Advanced non-light-water reactors in Japan
- Experience and path forward-

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Contents

• Non-LWR Advanced Reactors in Japan
  – Fast reactor, High temperature gas reactor
  – Operating experience and path forward

• Codes and Standards

• Concluding remarks
Fast Reactor
History and current status

- **FaCT Phase I**: Evaluated adoptability of innovative technologies for commercial fast reactor (FR) cycle

- **FaCT Phase 2**: Although scheduled to start in JFY2011, has not started yet due to the need for review of Japan’s nuclear energy policy after the 1F accident

- Decision of the Japanese Government on the Monju decommissioning on Dec. 21st 2016

- Now, focus on R&D for:
  - FBR/FR Safety enhancement
  - Reduction of waste volume and radio-toxicity

Plan

Under the 4th Strategic Energy Plan and the medium and long-term targets (JFY2015-2021) newly established by the government authorities,

- conduct R&Ds aiming mainly for the reduction of waste volume and radio-toxicity and safety enhancement fully utilizing international cooperation

Conducting efficient and effective R&D at existing facilities in Tokai, Oarai, Tsuruga sites
Joyo and Monju toward Commercial Reactors

Prototype fast breeder reactor power plant MONJU 280MWe (714MWt)

To confirm the performance of sodium cooled FBR power plant

Experimental fast reactor JOYO 140MWt

- To confirm the principle of sodium cooled FBR
- To establish operation, maintenance technology
- Irradiation for fuels and materials development using fast neutron field
History of Joyo

**MK-III**
- 2004.5~ Rated Power Operation
- 2003.7  First Criticality of MK-III Core

**MK-III Renovation**
- 2000.6
  - Carbide and Nitride Fuel Irradiation (Collaboration with JAERI)
  - Power-to-Melt Test (PTM)
  - Fuel Failure Simulation Test
  - High Burn-up Test (Peak Burn-up of 144 GWD/t, Collaboration with CEA France)
  - Served Mainly as an Irradiation Facility for FBR Fuel and Material

**MK-II**
- 1982.7  First Criticality of MK-II Core
  - Natural Circulation Test
  - Confirm Breeding Ratio
  - Accumulate Technical Experience Through Planning, Construction and Operation

**MK-I**
- 1981
- 1977.4  Attain First Criticality
Developments of Fuel and Cycle

- Confirmation of Breeding Ratio
  MK-I Core: $1.03 \pm 0.03$

- Completion of Nuclear Fuel Cycle (Sept. 1984)
  Reprocessed Pu from Joyo spent fuels were reused in Joyo.

- Power-to-Melt Test
  Linear Heat Rate: $\sim 700\text{W/cm}$

- Development of Advanced Fuel
<table>
<thead>
<tr>
<th>Fuel Type</th>
<th>Burn-up</th>
<th>L.H.R.</th>
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<tbody>
<tr>
<td>Nitride</td>
<td>39.4G WD/t</td>
<td>750W/cm</td>
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<tr>
<td>Oxide</td>
<td>35.4G WD/t</td>
<td>280W/cm</td>
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Plan to meet new regulatory requirements

- JAEA submitted an application for Joyo restart to comply with the new regulatory requirements on March 30th, 2017, and amendments to the application on October 26th, 2018.
- JAEA implemented the evaluation of reactor core design with 100MW, beyond design basis accidents, and post-accident measures.
- Safety review by NRA is underway.

<table>
<thead>
<tr>
<th>JFY (20XX)</th>
<th>16</th>
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<td>Reinforcement measures *</td>
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<td>Resuming operation</td>
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*- against earthquakes, tornadoes, volcanic eruptions, internal fires, internal flooding
- Dot lines mean indicated schedules may delay.
History of Monju

May 2010  Restart of SST-1

July 2010 Completion of SST-1

Sep. 2013  Research Plan for MONJU (MEXT)

Apr. 2014  Basic Energy Plan (Japan’s gov.)

Revise plant maintenance plan

Aug. 2012 Recovery of IVTM trouble
- Aug. 2010 IVTM trouble

May 2013  Ordered to suspend the preparatory activities for SST by NRA
- Nov. 2012  Identified inadequate maintenance management by NRA

Aug. 2012 Recovery of IVTM trouble

Mar. 2011  Fukushima Daiich NPS Accident

2005-2007  Plant modification to improve sodium safety

Dec. 1995  Sodium leak accident

Aug. 1995  First grid

Apr. 1994  Criticality

Achievement of Monju

A: Design, manufacture, construction

1. Core design method
2. Fuel design and fabrication method
3. Elevated temperature structural design method
4. Thin and elevated temperature structure design and manufacturing
5. Manufacture and install of large structure
6. Steam generator design and manufacturing
7. Sodium handling device design and manufacturing
8. In-service inspection device
9. Safety evaluation method

B: Evaluation of plant property based on operation data

1. Breeding ratio
2. Reactor physics property at 40% power
3. Reactor physics property related on fuel burnup
4. Reactor physics property of high Am content fuel
5. Fuel integrity after high burnup
6. Irradiation of new fuel material
7. Irradiation of MOX fuel containing minor actinide
8. Enhancement of core design method
9. 40% power operation (Duration: 5300h, Electricity generation: 883h)
10. Enhancement of thermal-hydraulic analysis method
11. Enhancement of standards and criteria (structure, material, nuclear data, maintenance)
12. Plant transient feature at 40% power
13. 100% power operation
14. Plant stability and reliability
15. Aging degradation of plant system
16. Fuel handling experience
17. Decay heat removal by natural circulation of Na

C: Operation and maintenance

1. Valuation of sodium quality control technique
2. Valuation of in-service inspection device
3. Improvement of plant maintenance technique
4. Update of manuals based on plant experience

D: Trouble management

1. Sodium leakage management
2. Repair technique of sodium handling system
3. Accumulation of trouble management experience
4. Enhancement of safety evaluation method based on plant data
5. Improvement of operation data based on degradation mechanism
6. Enhancement of severe accident management

E: Activity to meet new safety standard

1. Breeding ratio
2. Reactor physics property at 40% power
3. Reactor physics property related on fuel burnup
4. Reactor physics property of high Am content fuel
5. Fuel integrity after high burnup
6. Irradiation of new fuel material
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**Elevated Temperature Design Methods**

**R&D**
- Large-scale products, elevated temperature
- Safety, economy
  - Elevated temperature material properties
  - Failure mode of components

*Precedent* ASME Code Case N-47 (1979) “Class 1 Components in Elevated Temperature Service

**Design**
- Primary coolant system

ASME CC N-47

Monju design

**Codes and Standards**
- Systematic development of codes and standards for fast reactor is continued in JSME.

**Fabrication**
- Elevated temperature material tests
  - Component tests in air and sodium

Horizontal alignment of coolant piping to maintain necessary sodium level in case of failure that achieved shortened lay out

* RV
  * IHX
High Temperature Gas Reactor
History and status of Japan’s HTTR

Japan’s HTTR

Purpose
- Establishment of HTGR technologies
- Establishment of heat application technologies

Specifications
- Reactor thermal power: 30MW
- Coolant: Helium gas
- Reactor inlet temperature: 395°C
- Reactor outlet temperature: 850°C, 950°C
- Core material: Graphite
- Fuel: UO₂ coated particle fuel
- Uranium enrichment: 3% - 10% (Av. 6%)

Construction of reactor
- 1985: Detail design
- 1989: Application and permission of construction
- 1991: Construction
- 1997: First criticality
- 1998: Establishment of fundamental technologies
- 2001: Attainment of reactor outlet coolant temperature 850°C
- 2002: Safety demonstration test (Control rod withdrawal test)
- 2004: Attainment of reactor outlet coolant temperature 950°C
- 2007: 850°C / 30 d operation
- 2010: Loss of forced cooling test (from reactor power 30%)
- 2014: Conformity review on new regulatory requirements by NRA

First in the world

Research and development

Results of loss of forced cooling test
- Reactor is naturally shut down as soon as the core cooling flow rate become zero. Reactor is kept stable long after the loss of core cooling.

Elapsed time (h)

Test result

Maximum fuel temperature

Analysis

Research and development and design

Fuel & Material
- In-pile helium loop (OGL-1) installed at JMTR

Reactor physics
- Very High Temperature Reactor Critical assembly (VHTRC)

Thermal hydraulics
- Helium Engineering Demonstration Loop (HENDEL)
The HTTR system includes the following key components:

- **Design, construction and operational experiments** (MHI, Toshiba/IHI, Hitachi, Fuji Electric, KHI, JAEA)
  - Design optimization based on extensive technical database

- **Primary cooling system** (MHI)
  - Construction of cooling system for high temperature heat (950°C)

- **He/He intermediate heat exchanger (IHX)** (Toshiba/IHI)
  - Developed heat resistance material to enable extraction of heat up to 950°C

- **Reactor pressure vessel** (Hitachi)
  - Developed new material having high resistance to very high temperature and pressure and construct new pressure vessel using such material

- **Fuel** (Nuclear Fuel Industries)
  - Advanced technology to coat uranium fuel using ceramics with high radioactivity retaining performance

- **Reactor internals** (Fuji Electric)

- **Graphite material IG-110** (Toyo Tanso)
  - High strength
  - High heat conduction
  - Irradiation-resistance
HTGR and Heat Application Technologies Development in JAEA

(1) Reactor technology
- HTTR tests for HTGR safety enhancement
- Advanced fuel development (Pu-Burner HTGR, etc.)

(2) Heat application technologies
- Completion of basic technologies related to hydrogen production facility and gas turbine power generation
- Establishment of operation control technology and facility reliability for IS process
- Development of materials for gas turbine blade

(3) Commercial HTGR design
- Design study of commercial HTGR systems
- Core design of plutonium burning HTGR
- Establishment of safety design philosophy and international standardization
- Design of HTGR for exporting overseas

(4) HTTR – GT/H₂ test
- Coupling to HTTR
- Licensing demonstration
- Plant performance test
- Integrated demonstration of HTGR heat application system technologies
## Schedule towards Restart of the HTTR (tentative)

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<td>Re-evaluation of seismic design classification</td>
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<td>Periodic inspection</td>
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- Re-evaluation of seismic design classification: FY2015
- Seismic evaluation: FY2016, FY2017, FY2018
- Documentation of verification results, including evaluation of BDBA: FY2016, FY2017
- Evaluation by NRA: FY2017, FY2018
- Periodic inspection: Application Nov. 26

**Note:** Restart date is not specified in the schedule.
Codes and Standards – Fast reactor

Design

Fitness-for-service

Monju

New materials

60-year design

Wider variations

System Based Code

Reliability evaluation

LMR characteristics

2017～

Life-cycle optimization

1984

Monju Standard

1984

JSME Codes

1984

JSME 2012 edition

New materials

Fitness-for-serviced LBB evaluation

Reliability evaluation

Future reactors
JSME Code development – Collaboration with ASME

Needs from design

New materials
60-year design
Higher temperatures
Higher stresses
Higher efficiency
Higher safety

Major revisions

New issues: 2018-
Guidelines for reliability evaluation
LBB Guidelines
Fitness-for-service Code

System Based Code Concept

Reliability evaluation
Break Size Limitation Analysis
Inservice Inspection requirements

ASME CC N-875 (ASME-JSME Joint Task)
**ASME Code Case N-875**

**Code Case N-875, Alternative Inservice Inspection Requirements for Liquid-Metal Reactor Passive Components Section XI, Division 3**

**Inquiry:** Under what conditions may the System Based Code (SBC) be used to determine alternative examinations to Table IMB-2500-1, Examination Categories B-A, B-B, B-J-1, B-J-2, and B-N, when examining Class 1 liquid-metal-retaining components and their integral attachments in accordance with Section XI, Division 3, IMB-2500?

**Reply:** It is the opinion of the Committee that the examination methods shown in Tables 2A through 2E of this Case may be used as an alternative to the methods shown in Table IMB-2500-1, Examination Categories B-A, B-B, BJ-1, B-J-2, and B-N, provided the following requirements are met.

**Table 1: Alternative requirements to Table IMB-2500-1 EXAMINATION CATEGORIES, B-A**

### Notes
1. The system design shall consider the access requirements for performance of alternative or additional examinations to these specified items if structural defects or indications are revealed that might require such examinations.
2. Parts to components in loop type primary systems, not to licensed components in pool type primary systems.
3. If it is not the intent that all leak detectors be in service 100% of the time, the maximum percentage of leak detectors that may be out of service at any one time shall be specified in the Technical Specifications.
4. Leakage indications shall be evaluated as either confirmed or unconfirmed in accordance with the procedure determined by the owner as advance, in addition to the examination of leakages. Leakage indications shall be considered as confirmed for the purposes of leak detection system.
Concluding remarks

• Development of fast reactor cycle system will be continued following the roadmap issued in December 2018, utilizing Joyo, currently under review by the NRA for restart, and the experience of Monju and the various projects and R&Ds.

• High temperature gas reactor technology has been developed since 1960s in Japan. The NRA review of HTTR is now underway based on the new regulations. The HTTR is expected to restart in FY2019.