# Fusion Reactor Technology I
## (459.760, 3 Credits)

**Prof. Dr. Yong-Su Na**

(32-206, Tel. 880-7204)

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# Current status and future issues of blanket development

# Specifications and characteristics

- **Tritium production and release characteristics**
  - fuel self-supply
- **Temperature control characteristics by the high-temperature coolant for electricity generation**
  - high temperature for high efficiency
- **Sufficient shielding characteristics**
  - VV, superconducting magnets, surrounding components, bio-shield
- **Long term durability of the blanket structure**
  - withstanding high surface heat flux, neutron wall loads, strong EM loads, high irradiation fluence, many operation cycles, exposure to chemicals during operation

# Specifications and characteristics

- **High safety, reliability, and environmental susceptibility**
  - no triggering of an initiating event for accidents in an off-normal condition
  - minimizing the potential hazards and radio-activation

- **High economic factors**
  - high efficiency electricity generation
  - reducing fabrication costs
  - recycling used breeder material
  - reducing the duration of remote handling blanket maintenance (availability)

# Blanket type and the development status - Unique merit of solid blanket

- **Selection of solid breeder type**
  - small thermo-chemical activity of elemental material
  - Tritium inventory can be kept relatively low.
  - Basic technology for tritium recovery is already established.

- **Pebble bed utilisation for breeder and multiplier layers**
  - Application of a pebble bed structure may reduce fracture by reducing the influence of the degradation of the thermo-mechanical properties.

- **Pressurised water as the candidate coolant**
  - having sufficient experience

# Blanket type and the development status - Unique merit of solid blanket

- **RAFS as the candidate structural material**
  - superior characteristics for both irradiation and a wide range of high-temperature usage in industry

- **Possibility of performance upgrade**
  - upgrading to He gas coolant and innovative structural materials, such as ODS, SiC/SiC composites
  - Upgrades do not require major changes in the design.

# Blanket type and the development status - Liquid blanket

- **Merits**

- not having irradiation degradation in the breeder material
- less stringent high-temperature limit for the breeder material

- **Potentials**

- high heat removal
- Adequate tritium breeding ratio appears possible without beryllium neutron multiplier in Li, PbLi (Pb serves as a multiplier in PbLi).
  Cf. Note that molten salts, e.g FLiBe has beryllium part of the salt and generally requires additional separate Be.
- Relatively simple design
- Low pressure, low pumping power (if MHD problems can be overcome)

# Blanket type and the development status - Liquid blanket

- **Issues**

  - LiPb: Po (polonium) generation by nuclear transmutation,
    Li-Pb fraction change in the course of breeding T from Li

- **Molten salt blanket**

  - reducing the MHD pressure loss
  - reducing the chemical reactivity compared to the liquid Li blanket

# Blanket type and the development status - Solid blanket

- **Major blanket types under development**

| type | | Solid Breeder Blanket | | Liquid Breeder Blanket | |
|---|---|---|---|---|---|
| Material | Breeder | Ceramic Breeder | Ceramic Breeder | LiPb | Li |
| | Structure | Ferrite | Ferrite | Ferrite | V Alloy |
| | Coolant | Pressurized Water | Helium | Pressurized Water | Liquid Li self-cool |
| Advantages | | Safety<br>Sufficient database<br>Wide base of in-dustrial technology | Safety<br>Sufficient database<br>High electricity generation effi-ciency | Less irradiation damage on breeder<br>Breeder multiplies neutron | No irradiation damage on breeder<br>Simple blanket ge-ometry |
| Disadvantages | | Irradiation damage<br>Complicated con-figuration | Irradiation damage<br>Complicated con-figuration<br>Shielding perform-ance | Tritium permeation<br>Heavy breeder mass and high power for forced-flow<br>Safety concern for liquid metal<br>Less database | MHD<br>Pressure drop<br>Uncertain tritium recovery technol-ogy<br>Safety concern for liquid metal<br>Less database |
| Working Party | | Japan | Japan, EU, RF, US | EU | RF, US |

# Blanket Concepts

- 증식재의 형태에 따라 고체 증식재와 액체 증식재로 나뉨.

| 고체증식재 | 액체증식재 |
|---|---|
| $Li_2O$, $Li_2TiO_3$, $Li_2ZrO_3$, $Li_2SiO_4$ | 액체 리튬, 액체금속,<br>FLiBe (용융염: molten salt fluids) |
| 화학적으로 안정<br>잠재적인 안정성 높음<br>구조재와의 양립성이 우수함 | 방사선 손상 경미<br>높은 TBR |
| 중성자 조사에 의한 손상 | 화학적으로 활성 – 구조재 부식<br>액체리튬의 안정성 문제<br>MHD 압력 손실로 인한 유속 감소:<br>전기절연막 필요 |

# Tritium Breeding

- **Liquid Breeder Concepts**

| Li | $Li_{17}Pb_{83}$<br>(Lithium-lead eutectic) | FLiBe<br>($LiF \cdot BeF_2$)<br>(Molten salt fluids) |
|---|---|---|
| 트리튬 회수 어려움:<br>수소동위체의 용해도가 큼 | 트리튬 회수 쉬움:<br>트리튬 용해도가 매우 작음 | |
| 트리튬의 구조재료를 통한 투과누출이 작음:<br>트리튬이 Li 중에 모임 | 투리튬의 투과누출이 큼:<br>구조재로의 세라믹 코팅막 등의 투과장벽이 필요 | FLiBe 중의 트리튬 화학형 TF나 $T_2$에 의해 구조재의 부식 증가 또는 트리튬 추가누출 증대 |
| | | 화학적으로 안정하고 고온 사용이 가능 |

# Power Extraction

| 냉각재 | 장점 | 단점 |
|---|---|---|
| 물(경수) | - 전열특성이 좋음<br>- 비교적 저유속으로 큰 제열성능을 얻음.<br>- 자장의 영향을 받지 않음.<br>- 펌프동력 양호<br>- 구조재와의 공존성 높아 차폐성능 양호<br>- 경수로 기술 적용 가능 | - 중성자 흡수반응 단면적이 큼 (TBR 저하)<br>- 냉각수의 로내 및 증식영역으로의 누출에 의한 압력상승 대책 필요 |
| He gas | - 화학적으로 불활성, 취급 용이<br>- 구조재와의 공존성 양호<br>- 고온 취급 가능으로 고발전효율 기대 | - 열용량이나 열전달률이 비교적 작아 제열한계가 낮음.<br>- 펌프동력이 커짐.<br>- 차폐성능이 낮아 차폐체가 두꺼워짐. |
| 액체 금속 | - 전열특성이 양호<br>- 저압에서 고온운전 가능<br>- 냉각재와 증식재를 겸함으로 인해 블랭킷 구조의 간략화<br>- 반응생성물의 인출이나 성분조정 등을 연속해서 할 수 있음. | - 화학적으로 활성<br>- MHD 압력 손실이 큼 (전기절연피복 설치 또는 기액이층류로 전기전도율 내리는 방법 등 고려) |

# Flows of electrically conducting coolants will experience complicated magnetohydrodynamic (MHD) effects

**What is magnetohydrodynamics (MHD)?**

- Motion of a conductor in a magnetic field produces an EMF that can induce current in the liquid. This must be added to Ohm's law:

$$\mathbf{j} = \sigma(\mathbf{E} + \mathbf{V} \times \mathbf{B}) \qquad \mathbf{E} + \mathbf{V} \times \mathbf{B} = \eta \mathbf{j}$$

- Any induced current in the liquid results in an additional body force in the liquid that usually opposes the motion. This body force must be included in the Navier-Stokes equation of motion:

$$\frac{\partial \mathbf{V}}{\partial t} + (\mathbf{V} \cdot \nabla)\mathbf{V} = -\frac{1}{\rho}\nabla p + \nu \nabla^2 \mathbf{V} + \mathbf{g} + \frac{1}{\rho}\mathbf{j} \times \mathbf{B}$$

- For liquid metal coolant, this body force can have dramatic impact on the flow: e.g. enormous MHD drag, highly distorted velocity profiles, non-uniform flow distribution, modified or suppressed turbulent fluctuations

# What is turbulence?

- Reynolds number: Re=VL/ν ← $(V^2/L)$ / $(\nu V/L^2)$

$$\frac{\partial \mathbf{v}}{\partial t} = -\underbrace{(\mathbf{v}\cdot\nabla)\mathbf{v}}_{V^2/L} + \underbrace{\nu\nabla^2\mathbf{v}}_{\nu V/L^2}$$

- When Re << $Re_{critical}$, flow = laminar
  When Re >> $Re_{critical}$, flow = turbulent

# Main Issue for Flowing Liquid Metal in Blankets: MHD Pressure Drop

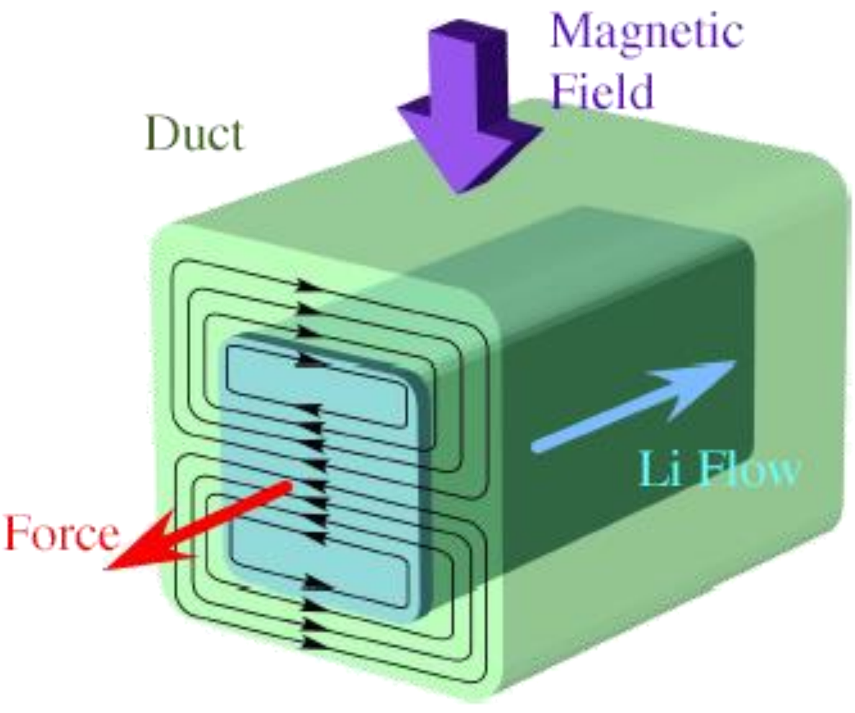
Feasibility issue – Lorentz force resulting from LM motion across the magnetic field generates MHD retarding force that is very high for electrically conducting ducts and complex geometry flow elements

***Thin wall MHD pressure drop formula***

$$\Delta p_{MHD} = LJB \approx L\sigma VB^2 \underbrace{\frac{\sigma_w t_w}{\sigma a}}_{c}$$

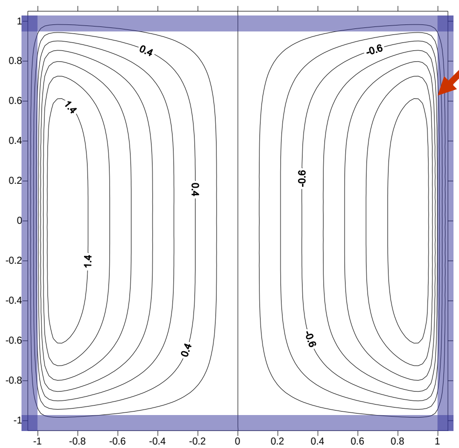
***p, pressure***
***L, flow length***
***J, current density***
***B, magnetic induction***
***V, velocity***
***$\sigma$, conductivity (LM or wall)***
***a,t, duct size, wall thickness***

**Why?**

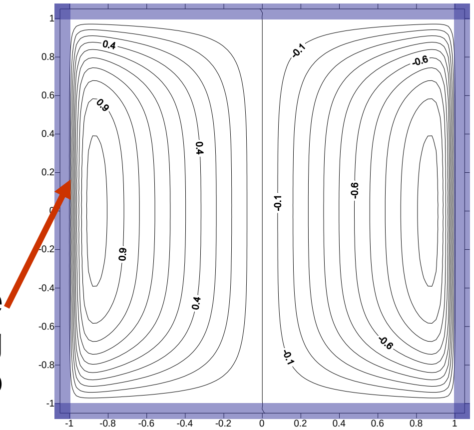


# A perfectly insulated "WALL" can eliminate the MHD pressure drop. But is it practical?

**Conducting walls**



**Insulated walls**



**Lines of current enter the low resistance wall – leads to very high induced current and high pressure drop**

**All current must close in the liquid near the wall – net drag from jxB force is zero**

- **Net JxB body force $\nabla p = c\sigma VB^2$ where $c = (t_w \sigma_w)/(a\, \sigma)$**
- **For high magnetic field and high speed (self-cooled LM concepts in inboard region) the pressure drop is large**
- **The resulting stresses on the wall exceed the allowable stress for candidate structural materials**

- **Perfect insulators make the net MHD body force zero**
- **But insulator coating crack tolerance is very low ($\sim 10^{-7}$).**
  - **It appears impossible to develop practical insulators under fusion environment conditions with large temperature, stress, and radiation gradients**
- **Self-healing coatings have been proposed but none has yet been found (research is on-going)**

# Major R&D status and future issues – Solid blanket

- **Fabrication technology development**
  - **Structural material:** RAFS (JLF-1, F82H, etc) optimised, need to adjust the composition to meet specific mechanical strength requirements
  - **Blanket box structure fabrication including the FW:** need to optimise bonding conditions and the accumulation of mechanical data on bonded materials
  - **Breeder and multiplier pebble mass fabrication technology:** agglomeration method and the sol-gel method fabrication techniques for breeder pebble fabrication, rotating electrode method for multiplier pebble fabrication

# Major R&D status and future issues – Solid blanket

- **T breeding and recovery technology development**
  - **Thermo-mechanical characteristics research for breeder and multiplier pebble bed:** appropriate temperature range for proper T release and preserving the mechanical integrity of the pebble bed. The mechanical characteristics of a pebble bed are a new area of research incuding combined behaviour of the thermal and mechanical characteristics and irradiation effects.
  - **T generation and release characteristics:** BEATRIX-II, JMTR
  - **T recovery and fuel cycle technology:** TPL (JAEA), TSTA (LANL)

- **Cooling technology development**
  - **Coolant handling technology:** pressurised water and He cooling technology already established by experience with PWR, BWR, HTG test R
  - **FW cooling technology:** improving heat transfer of the built-in cooling channel of the FW panel

# Major R&D status and future issues – Solid blanket

- **Durability development, such as irradiation characteristics**
  - **General aspects:** certifying the irradiation performance of materials, degradation of materials by thermal cycles and long-term operation, FW durability in high-heat flux, and chemical effects (corrosion, mass transfer, etc)
  - **Structural material:** need to clarify He production and the H embrittlement effect
  - **Breeder material:** need to investigate the irradiation effects on the thermo-mechanical characteristics
  - **Multiplier material:** need to formulate Be oxidation rate and corrosion rate of contacting structureal materials

# Major R&D status and future issues – Solid blanket

- **Safety and environmental susceptibility development**
  - **General issue:** T inventory reduction, evaluation of off-normal performance, development of reduced-activation materials, reduction and recycling of radioactive waste
  - **T inventory:** by adjusting breeder temperature within the proper range, it can be reduced to less than 1 kg.
  - **Reduction of induced activation:** RAFS
  - **Off-normal performance evaluation:** The largest impact caused by loss of coolant in TBM box (ITER). Further investigation needed on H generation reaction between Be in contact with water in a high temperature environment
  - Innovative material development

# Major R&D status and future issues – Solid blanket

- **Economically reduced cost development**
  - **Remote handling technology:** important to increase the reactor availability, affecting the design of the hot cell facility, reactor building, etc.
  - **Blanket replacement strategy:** time saving replacement method (whole sector replacement)

# Major R&D status and future issues – Liquid blanket

- **Liquid Li self-cooled blanket**
  - development of an electrical insulation coating to reduce MHD pressure drop
  - evaluation of heat transfer and hydraulic characteristics of liquid Li in a strong magnetic field
  - evaluation of compatibility between liquid Li and structural materials (화학적으로 활성)
  - establishment of safe handling techniques for liquid Li
  - development of industrial bases for V alloys and box structure fabrication technology
  - heavy irradiation data for V alloys

# Major R&D status and future issues – Liquid blanket

- **LiPb blanket**

- development of T permeation barrier coatings
- evaluation of the corrosion effect of LiPb on structural materials
- establishment of T recovery technology

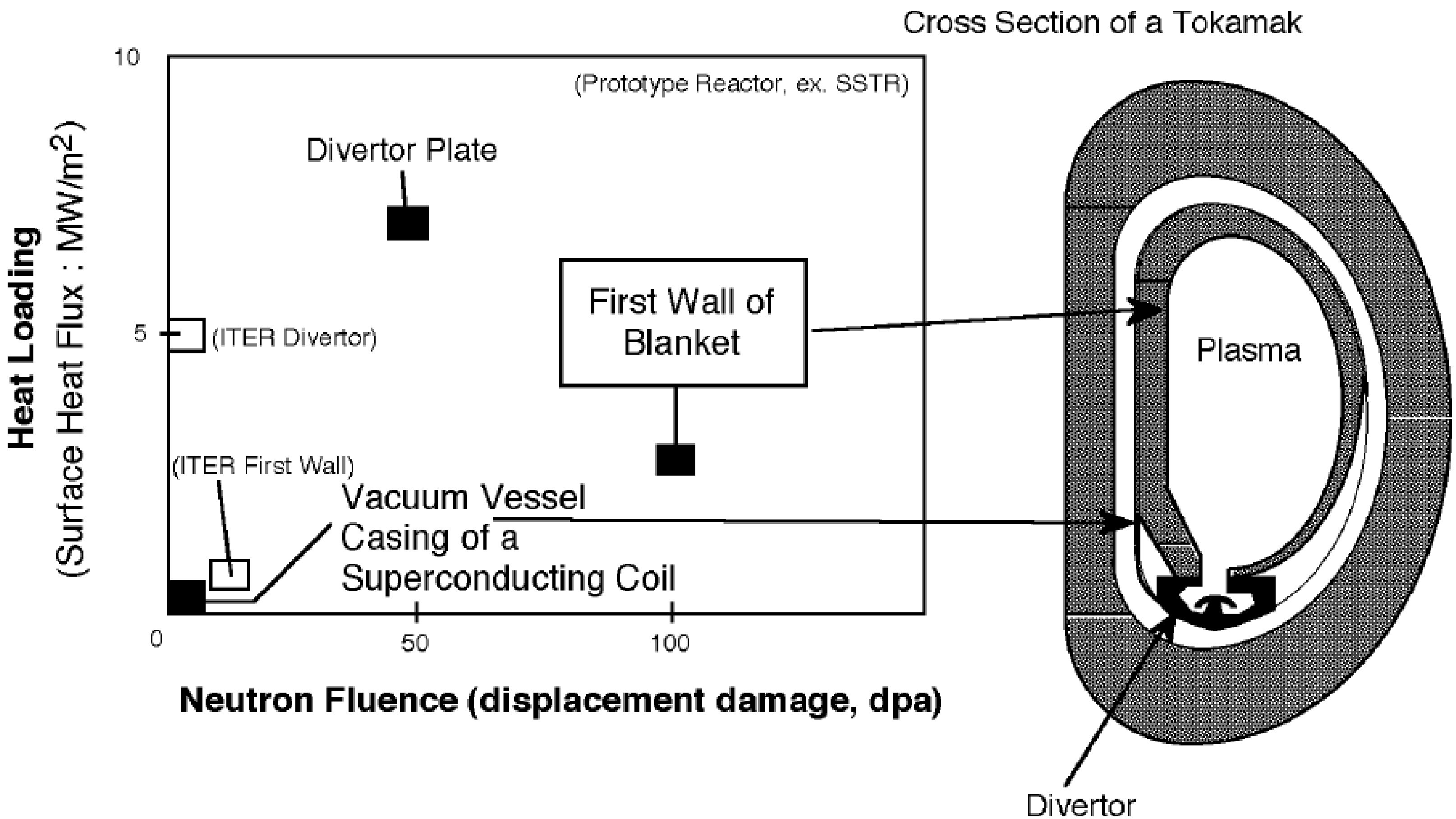
- **Molten slat (FLiBe) blanket**

- development of T safe confinement technology
- development of corrosion resistance technology
- development of T and chemical stability control technology
- development of FLiBe handling technology and F chemical potential control technology

# Current status and future issues of materials development

# Structural materials

- **Major structures of the fusion reactors and their operating conditions**



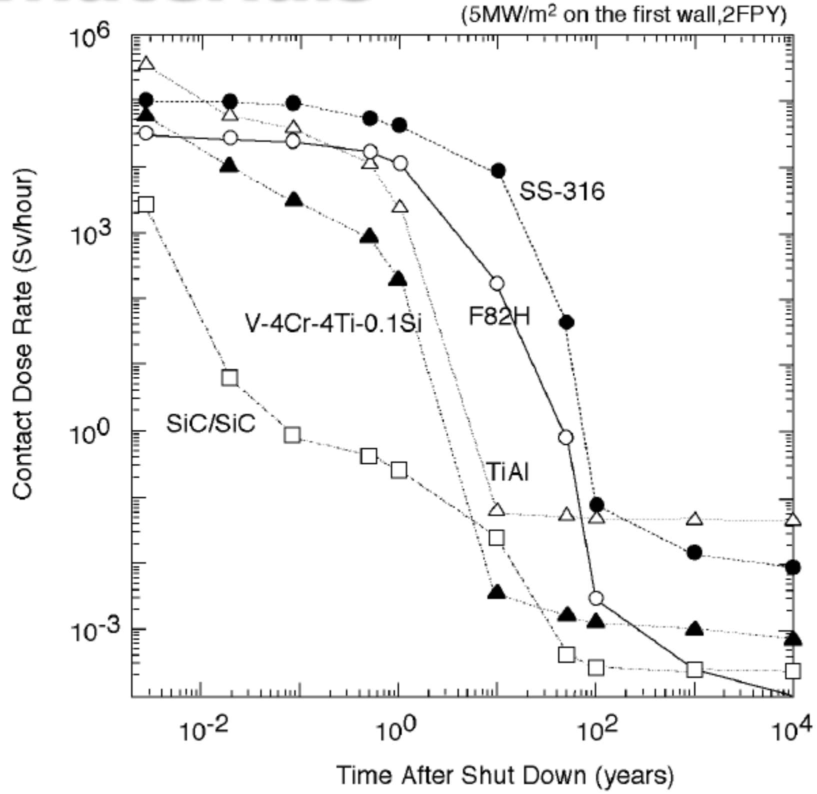
# Structural materials

- **Requirements for reduced activation**
  - low decay heat during maintenance
  - low-induced activity acceptable for the shallow land burial and materials recycling

- **Alloy development**
  - need to manage the property changes during service for extending the lifetime of the power plant
  - need to apply the alloy designing method based on the knowledge of the radiation induced microstrucural change

- Due to severe service condition, a rather long time will be needed for the development and this program should be carefully planned and managed.

# Structural materials

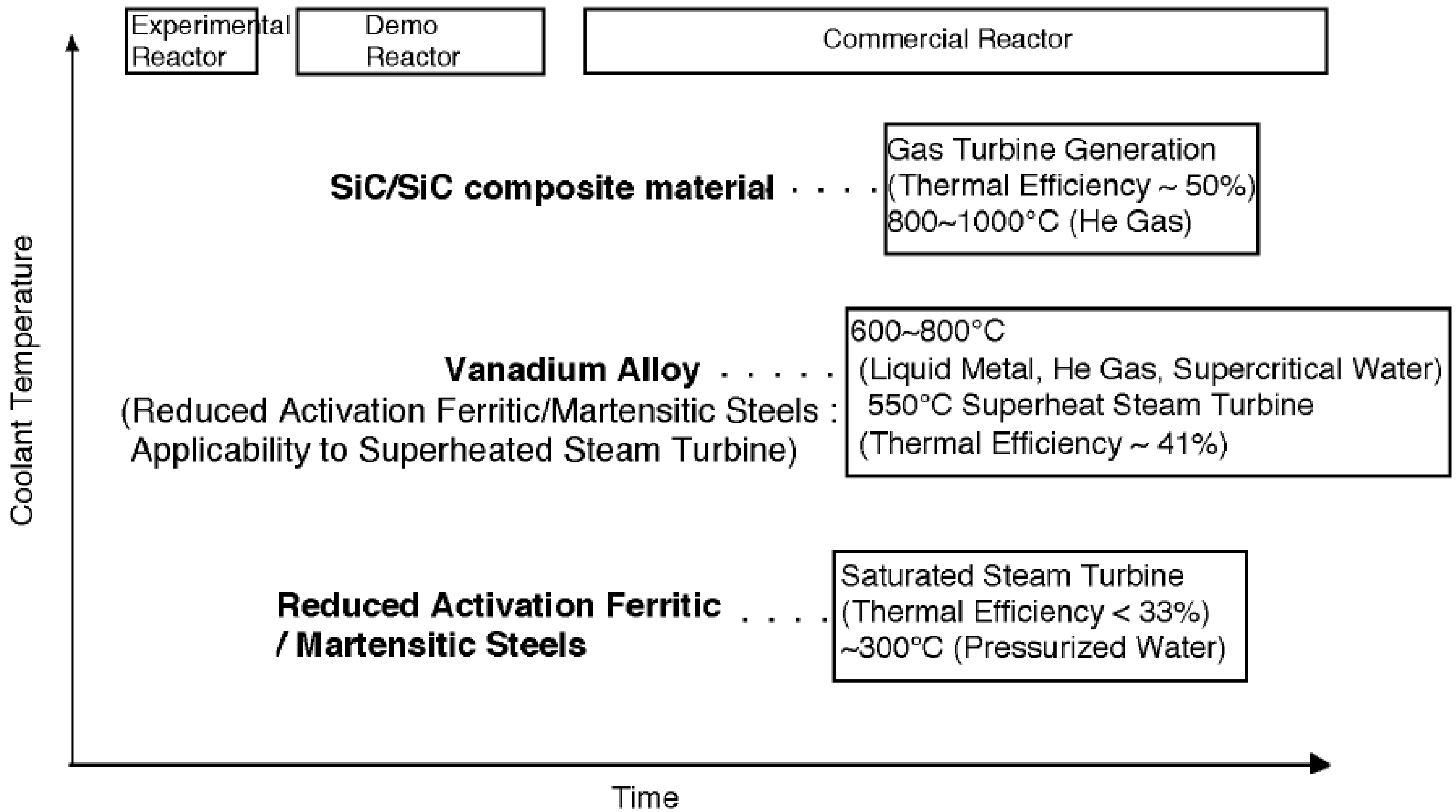



- **Time evolution of contact dose rate in the FW assuming the periodic displacement at a fluence of 10 MWa/m²**

- shallow land burial supposed to be utilised after 100 years of cooling
- replacement of alloying elements by reduced activation elements is essential to reduce the induced activity below the acceptable level for shallow land burial.

# Structural materials

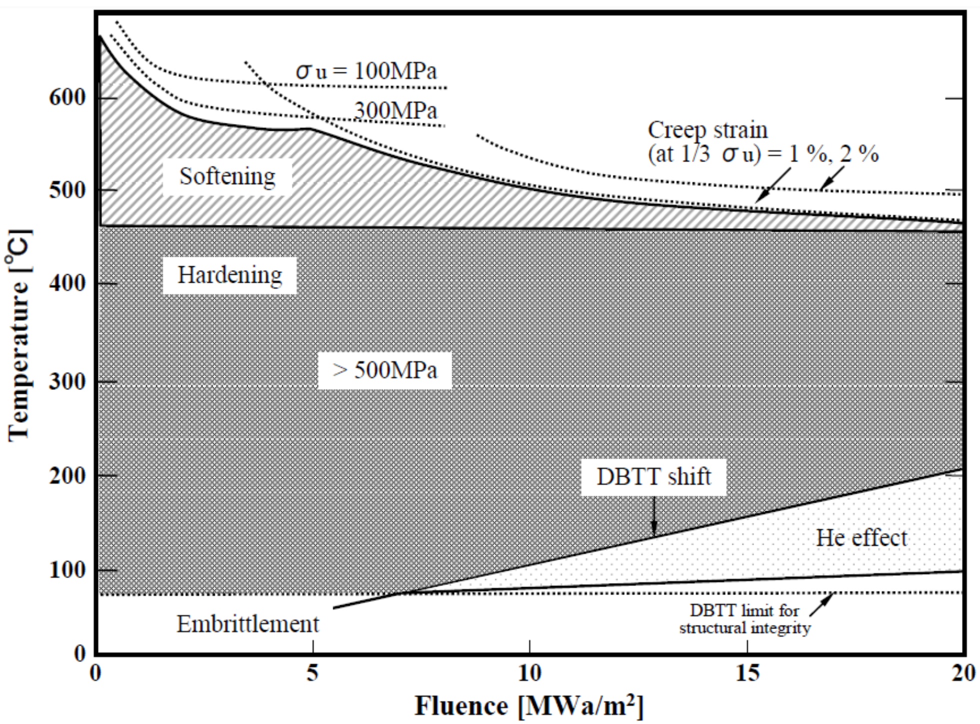
- **Relations of energy systems and their materials**



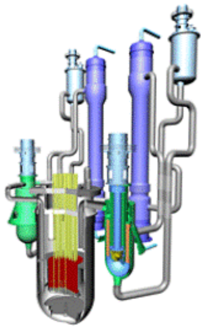
# Structural materials

- Development of structural materials and their target performances in feasible temperatures and neutron fluences

- lower bound temperature: limited by the embrittlement during irradiation (DBTT; Ductile-Brittle Transition Temperature)
- upper bound temepráture: limited by the transmutation-produced He embrittlement and irradiation creep




- Ferritic/martensitic steels have been used successfully as duct materials of the fuel assemblies for FBRs to a displacement damage level of about 150 dpa (~15 MWa/$m^2$).



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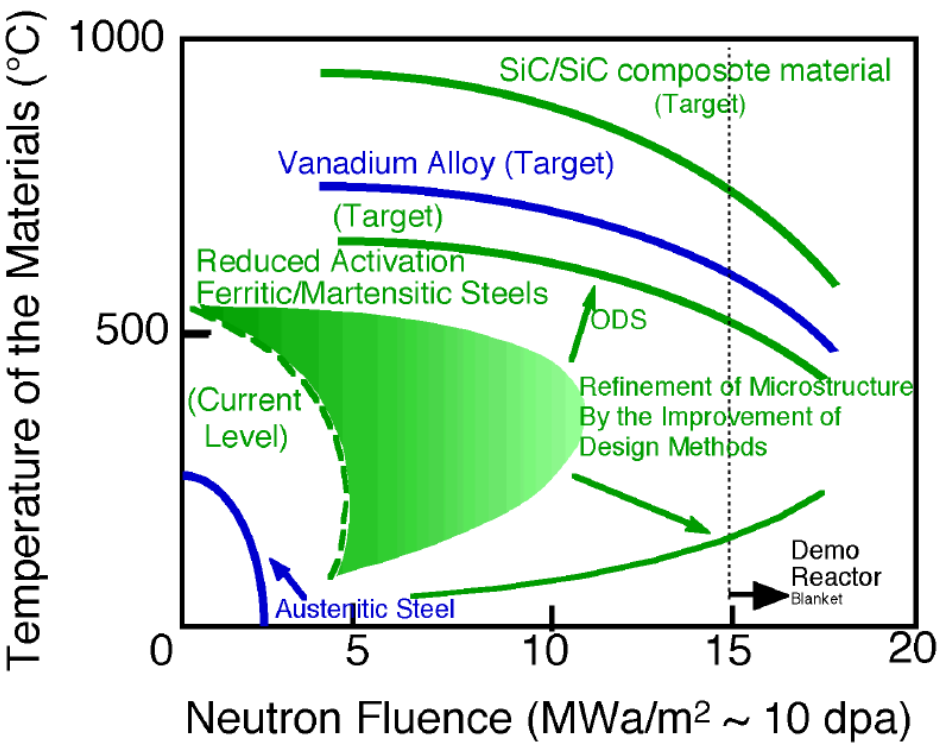


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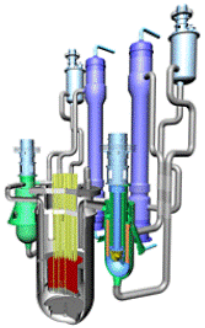

# Structural materials

- **Critical issues for the development of RAFS**

- to manage the radiation induced embrittlement at low temperatures
- **improvment of the high-temperature strength:** dispersion within the alloy of nanometer-size oxide particles
- **improvment of the corrosion resistance:** composite materials including graded materials technology
- improvment of processes to enable the large-scale production

- **Managing the radiation induced embrittlement**

- strongly affected by the produced He and H due to transmutation
- Addition of the minor alloying elements and the optimisation of the mechanical heat treatment to make fine dispersion of radiation produced He and H cavities seems to be effective in retarding embrittlement.
- need to apply the recent progress of the fracture mechanics utilising the margins of the small-size components to brittle fracture and utilising the experience of ITER

# Structural materials

- **Load during the 30-year-operation of FPP**

  - beyond the present ability to estimate the lifetime of FS based on FBR experience and the present knowledge about irradiation induced property changes

| | |
|---|---|
| Neutron wall load | 90 $MWa/m^2$, 3 $MW/m^2$ |
| Displacement damage | 900 dpa |
| He generation ratio | 10000 appm |
| H generation ratio | 40000 appm |

# Structural materials

| Items | Experimental reactor | Prototype reactor | Commercial reactor |
|---|---|---|---|
| Blanket structural materials | - Shielding blanket: 316 stainless steel (ITER grade)<br>- Test blanket: Reduced activation ferritic/martensitic steel and Vanadium alloys | Reduced activation ferritic/martensitic steel (RAF/M), Vanadium alloys, SiC/SiC composite | Reduced activation ferritic/martensitic steel, Vanadium alloys, SiC/SiC composite |
| Technological goal | - Shielding blanket: Shielding of neutrons to a life corresponds to 0.3MWa/m2<br>- Test blanket: Trial of tritium production. Demonstration of life corresponds to 0.3MWa/m2 at high temperatures | Structural materials (and the structure) are required to be compatible with the tritium production and with the heat removal to extract thermal energy during a lifetime corresponds to 10MWa/m2 of neutron wall loading.<br>Maximum temperature for RAF/M is 500C (for water cooling system), that for the vanadium alloy is 700C (for liquid metal cooling) and that for SiC/SiC composite is expected to be 1000C for He gas cooling. | Material lifetime is required to extend to about 20MWa/$m^2$. |

# Structural materials

| | | | |
|---|---|---|---|
| Current status | - Most of the basic properties including the irradiation effect have been evaluated. Results are indicating the alloys are acceptable for application. Some of the subjects, such as the effect of thermal cycling during fabrication on IASCC will be evaluated.<br>- Test blanket: Major composition of the reduced activation ferritic/martensitic steels and vanadium alloys have been established. Because the damage level is expected to be relatively low, materials available are expected to be acceptable. | Major chemical compositions of the three leading materials have been established. Irradiation properties of these materials are in progress. Technologies of processing and engineering basis for application except for the irradiation effect have been almost established from the industrial experiences of non-reduced activation ferritic/martensitic steels. For vanadium alloys and SiC/SiC composites, processing of the materials and the fabrication technologies of components are the important issues. | Requirements about life time and service temperature are rather sever comparing with those for the materials of prototype reactor. The materials are expected to be obtained by the improvement of the materials for prototype reactor. |

# Structural materials

| | | | |
|---|---|---|---|
| Methodology for the development | - Shielding blanket: Establishing of available conditions by examining of the effect of fabrication process on the properties.<br>- Test blanket: Optimization of joining (HIP) conditions in view of the effect of the process on the properties. Evaluation of the compatibility with the breeding materials is needed. Depending on the needs, improvement of design methodology taking the irradiation effect into account may be accomplished. Development of the accelerator driven neutron source is also important. | Improvement of properties by optimization of the additional elements, refinement of structure and the improvement of the processing are expected. Available service condition may be extended by the improvement of the design methodology.Refinement of microstructure before irradiation and that of the damage microstructure produced by irradiation by optimizing alloying element and the heat mechanical treatment are expected to be effective to suppress the degradation of fracture toughness and the He embrittlement at elevated temperatures for the reduced activation ferritic/martensitic steels.<br>CaO and AlN are expected to be promising self-healing insulator coatings for the vanadium alloys. Reaction bonding method for SiC/SiC composite is expected to be effective to improve thermal resistance.<br>Promotion of international collaboration is essential for the acceleration of the development including the task sharing. | For the application of the reduced activation ferritic/martensitic steels, it is essential to improve the high temperature strength by ODS and other technologies. |

# Structural materials

| | | | |
|---|---|---|---|
| Prospects | - Shielding blanket: No major issue is expected except for the effect of the disruption, because of the rather mild service condition.<br>- Test blanket: Also, no major issue is expected except for the effect of the disruption, because of the rather mild service condition. It is expected to establish the design methodology taking the irradiation effect into account. Conceptual design activities and other activities are being carried out. | Alloy development and the development of the design technology are expected to be effective to satisfy the requirements. This seems to me more feasible comparing with the application of SiC/SiC composite materials and other materials. For the application of the vanadium alloys, development of self-healing insulator coating is an essential way. The large scale ingot making technology for the vanadium does not seem to be difficult comparing with other issues. Improvement of irradiation resistance is expected for the SiC/SiC composite. However, it needs some time to examine the feasibility. | Extending of the lifetime will be required. From the experience of the development of the fuel cladding of FBR, some improvements are expected to be done. To satisfy the requirements, further improvement may be expected. |

# Tritium breeder materials

| Items | Experimental reactor (ITER) | Demonstration reactor (DEMO) | Commercial reactor |
|---|---|---|---|
| Li burn-up | ~5% | 5~20% (replace per 2 years) | 5~10% (replace per 2 years) |
| Nuclear heating ($MW/m^3$) | ~50 | ~150 | ~150 |
| Temperature (°C) | 200~400* | 400~1000* | 600~1000* |
| Environment | He (~0.1MPa) | He (~0.1MPa) | He (0.1~10MPa) |

* : These values are example for $Li_2O$ and are difference with tritium breeder materials.

# Tritium breeder materials

| Items | Experimental reactor (ITER) | Demonstration reactor | Commercial reactor |
|---|---|---|---|
| Tritium breeder | Material : $Li_2O$, $Li_2ZrO_3$, $Li_2TiO_3$, $Li_4SiO_4$<br>Shape : pebble | Material : $Li_2O$, $Li_2TiO_3$ or improved material<br>Shape : pebble<br>For the other material, i.e. liquid lithium, molten salt , etc | The same as demonstration reactor. |
| Technical target | Solid Blanket System :<br>• Mass production development of tritium breeder pebbles (large size pebble : $\phi$1~2mm, small size pebble : $\phi$0.1~0.2mm) (Amount of tritium breeders : about 50 tons)<br>• Good tritium breeding and release behaviors of tritium breeders (Enrichment of $^6Li$ : ~90% (in the case of $Li_2ZrO_3$ and $Li_2TiO_3$)) (Tritium inventory : <100g)<br>• Stability of tritium breeder at ~5% Li burn-up and 400 ~1000°C (No crack, swelling, etc.). | Solid Blanket System :<br>• Development of the mass-product method for pebble of tritium breeders (large size pebble : $\phi$1~2mm, small size pebble : $\phi$0.1~0.2mm) (Amount of tritium breeders : about 100 tons)<br>• Good tritium breeding and release behaviors of tritium breeders (Enrichment of $^6Li$ : ~90% (in the case of $Li_2ZrO_3$ and $Li_2TiO_3$)) (Tritium Inventory : <100g)<br>• Stability of tritium breeder up to 20% Li burn-up and 400~1000°C (No crack, swelling, etc.)<br>Liquid blanket :<br>• Liquid lithium<br>Reduction of tritium inventory, purity controls in liquid lithium and decrease of MHD pressure loss.<br>• Lithium-lead alloy<br>Development of coating materials for insulation, prevention at tritium penetration and anticorrosion.<br>• Molten salt (FLiBe, etc.)<br>Development of coating materials for insulation, prevention at tritium penetration and anticorrosion. | The same as demonstration reactor. |

# Tritium breeder materials

| | | | |
|---|---|---|---|
| Status | Solid Blanket System :<br>• Pebble fabrication by wet process with substitution reaction (150kg/year).<br>• Evaluation of tritium release behavior from tritium breeding materials in low-neutron irradiation.<br>• Stability of $Li_2O$ and $Li_2ZrO_3$ irradiated at the condition of ~5% Li burn-up by BEATRIX-II. | Solid Blanket System :<br>• Development of pebble fabrication by wet process.<br>• Development of improved materials with good tritium release behavior (start of the grain size control tests by the addition of other material).<br>• High neutron irradiation test by fission reactor at international cooperation.<br>Liquid Blanket System :<br>• Liquid lithium<br>1) Compatibility tests between liquid lithium and another materials by heat flow loop test at 200 Liters.<br>2) Fabrication and evaluation of insulation coating for the reduction of MHD pressure.<br>• Lithium-lead alloy<br>1) Compatibility tests between liquid lithium and another materials.<br>• Molten salt (FLiBe, etc.)<br>1) Compatibility tests between liquid lithium and another materials by heat flow loop test. | The same as demonstration reactor. |

# Tritium breeder materials

| | | | |
|---|---|---|---|
| Subjects | Solid Blanket System :<br>• Development of the fabrication method for tritium breeder pebbles | Solid Blanket System :<br>• Development of pebble fabrication by wet process.<br>• Development of improved materials.<br>• Stability evaluation of neutron multiplier at high-neutron irradiation and at neutron irradiation by neutron spectrum of fusion reactor (including the determination of the material specification).<br>• Development of $^{6}$Li enrichment technique and recycle technique.<br>Liquid Blanket System:<br>• Common research and development of heat flow behavior, chemical compatibility of liquid lithium, tritium recovery behavior, etc. by IFMIF<br>• Start of common research and development on liquid lithium and molten salt (consideration of the application for advanced divertor and first wall).<br>• Development of irradiation capsule for irradiation tests with liquid metals and molten salt. | The same as demonstration reactor. |
| The concrete solution | Solid Blanket System :<br>• Fabrication development of large-size pebbles by wet process with dehydration reaction. | • Same as the experiment reactor.<br>• Characterization of improved materials.<br>• High-neutron irradiation tests with IFMIF by international cooperation.<br>• Development of lithium isotope separation with lithium ionic conductor. | The same as demonstration reactor. |
| The prospect | Achievement possible | Prospect of achievement at about 2010. | Prospect of achievement |

# Neutron multiplier materials

| Items | Experimental reactor (ITER) | Demonstration reactor (DEMO) | Commercial reactor |
|---|---|---|---|
| Li burn-up | ~5% | 5~20% (replace per 2 years) | 5~10% (replace per 2 years) |
| Nuclear heating ($MW/m^3$) | ~50 | ~150 | ~150 |
| Temperature (°C) | 200~400* | 400~1000* | 600~1000* |
| Environment | He (~0.1MPa) | He (~0.1MPa) | He (0.1~10MPa) |

* : These values are example for $Li_2O$ and are difference with tritium breeder materials.

| Items | Experimental reactor (ITER) | Demonstration reactor (DEMO) | Commercial reactor |
|---|---|---|---|
| He generation ratio (appmHe) | ~3000 | ~20000 (replace per 2 years) | ~20000 (replace per 2 years) |
| Nuclear heating ($MW/m^3$) | ~10 | ~30 | ~30 |
| Temperature (°C) | 150~350 | 400~900 | 600~900 |
| Environment | He (~0.1MPa) | He (~0.1MPa) | He (0.1~10MPa) |

# Neutron multiplier materials

| Items | Experimental reactor (ITER) | Demonstration reactor | Commercial reactor |
| --- | --- | --- | --- |
| Neutron Multiplier | Material : Be<br>Shape : Pebble | Material : Be or advanced material (beryllium intermetallic compounds)<br>Shape : Pebble | The same as demonstration reactor. |
| Technical target | • Mass production development of neutron multiplier pebbles (large size pebble : $\phi$1~2mm, small size pebble : $\phi$0.1~0.2mm) (Amount of tritium breeders : about 150 tons)<br>• Stability of neutron multiplier at ~3000appmHe, ~3dpa and 150~350°C (No crack, swelling, etc.). | • Mass production development of neutron multiplier pebbles (large size pebble : $\phi$1~2mm, small size pebble : $\phi$0.1~0.2mm) (Amount of tritium breeders : about 300 tons)<br>• Stability of neutron multiplier at ~20000appmHe, ~20dpa and 400~900°C (No crack, swelling, etc.). | The same as demonstration reactor. |
| Status | • Pebble fabrication by the rotating electrode method (120kg/year).<br>• Stability of neutron multiplier irradiated at the condition of ~3000appmHe and ~30dpa. | • Development of pebble fabrication by the rotating electrode method.<br>• Development of beryllium intermetallic compounds.<br>• High neutron irradiation tests by fission reactor at international cooperation. | The same as demonstration reactor. |
| Subjects | none | • Development of beryllium intermetallic compounds.<br>• Stability evaluation of neutron multiplier at high-neutron irradiation and at neutron irradiation by neutron spectrum of fusion reactor (including the determination of the material specification).<br>• Development of beryllium reprocessing technology. | The same as demonstration reactor. |
| The concrete solution | | • Fabrication development of beryllium intermetallic compounds by the rotating electrode method.<br>• High-neutron irradiation tests with fission reactor and IFMIF by international cooperation.<br>• Developments of beryllium reprocessing with dry process. | The same as demonstration reactor. |
| Prospect | Achievement possible | Prospect of achievement at about 2010. | Prospect of achievement |