Recent Progress in Fusion Technologies under the BA DEMO-R&D in Phase1 in Japan (FTR/3-2Ra)

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Tritium Recovery Experiment from Li Ceramic Breeding Material Irradiated with DT Neutrons (FTR/3-2Rb)


1): Japan Atomic Energy Agency, Tokai, Ibaraki, JAPAN,
## 1. DEMO R&D Subjects and schedule of JA

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### Structure materials
- **2007:** Test melting and characterization
- **2008:** Large scale melting. Development of joining and inspection methods
- **2009:** Irradiation effects on mechanical properties
- **2010:** Preparation of material and test equipments

### SiC/ SiC composites
- **2007:** Preparation of material and test equipments
- **2008:** Mechanical properties
- **2009:** Irradiation effect on physical properties

### Tritium technology
- **2007:** Preparation of Tritium and RI handling facility
- **2008-2017:** Tritium Accountancy, Basic Tritium Safety data, Tritium Durability

### Neutron multiplier
- **2007-2017:** Preparation of fabrication and test equipment

### Tritium breeder
- **2007-2017:** Preparation of fabrication and test equipment

### Comments
- The table outlines the subjects and schedule of R&D activities for the DEMO project in Japan, focusing on key areas such as structure materials, SiC/SiC composites, tritium technology, neutron multiplier, and tritium breeder.
- Each phase is detailed with specific activities planned for each year from 2007 to 2017.

### Key Technologies
- **Structure materials (Reduced activation ferritic/martensitic)**
- **SiC/SiC composites**
- **Tritium technology**
- **Neutron multiplier (Beryllium compounds)**
- **Tritium breeder (Lithium compounds)**
2. RI handling equipment at Rokkasho site

<table>
<thead>
<tr>
<th>Group</th>
<th>A</th>
<th>B</th>
<th>C</th>
<th>D</th>
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<tbody>
<tr>
<td>RI</td>
<td>Tritium</td>
<td>Ceramics</td>
<td>Steel</td>
<td>Other metals</td>
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<tr>
<td>Typical RI</td>
<td>H-3</td>
<td>P-32</td>
<td>Fe-59, Cr-51, Ta-182</td>
<td>W-188, Re-188</td>
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<tr>
<td>Usage</td>
<td>GBq/day</td>
<td>MBq/day</td>
<td>MBq/day</td>
<td>MBq/day</td>
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<td>3700</td>
<td>370GBq/hood</td>
<td>100</td>
<td>61</td>
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<td></td>
<td></td>
<td>950</td>
<td>46</td>
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<tr>
<td>Storage</td>
<td>7400 GBq</td>
<td>500 MBq</td>
<td>915 MBq</td>
<td>220 MBq</td>
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<td>14 GBq</td>
<td>220 MBq</td>
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<td>690 MBq</td>
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</table>

*: In this area, tritium and some RI species can be handled (Group A, B, and D).

**: In this area, a hot cell and a temperature controlled section are installed. The RI species of Group B, C, and D, and a small amount of tritium are handled.
2. RI handling equipment at Rokkasho site

GB = continuously detritiation, negligible tritium permeation.
Permation and leakage of the devices in hoods and detritiation systems
Distance between workers and RI = 50 cm
Concentration of tritium in room = \(\sim 5 \times 10^{-4}\) Bq/cm\(^3\)(\(\sim 1/1000\) of regulation).
Concentration of tritium at a stack = \(\sim 4 \times 10^{-4}\) Bq/cm\(^3\)(\(\sim 1/10\) of regulation).
0.783 mSv/week for workers (\(\sim 0.8\) of regulation); 1.2 mSv/3 months at a boundary of
the radiation controlled cold area (\(\sim 0.9\) of regulation), and 43 \(\mu\)Sv/3 months at a site
boundary (\(\sim 0.2\) of regulation).

3. Recent Progress in Phase1 in Japan

3.1 SiCf/SiC Composites
Failure behavior of SiC/SiC composites by various failure modes such as tensile, compressive, and shear modes was studied.
Basic data for Radiation-induced conductivity (RIC) and Radiation-induced electrical degradation (RIED) irradiation at room-temperature in air was obtained (See Figure).
3. Recent Progress in Phase 1 in Japan

3.2 Structure Materials
Various properties of the plates of the F82H steel melted and forged were studied to optimize the fabrication technology of F82H. Assessment on the specific welding technologies for DEMO was carried out to obtain basic data on the joining of F82H.

3.3 Advanced Neutron Multiplier
Preliminary tests using mixed beryllium and titanium (Be, Ti) powder by the plasma sintering method. The formation of Be-Ti intermetallics (Be$_{12}$Ti, Be$_{17}$Ti$_2$ and Be$_2$Ti) was observed for the sintering temperature of 1073~1273K. Elemental Be and Ti decreased with increase in sintering temperature.

3.4 Tritium Technology
Tritium analysis technology = imaging plate method; Basic tritium safety data = behaviors in tungsten, stainless steel etc. Tritium durability test = organic compounds durability tests by gamma and beta ray irradiation.

![Graph showing ionic conductivity of the Nafion membrane under different gamma and beta ray irradiations.]
3.5 Advanced Tritium Breeders
A rotating granulation method for making \( \text{Li}_2\text{TiO}_3 \) pebbles from powder of 5 μm sintered at 1200° C. Successful production of pebbles with a sphericity of 1.04 (See figure). Reprocessing tests: Solvent = Mixture of peroxide hydrogen (H\(_2\)O\(_2\)) and nitric acid (HNO\(_3\)) ; Dissolution rates of Li from lithium ceramic powder = 90%.

Summary
(1) The RI handling equipment has been constructed at Rokkasho site as the first and quite unique facility in Japan, where tritium (3.7 TBq/day), beta and gamma RI species, and beryllium (Be) can simultaneously be used. The tritium, Be, and other RI handling equipment, such as hoods have been designed, made, and installed.

(2) As the BA activities, the studies on structural materials (F82H); SiC/SiC: advanced tritium breeders (production of pebbles of \( \text{Li}_2\text{TiO}_3 \)), neutron multiplier (production of beryllides from Be and Ti); and tritium (durability) are carried out.
R&D on blanket technology at JAEA

DEMOnstration blanket

R&D on structural materials (SiC, structure materials)

R&D on tritium breeder (Li compound fabrication, characteristics)

R&D on tritium technology (analysis, material interaction, durability)

R&D on neutron multiplier (pebble fabrication, characteristics)

Integrated test for tritium production (multiplier, breeder, tritium, neutron irradiation)

BA activities
1. Background and Objectives

- It is an essential issue to verify the nuclear performance concerning tritium breeding in fusion blanket including ITER-TBM.
- Japan and EU have progressed to verify the tritium production ratio in simulated blanket assemblies with DT neutron irradiation. Fusion Neutronics Source (FNS) facility in JAEA has verified tritium production rate of Li\textsubscript{2}TiO\textsubscript{3}/Beryllium type.
- However, no DT neutron irradiation experiment for the tritium recovery properties of the solid blanket has been conducted and the tritium recovery ratio is one of urgent technical issues for the development of the fusion breeding blanket system.
- Based on the above DT neutron irradiation technology, JAEA-FNS has conducted the first tritium recovery experiment for the solid breeding blanket as a next step.
Tritium Recovery Experiment from Li Ceramic Breeding Material Irradiated with DT Neutrons

2. Experimental setup

The Tritium recovery ratio of the ceramic breeding blanket was measured with the 14 MeV DT neutrons for the first time in the world.

DT neutron irradiation arrangement (JAEA-FNS)

Number of DT neutrons = 10^{15}

measured by 3.5 Mev alpha with 2% error
Tritium Recovery Experiment from Li Ceramic Breeding Material Irradiated with DT Neutrons

3. Measurement of amount of tritium produced and recovered

The tritium recovery ratio of $1.05 \pm 0.08$ was obtained at 873 K, which indicates that the design of Japanese solid breeder blanket promises a good prospect of tritium recovery at 873 K.
4. Conclusion

We have performed the first tritium recover experiment with DT neutrons in the world at FNS in JAEA. From the measured tritium recovery ratio, a good prospective is obtained for the solid breeder blanket at 873 K.

In order to progress the investigation of tritium recovery property for the ITER test blanket module and DEMO blanket designs, we are going to examine the dependency of the temperature and sweep gas with DT neutron source as the next step.