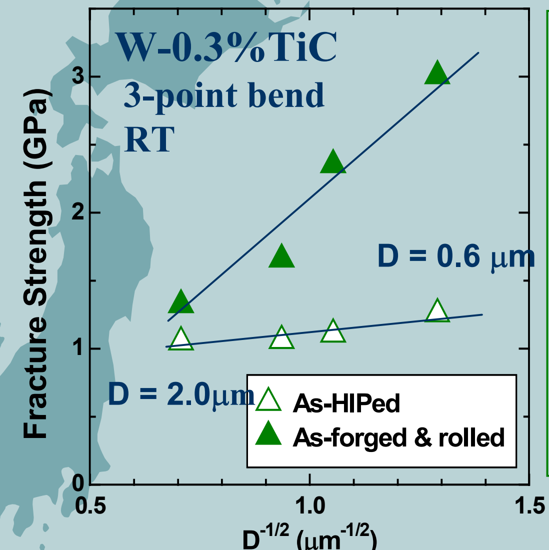
Feasibility studies and design activities have been started since 2003.

Research Emphasis and Status (4)

Refractory Alloys

Development of Ultra-Fine Grained, Nano-Particle Dispersed W-TiC Alloys with Improved Ductility





The beneficial effect of hot working becomes prominent with decreasing grain size of HIPed compacts.

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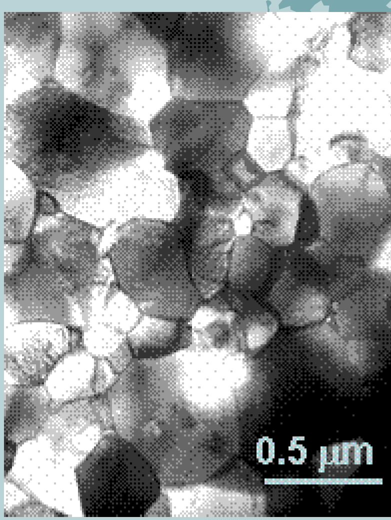
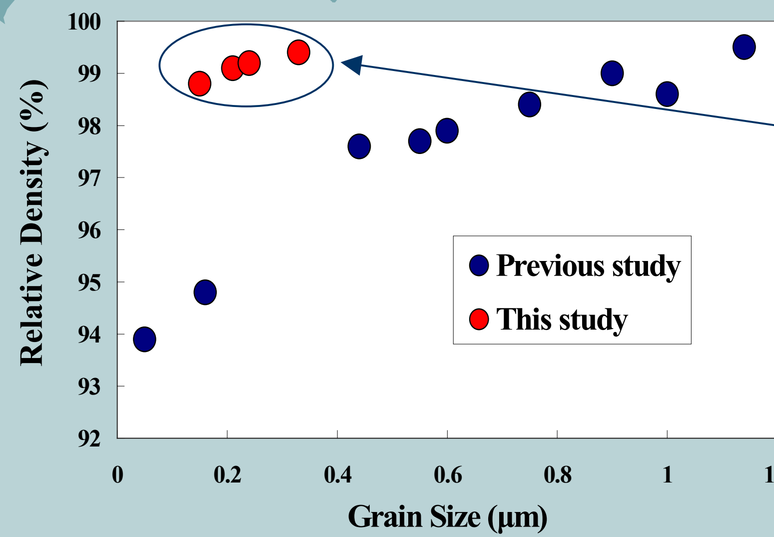
Need of consolidated body with very small grain size less than 0.5 μm

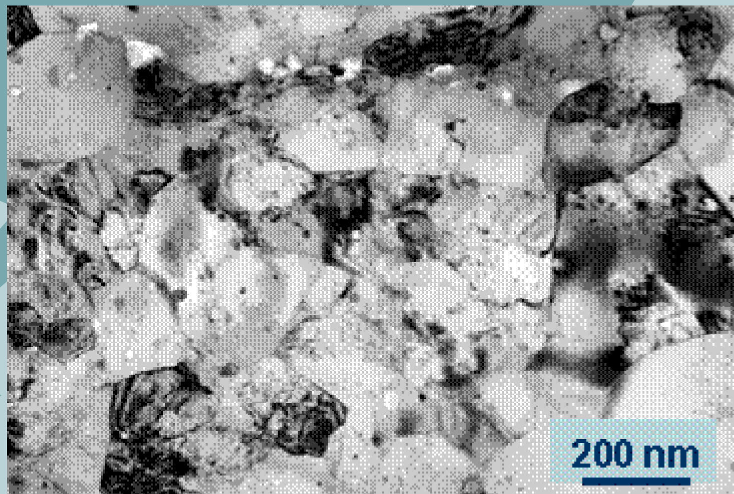


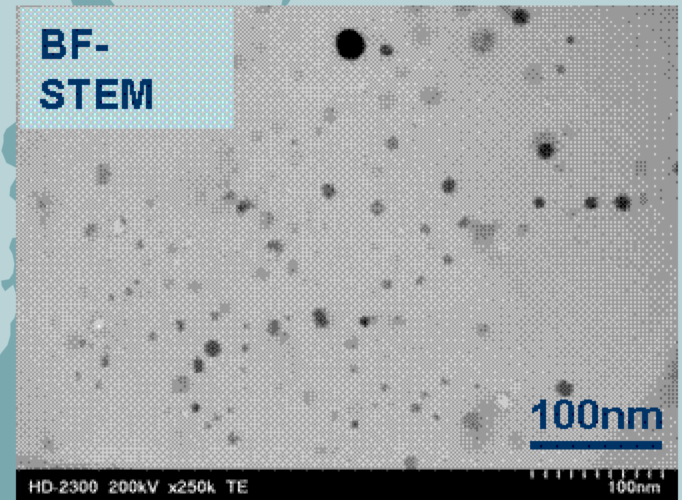
Relative Density vs Grain Size

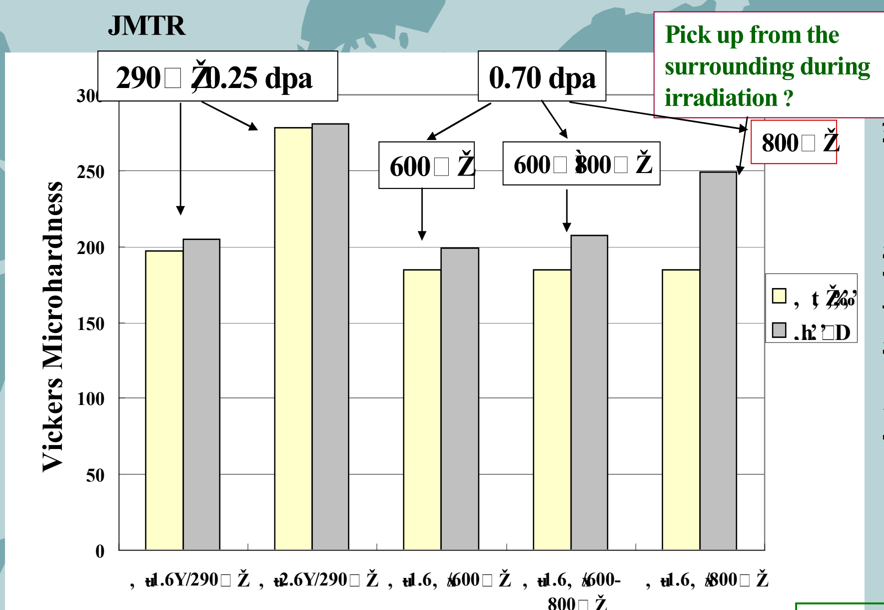


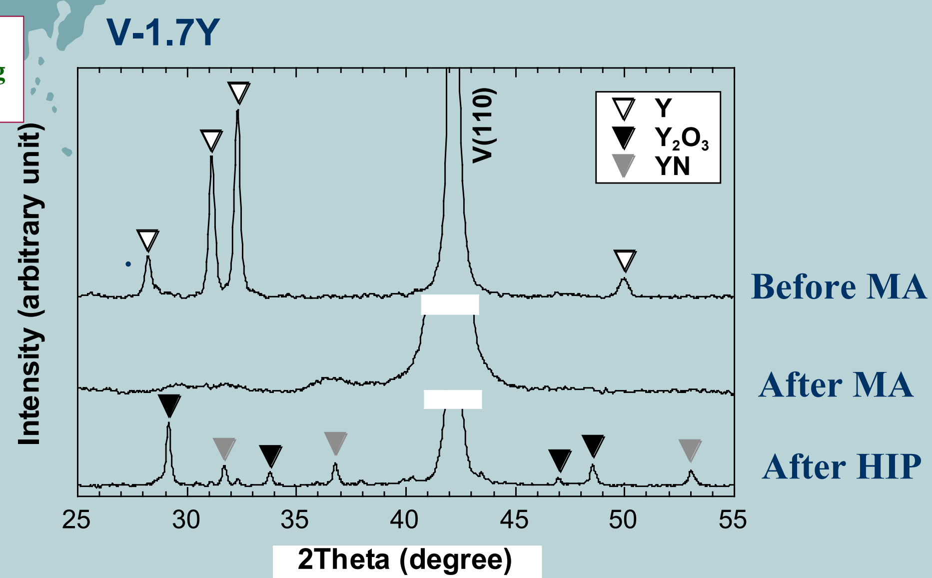
Success in fabrication of ultra-fine grained W-TiC with a high relative density of ~99%

Development of Ultra-Fine Grained, Nano-Particle Dispersed V-Y Alloys with Good Radiation Resistance









Less hardening by JMTR irradiation

MA causes the added Y particles to dissolve into V. Y_2O_3 and YN form by reaction of Y with solute O and N during HIP.

Research Emphasis and Status (5)

Modeling and Mechanistic Studies

Emphasis in Modeling and Mechanistic Studies

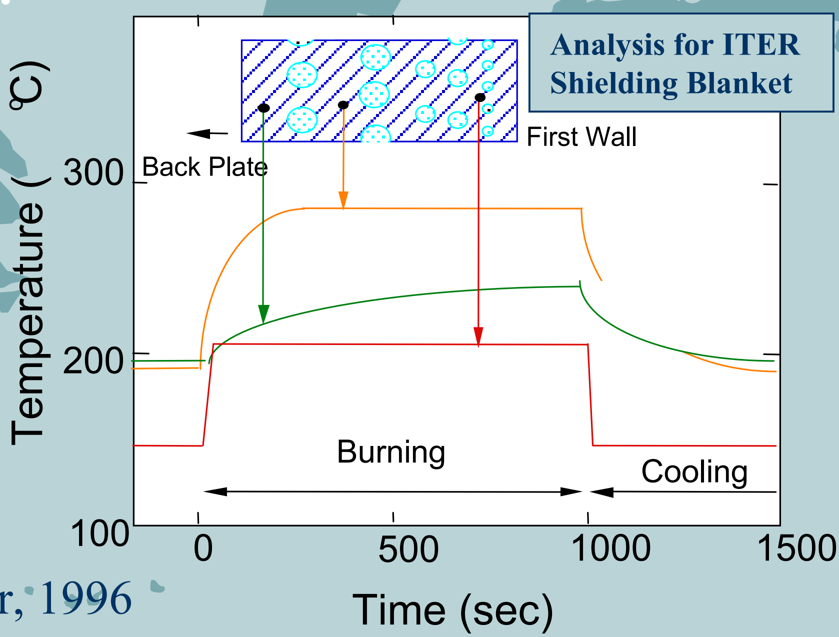
- **Issues of simulation-fusion correlation**
  - Recoil spectrum
  - Transmutation
  - Damage rate
  - Temperature variation
- **Materials performance mechanisms**
  - Fracture and deformation of irradiated materials
- **Test technology**
  - Qualification of small scale test technology
- **System oriented issues**
  - Life time evaluation

Multi Scale Modeling

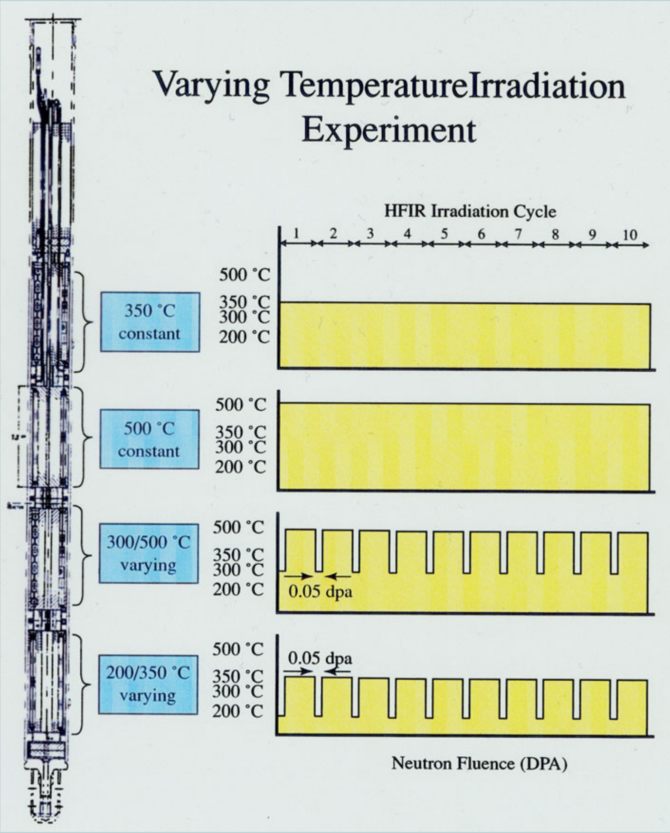
- For understanding materials performance under irradiation, analyses of physical phenomena in quite wide range of time and scale are required
  - psec to decades
  - sub nm to m
- Various modeling with quite different length and time scales are developed and interrelated with each other
  - First principle atomistic modeling of point defect properties
  - Molecular Dynamics simulation of primary defect production
  - Kinetic Monte Carlo simulation of defect diffusion
  - Rate Theory of defect accumulation, solute segregation and precipitation, void swelling and creep
  - 3-D Dislocation Dynamics simulation for mechanical property change
- Multi Scale Testing and Evaluation Research (MUSTER) facility is introduced in IAE Kyoto University including JEOL F-2200 FE TEM with omega filter

Varying Temperature Irradiation Experiment

- Fusion reactor materials will be subject to temperature transient during irradiation
- Symmetrical isothermal and temperature-variant irradiation was carried out in HFIR by JUPITER project and have been conducted in JMTR
  - 10% low temperature excursion



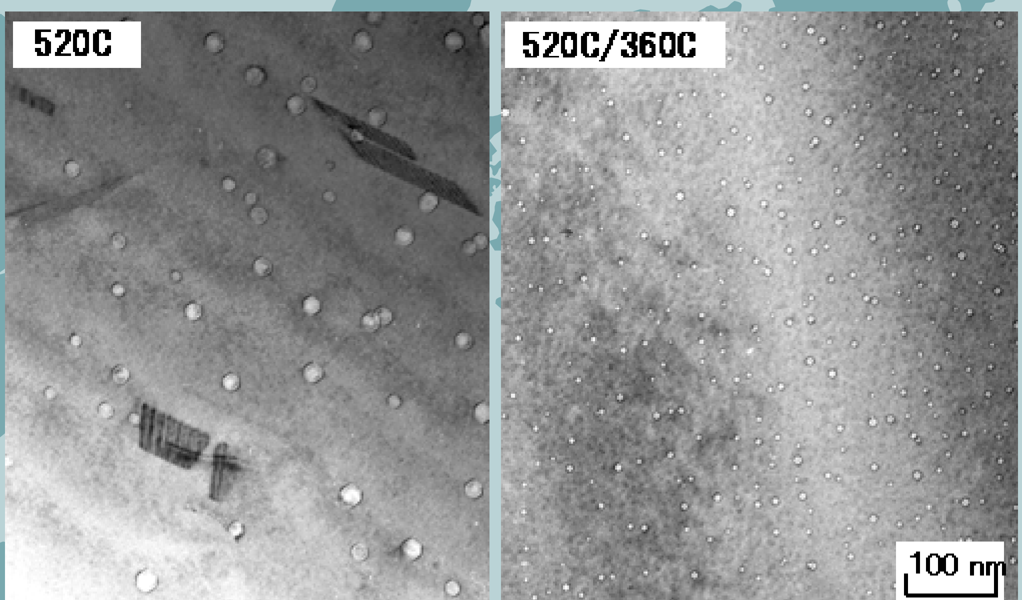
Heller, 1996

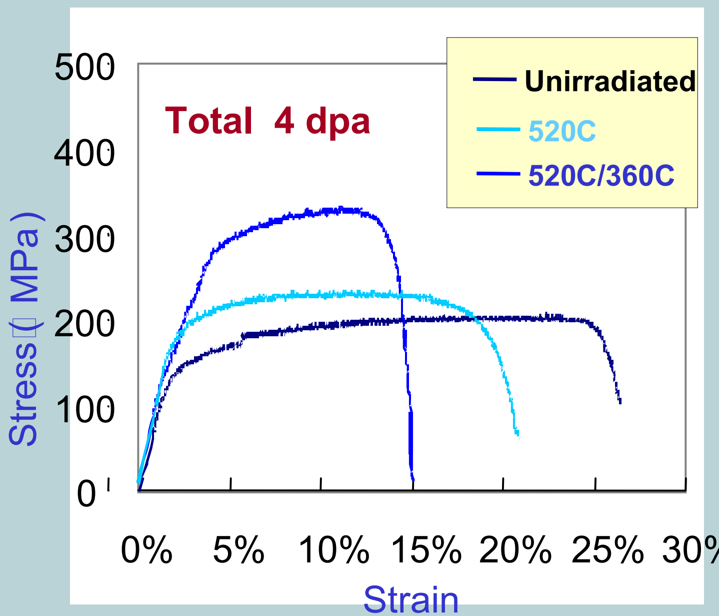


Muroga, 2001

Examples of Varying Temperature Effects

- **Lower temperature transient resulted in increase in void density and decrease in void size in unalloyed vanadium**
  - **Mechanical property is consistent with the microstructures**
  - **The difference explained by rate theory analysis**
  - **However, the temperature transient effects are generally more moderate for radiation-resistant materials (e.g. V-4Cr-4Ti)**





Watanabe, Fukumoto unpublished

Summary (1)

- The realization of fusion power system with advantages in safety, environment and economy depends on the development of high-performance and low activation materials
- Efforts are focused on developing RAFM, vanadium alloys and SiC/SiC composites
  - Compositional and microstructural optimization to yield the necessary properties
  - Technology development specific to the design concepts
- Irradiation effects with helium is the common critical issue
  - Strongly motivates intense 14MeV neutron irradiation facility

Summary (2)

- Continuous use of the present irradiation facilities in elaborate manners will enhance the efficiency and soundness of the materials development
- Modeling and fundamental studies of radiation effects play a very important part in enhancing the predictability of materials performance in fusion conditions based on the limited irradiation data.

核融合炉用低放射化フェライト JLF-1 鋼

大同 9 ﾄﾝ炉溶解スラブ

〔インゴット〕

表1　溶解材の化学組成 mass%

C	Si	Mn	P	S	Cr	V	Al	Ta	W	Ni	Co	Mo	B	N
0.10	0.10	0.46	0.003	0.003	8.94	0.20	0.002	0.08	1.99	0.06	0.001	0.006	0.0003	0.019

〔ｽﾗﾌﾞサイズ〕

- 厚さ 130×幅 1200×長さ 2000　→40mm 厚に圧延
- 厚さ 110×幅 1200×長さ 2000　→18mm または 25mm 厚に圧延

〔製造条件（予定）〕

20≦t≦40mm：	1150℃加熱－860℃CR－CLC	760℃Temper
10≦t＜20mm：	1150℃加熱－900℃CR－AC	760℃Temper