Conceptual core design study for Japan sodium-cooled fast reactor: review of sodium void reactivity worth evaluation

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2. Design requirements and conditions
3. Core design results
   (featuring the characteristics of sodium void reactivity worth)
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Introduction

FBR development program in Japan

FS (1999-2005)
Feasibility Study on Commercialized Fast Reactor Cycle System
- Clarification of several promising candidates for FBR cycle system

FaCT Project (2006-2015)
Fast Reactor Cycle Technology Development Project
- Establishment of the most prominent FBR cycle system technologies

Suspending since Fukushima-daiiichi accident

The reference concept

◆ The reference core concept:
  - JSFR MOX fuel core
    “High internal conversion” type
◆ TRU recycling mode:
  Homogeneous

JSFR: Japan Sodium-cooled Fast Reactor
- 1500 MWe for commercial use
- 750 MWe for demonstration

Joyo
Monju
(280 MWe)

Fuel Fabrication

- High burnup and long operation period
- Passive safety & recriticality free

Fuels with TRU

No Pure Plutonium

Fast Reactor

- Sustainable usage of nuclear energy
- Reduce the environmental burden

Reduction of Radiotoxicity

Reprocessing

U/TRU mixed product

Reduction of Waste

Homogeneous TRU Recycling

Geological Disposal
After 2030: 58 GWe (constant)

After 2050: replacing 1-GWe LWR with FBR per year

Typical scenario for FBR deployment in Japan

Nuclear Installed Capacity [GWe]

- Existing LWR
- Advanced LWR
- MOX fuel utilization in LWR
- FBR (Breeding core)
- FBR (Break-even core)

[Graph showing the integration of FBR deployment in Japan over the years, with a focus on replacing LWR plants with FBRs after 2050.]
Design requirements and conditions
## Safety requirements for JSFR core design

<table>
<thead>
<tr>
<th>Event</th>
<th>Design basis accidents (DBA)</th>
<th>Anticipated transient without scram (ATWS)</th>
<th>Core disruptive accidents (CDA)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Two active shutdown systems:</td>
<td>Self-actuated shutdown system (SASS) on BCR</td>
<td></td>
</tr>
<tr>
<td>Measure</td>
<td>- Primary control rods (PCR)</td>
<td></td>
<td>- Core height: &lt; 100 cm</td>
</tr>
<tr>
<td></td>
<td>- Backup control rods (BCR)</td>
<td></td>
<td>- Sodium void reactivity worth: &lt; about 6 $</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- Average core specific heat: &gt; 40 kW/kg-MOX</td>
</tr>
<tr>
<td></td>
<td>Fuel assembly with inner duct structure (FAIDUS) for recriticality-free</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Design requirements to prevent super-prompt criticality in the initiating phase of CDA
Fuel Assembly with Inner Duct Structure (FAIDUS)

- FAIDUS has inner duct installed at a corner, and a part of upper shielding element is removed.

- At CDA (Core Disruptive Accident), molten fuel enters the inner duct channel and goes out into the outside through the upper shielding.

FAIDUS has superior performance for discharge of molten fuel to prevent recriticality.
Other design requirements and conditions (1/2)

◆ **Sustainability** *(waste management, efficient utilization of nuclear fuel resources)*
  - MA contents in the fuel: from 1 to 5 wt%
  - Breeding ratio: 1.03~1.1 (for low breeding core)
    1.2 (for high breeding core)

◆ **Economic Competitiveness**
  - Operation period: more than 18 months
  - Average discharge burnup
    - for driver fuel: 150 GWd/t
    - for whole core including blanket:
      80 GWd/t (for low breeding core)
      60 GWd/t (for high breeding core)
Other design requirements and conditions (2/2)

- **Plant conditions**
  - Power output: 1500 MW\(_e\) / 3530 MW\(_t\) (commercial core)
  - 750 MW\(_e\) / 1765 MW\(_t\) (demonstration core)
  - Coolant temperature (outlet / inlet): 550 °C / 395 °C
  - Primary coolant flow rate: 18,000 kg/s (commercial core)
  - 9,000 kg/s (demonstration core)
  - Shielding region diameter: less than about 7.0 m

- **Thermal hydraulic condition**
  - Maximum cladding mid-wall temperature: 700 °C ※
  - Bundle pressure drop: less than about 0.2 MPa

- **Fuel integrity limits**
  - Maximum linear power: less than about 430 W/cm
  - CDF (steady state): less than 0.5
  - Maximum fast neutron fluence (E>0.1 MeV): less than about 5 × 10\(^{23}\) n/cm\(^2\) ※

※ Cladding material: ODS steel
Flow of core design and evaluation procedure

Plant condition
- Power output

Core performance targets
- Breeding ratio
- Core average discharge burnup

Fuel composition

Core and fuel specifications
- Core height
- Operation cycle length
- Fuel exchange batch
- Pin diameter
- Pin pitch
- Core configuration
- Axial blanket thickness

Make adjustments if necessary

Core Neutronic Evaluation
- Criticality and burnup calculations
- Control rod worth evaluation
- Reactivity coefficient evaluation

<Outputs>
- Pu enrichment
- Total average discharge burnup
- Sodium void reactivity
- Core average specific heat
- Doppler coefficient
- Control reactivity balance

Thermal-Hydraulic Evaluation
- Arrangement of coolant flow distribution
- Subassembly temperature evaluation (Sub-channel analysis)

<Outputs>
- Power distribution and history
- Hot spot factors

Fuel Integrity Evaluation
- Evaluation of cumulative damage fraction (CDF)

<Output>
- Maximum cladding CDF

<Outputs>
- Maximum linear heat rate
- Maximum fast neutron fluence

Plant condition
- Coolant outlet and inlet temperatures

<Outputs>
- Maximum cladding mid-wall temperature
- Maximum bundle pressure drop
- Cladding temperature history
- Necessary coolant flow rate
Core design results
**Layout of commercial core (1500 MWe)**

Breeding ratio: 1.03 ~ 1.1  
(High breeding option: 1.2)

- **Economical advantages**
  - Breeding ratio: 1.03 ~ 1.1  
  - Increasing fuel volume fraction  
  - Increasing internal conversion rate  
  - Reducing the amount of blanket  
  - Increasing total average discharge burnup (including blanket) (84-115 GWd/t)  
  - Long operation cycle length (18-26 month)

### Typical core neutronic characteristics (1/2)

<table>
<thead>
<tr>
<th>Item</th>
<th>Demonstration core 750 MWe</th>
<th>Commercial core 1500 MWe</th>
</tr>
</thead>
<tbody>
<tr>
<td>TRU Composition</td>
<td>FBR multi-recycle (MA: 1 wt%)</td>
<td>FBR multi-recycle (MA: 1 wt%)</td>
</tr>
<tr>
<td>Core height [cm]</td>
<td>100</td>
<td>100</td>
</tr>
<tr>
<td>Number of fuel assembly (IC / OC)</td>
<td>157 / 117</td>
<td>288 / 274</td>
</tr>
<tr>
<td>Axial blanket thickness (upper / lower) [cm]</td>
<td>20 / 25</td>
<td>20 / 20</td>
</tr>
<tr>
<td>Operation cycle length [month]</td>
<td>18 (6 batch)</td>
<td>26 (4 batch)</td>
</tr>
<tr>
<td>Pu enrichment (IC / OC) [wt%]</td>
<td>18 / 25</td>
<td>18 / 21</td>
</tr>
<tr>
<td>Burnup reactivity [%dk/kk’]</td>
<td>1.9</td>
<td>2.5</td>
</tr>
<tr>
<td>Breeding ratio</td>
<td>1.1</td>
<td>1.1</td>
</tr>
<tr>
<td>Sodium void reactivity (EOEC) [$]</td>
<td>4.7</td>
<td>5.2</td>
</tr>
</tbody>
</table>
Reactor size dependence of sodium void reactivity worth

![Graph showing reactor size dependence of sodium void reactivity worth.](image-url)
## Typical core neutronic characteristics (2/2)

<table>
<thead>
<tr>
<th>Item</th>
<th>Demonstration core (750 MWe)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>U-Pu High fissile</td>
</tr>
<tr>
<td>TRU Composition</td>
<td>LWR-4y spent fuel (MA: 0.2 wt%)</td>
</tr>
<tr>
<td>Operation cycle length [month]</td>
<td>18 (6 batch)</td>
</tr>
<tr>
<td>Pu enrichment (IC / OC) [wt%]</td>
<td>18 / 25</td>
</tr>
<tr>
<td>Burnup reactivity [%dk/kk’]</td>
<td>2.8</td>
</tr>
<tr>
<td>Breeding ratio</td>
<td>1.1</td>
</tr>
<tr>
<td>Sodium void reactivity (EOEC) [$]</td>
<td>4.2</td>
</tr>
<tr>
<td>Doppler coefficient (EOEC) [10^{-3} Tdk/dT]</td>
<td>- 5.6</td>
</tr>
</tbody>
</table>
Correlation of burnup reactivity and sodium void reactivity.
(○: U-Pu fuel, ●: TRU fuel)

Using the correlation between core characteristics, it is possible to define a design envelop that includes any fuel composition appears in the reactor lifetime.
Nuclide-wise sensitivity for sodium void reactivity
(750 MWe-JSFR)

Fissile (negative sensitivity)

Fertile (positive sensitivity)

Sodium void reactivity strongly depends on fissile and fertile compositions that have negative and positive sensitivities, respectively.
## Thermal-hydraulic characteristics and cladding CDF value

### Coolant flow distribution for demonstration core (750 MWe)

- **Flow Region Number**
  - Inner core: 1-5
  - Outer core: 6-10
  - Radial Blanket: 11-12

### Table: Demonstration core (750 MWe)

<table>
<thead>
<tr>
<th>Item</th>
<th>U-Pu High fissile</th>
<th>TRU Low fissile</th>
</tr>
</thead>
<tbody>
<tr>
<td>Maximum cladding mid-wall temperature [°C]</td>
<td>700.0</td>
<td>699.8</td>
</tr>
<tr>
<td>Maximum cladding CDF*1 (in a steady operation)</td>
<td>0.01</td>
<td>0.47</td>
</tr>
<tr>
<td>Necessary coolant flow rate [kg/s]</td>
<td>8594.4 (95.5%)</td>
<td></td>
</tr>
<tr>
<td>Maximum bundle pressure drop*1 [MPa]</td>
<td>0.22</td>
<td></td>
</tr>
</tbody>
</table>

*1 Assuming a lower gas plenum of 133 cm length.
Development of core neutronics design method and its Verification & Validation
Core neutronics calculation method for a license application of demonstration JSFR (1/3)

- **Nuclear data**  
  JENDL-4.0  
  (Latest Japanese evaluated nuclear data library)

- **Base calculation**
  - Cell calculation:  
    One-dimensional cylindrical representation  
    (Verification for FAIDUS is necessary.)
    Background cross section by Tone’s method  
    (cf. T. Tone, JNST 12, 467 (1975))
  - Control rod homogenization:  
    Reaction rate ratio preservation method  
    (cf. T. Kitada et al., JNST 31, 647 (1994))
  - Core calculation:  
    Three-dimensional (Tri-Z) diffusion burnup calculation  
    Benoist’s directional diffusion coefficient  
    Energy group: 70  
    (18 or 7 group is also possible.)
JENDL-4.0 benchmarking results on sodium void reactivity worth

Corrects

- Transport effect
- Mesh effect
- Energy group correction (to 70 groups)
- Ultra-fine group correction (to ~100,000 groups in order to treat resonance peaks explicitly.)

- Correction by integral experiments (e.g. bias factor correction)

  *It is necessary to evaluate a correction factor (or uncertainty) by extrapolating the mock-up experiment to JSFR core design.*

  Experimental information such as higher-mass plutonium isotopes and burnup effects are not enough. Experimental information from MONJU will play an important role.
Uncertainty evaluation: related to V&V methodology

<Currently underdevelopment>

**Verification** of design scheme (modeling and codes)
by reference calculation codes (e.g. Monte Carlo method, Method of Characteristics)

**Validation** of design scheme (nuclear data)
Experimental information are available or not.

- Full-Mockup experiment
  - Bias factor method
- Partial-Mockup or related integral experiments
  - Extrapolation
  - Cross-section adjustment
- No experimental information
  - Use of nuclear data covariance
  - Engineering judge?

Calculation model uncertainty
Experimental error
Nuclear-data induced uncertainty
Design-related uncertainties
(e.g. uncertainty due to fuel fabrication tolerance)
Summary

• The conceptual core design study for a large-scale Japan sodium-cooled fast reactor (JSFR) have been carried out in the framework of the FaCT project.

• The reference “High-internal conversion” core can satisfy the requirements for enhanced safety, as well as achieving economic competitiveness.

• In order to increase the design reliability, more rigorous uncertainty evaluation is important. Development of the verification and validation methodology of the core neutronic design method is currently underway.