Monju and related activities in JAEA

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Possible Technologies to be developed in Monju

Technologies on:

1. Core and Fuel
2. Components and System
3. Sodium Handling
4. Plant Operation and Maintenance
5. Confirmation & Evaluation on Safety Function

: today’s presentation

1. Core and Fuel (1/2)

Specific features on Core and Fuel of Monju

[1] Middle-size breeding core with thermal power of 714MW

- Core equivalent diameter: 1.8m, Core height: 0.93m
- Core fuel assembly: assembly length: 4.2m, pin length: 2.8m

[2] Fuel composition with degraded plutonium and accumulated americium

- Am-241 content: around 1.5 wt% as the core average

[3] Capability of assembly-scale irradiation tests with commercial-reactor size

- **Irradiation tests of MOX fuel assembly, and degraded Pu-MOX and MA (Minor Actinide) bearing MOX in Monju**

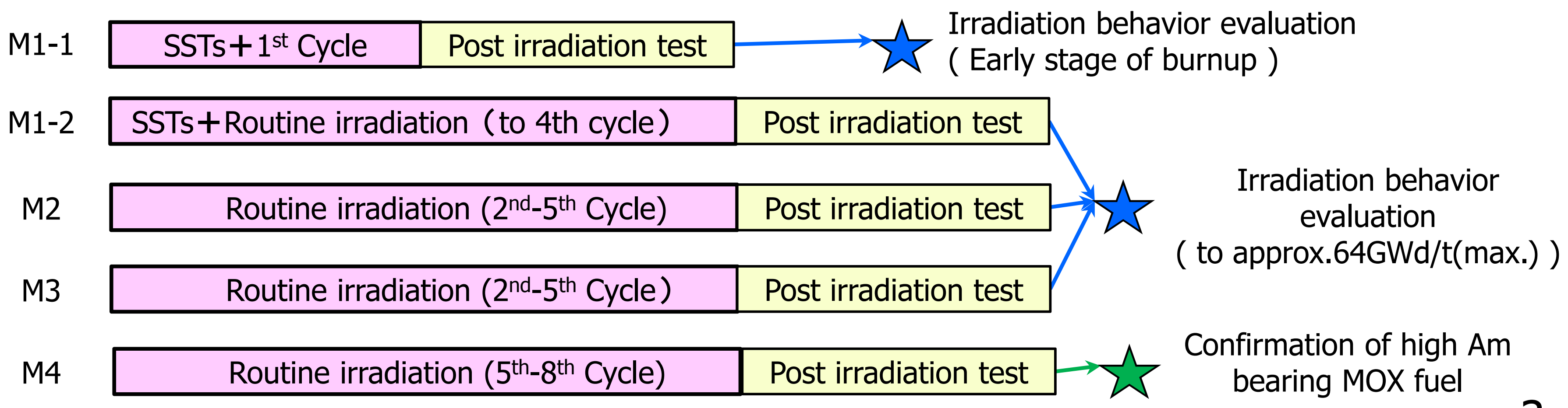
to confirm the irradiation behaviors of:

- Am-241 in industrial-scale MOX fuel assemblies (test M1)
- MOX fuel with degraded Pu (test M2)
- MA bearing MOX fuel (test M3 and M4)

1. Core and Fuel (2/2)

Irradiation tests in Monju

Test Code	Test	Achievement	Outline
M1	Irradiation test of MOX Fuel Assembly	Validation of Monju fuel design, confirmation of steady-state irradiation behavior and helium effect for Am-bearing MOX fuel.	Irradiation tests on MOX fuel assembly accumulated by Am during long term storage Low burnup and medium burnup
M2	Irradiation test of Degraded Pu MOX Fuel	Confirmation of irradiation behavior and helium effect for degraded Pu-MOX fuel.	Irradiation test on MOX fuel with degraded Pu from Fugen
M3	Irradiation test of MA-bearing MOX fuel	Confirmation of irradiation behavior for MA-bearing MOX fuel pin.	Irradiation test on fuel assembly including MOX fuel pin bearing MA
M4	Irradiation test of MA-bearing MOX fuel (GACID)	Confirmation of irradiation behavior for MA-bearing MOX fuel pin (different from M3 in MA content, fuel specifications and fuel fabrication procedure)	Process MA materials from US as MOX fuel pin in France, install it in Monju fuel assembly and implement irradiation and post irradiation tests



2. Components and System (1/3)

Specific features of Designs on Components and System of Monju

[1] System consisted of primary, secondary and water/steam system, primary and secondary system consisted of 3 loops, water/steam system consisted of 1 loop, which work each other to be controlled and harmonized
Flow rate of each system is controlled variably depending on reactor power

[2] Unique components / System structure
  a. Hot-vessel-type reactor vessel (RV)
  b. Separated helical-coil-type SGs with level control of sodium with open surface
  c. Stand-alone large-scale IHXs and primary sodium pumps connected with pipes
  d. Japan's unique simplified fuel handling system with horizontally/vertically moving fuel transfer and exchange machines

[3] Loop type reactor which advantages in terms of maintenance and quake resistance

[4] Nuclear instrumentation which is specific to the loop type reactor, tagging gas failed fuel detection and location system, and water leak detection system in SGs and sodium leak detection system which are specific to sodium cooled FBRs

2. Components and System (2/3)

- **Loop type plant system design and evaluation technology**
  - Obtaining the SSTs data on:
    - Response and control characteristics of plant control system with variable controlled flow rate depending on power, in the conditions of step power response test etc.
    - Axial temperature distribution concerning to thermal stratification of the upper plenum region in reactor vessel after emergency shutdown etc.
    - Plant response and control characteristics of sodium cooling loop type reactor, such as transients of temperature/flowrate at primary/secondary system in the conditions of emergency shutdown, load rejection etc.

  ➔Verification of evaluation method of plant dynamics in loop type reactor with SSTs data
  - Measurement of radiation dosimetry at surrounding of reactor and primary components

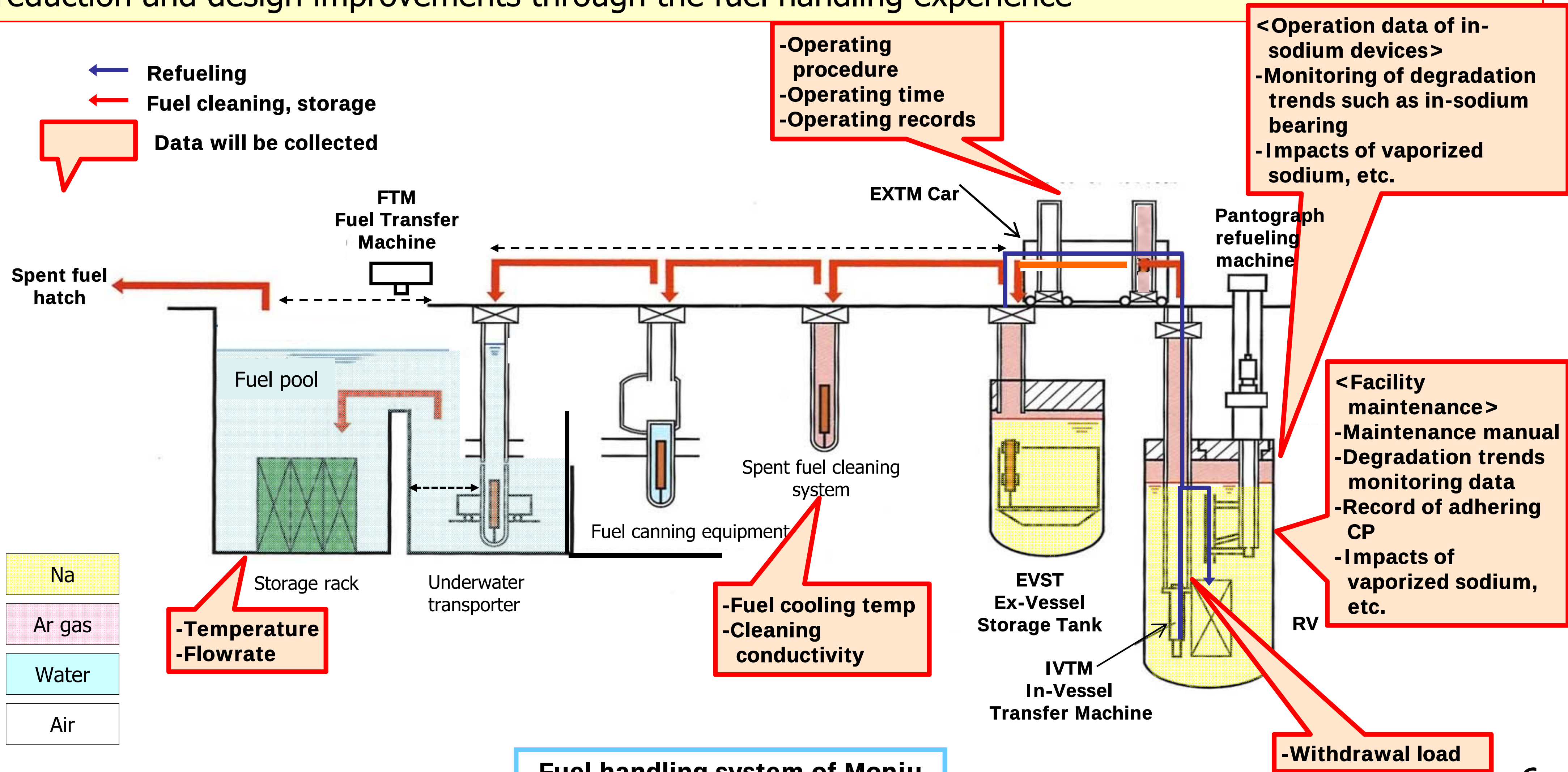
    ➔ Verification of evaluation method of radiation shielding with SSTs data

- **Fuel handling system design technology**
  - Performance verification for Japan's unique simplified fuel handling system with horizontally/vertically moving fuel transfer and exchange machines through fuel handling work including refueling, cleaning, storage, etc. (pulling load of fuels, operation time, and confirmation of sodium vapor measures, etc.)
  - System is verified in fuel handlings of initial core fuel (averaged burn-up of apx. 50GWd/t) and high burn-up fuel (apx. 80GWd/t ditto)

2. Components and System (3/3)

2) Fuel handling system

Demonstrating and validating performance of fuel handling system of Monju through fuel handling works including refueling, fuel cleaning, canning and storage, etc.
And accumulating knowledge for reliability improvement, operation know-how for working time reduction and design improvements through the fuel handling experience



Fuel handling system of Monju

5. Confirmation & Evaluation on Safety Function (1/2)

Specific features for Confirmation & Evaluation on Safety Function of Monju

[1] High ability for decay heat removal with easier sodium natural circulation than that of pool type reactor

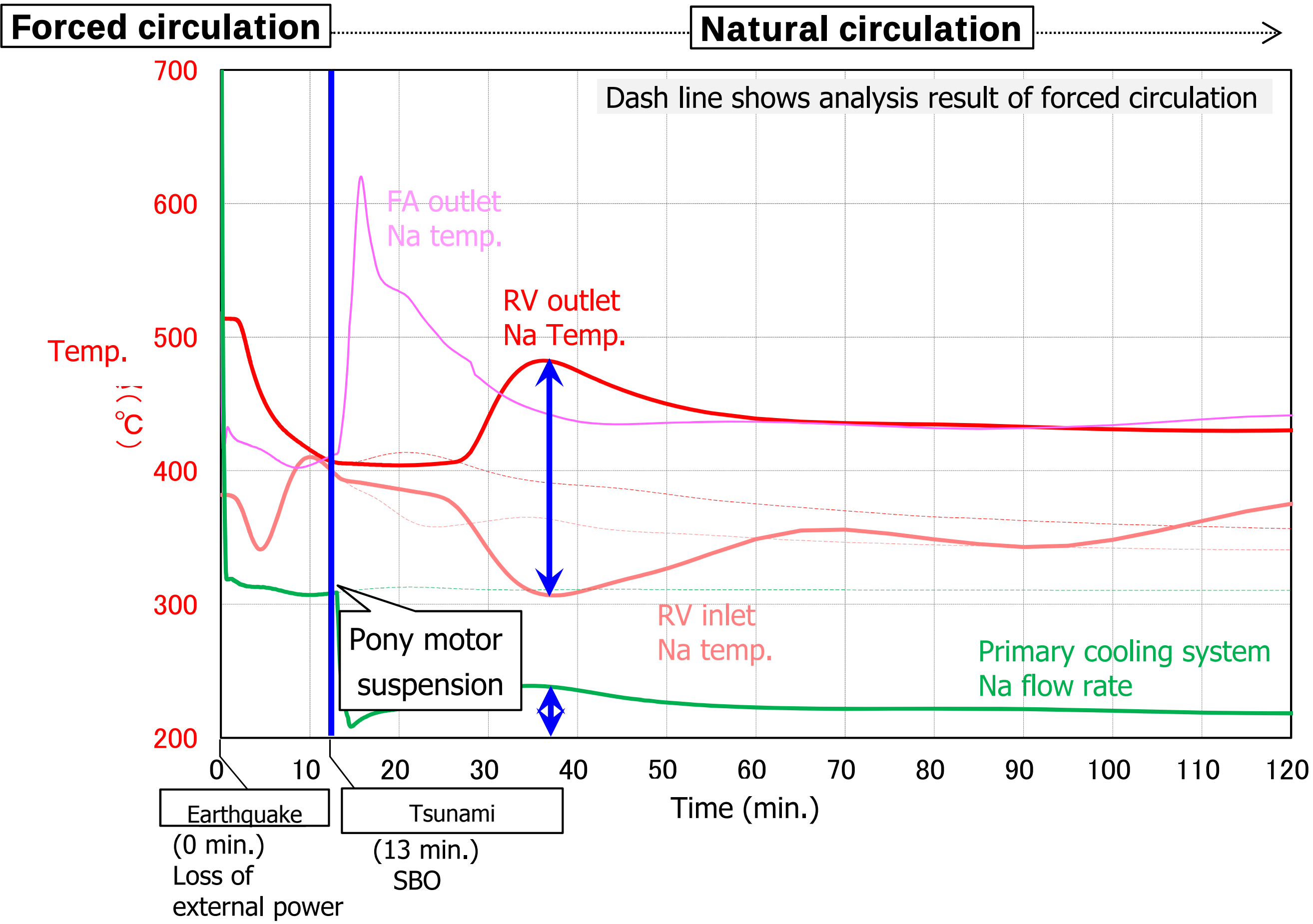
[2] Monju could or should examine and implement countermeasures against severe accidents of FBR plant after the Fukushima Daiichi accidents.

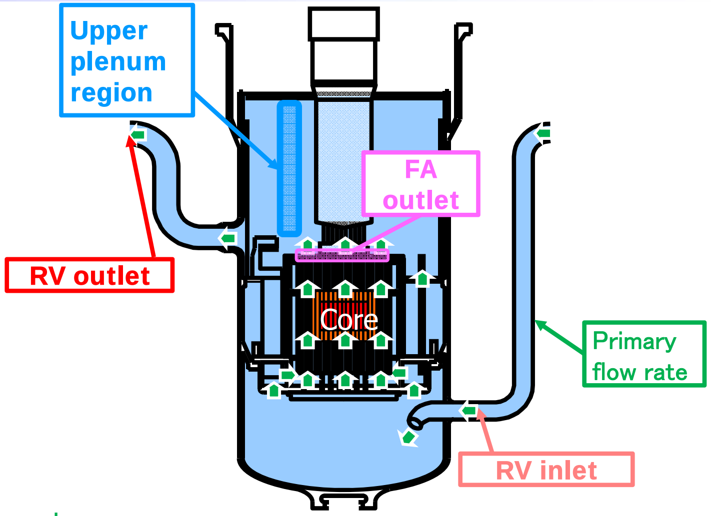
- **Verification of decay heat removal capability at FBR by natural circulation tests**
  - Demonstration of decay heat removal by high ability of natural circulation which sodium cooled reactor features
  - Verification of safety evaluation analysis code (plant dynamic characteristics analysis code) by actual plant data (SSTs data)

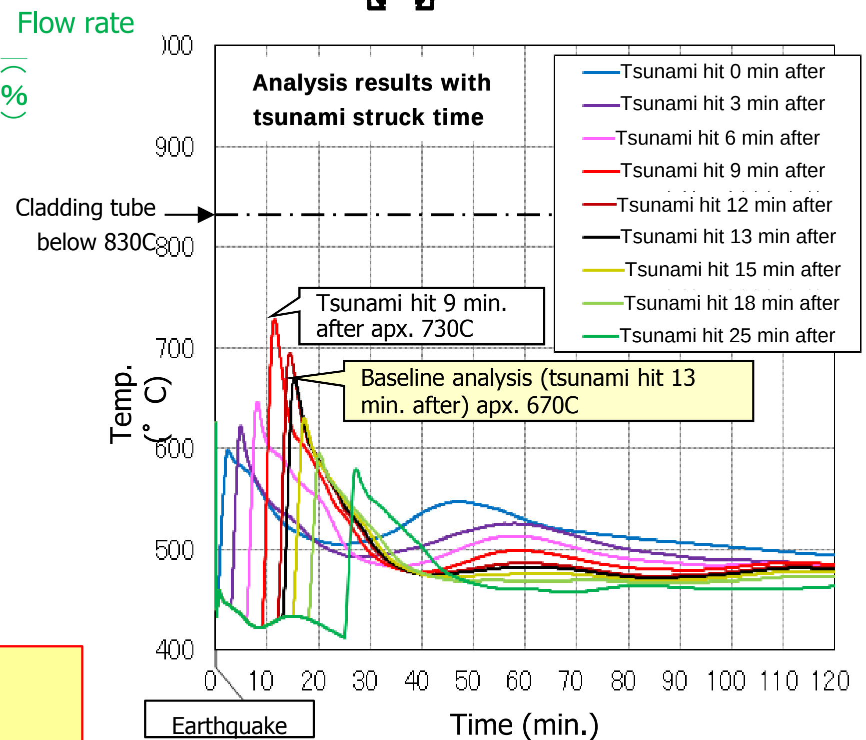
5. Confirmation & Evaluation on Safety Function (2/2)

Demonstration of decay heat removal capability by natural circulation test

Decay heat removal with natural circulation
(A result of analysis)







Cladding tube Maximum temperature transition

Natural circulation tests are planned considering several scenarios at SBO

Related R&D Facilities in JAEA

AtheNa:
Advanced Technology Experiment Sodium (Na) Facility

Experimental Plan:

- Safety Operation of Major Components in SFR
- Instrumentations
  - Under-Sodium Viewer
  - Ultrasonic Flowmeter
- Countermeasures for Severe Accident (SA)
  - International Cooperation
  - Reactor vessel model
  - Decay heat removal under severe conditions

Mother Loop:

- Sodium inventory: 240 ton
- Impurity control: < 2ppm O_2
- Design pressure: 0.9 MPa

Building:130m x 62m x 55m (Height)

	2013	2014	2015	2016	2017	2018
Cooperation in GIF	Discussion		International Research			
SA Experiments	Design and Construction			Experiments		

International collaboration

- Monju is an unique facility as an industrial scale loop-type FR in the world at present.
- Monju will be able to provide the data in the following technology fields;
  - Irradiation test data for reducing the volume and toxicity of radioactive wastes
  - Actual plant experience for the design of components and system, such as plant response and control characteristics, axial temperature distribution in the upper plenum of reactor vessel, operation/maintenance experience of simplified fuel handling system
  - Natural circulation test data for demonstration of decay heat removal by the highly facilitated natural circulation capability
- International collaboration would be implemented by multilateral and/or bilateral, and IAEA CRP framework and GIF framework would be used effectively in multilateral collaboration.
- In principle, JAEA requires counter values to its counterpart(s) for releasing the data of Monju on experiments and related design information.

Today's status of Monju

- Countermeasures as top safety priority
  - Installation of emergency power supply car, water tightening measures in seawater piping system
  - Establishment of the operating procedure with natural circulation for SBO condition
  - Carrying out the training for SBO condition, the training of connecting work to emergency power supply car etc.

Training of connecting work to emergency power supply car

- Measures to a new safety regulation standard, mainly on the severe accident on the basis of Fukushima-Daiichi severe accident
  - Parametric analysis on safety enhancement for severe accident
  - Identification of measures for safety improvements using PSA and its evaluation
- Investigation and evaluation of the crush zone underneath Monju site, and report to NRA
- Investigation of Monju research plan and preparing for 40% power stage of system start-up tests

Attachment

(Abstracts on all possible technologies to be developed in Monju)

1. Core and Fuel

1) Core design method and Core management technology

- Obtaining reactor core characteristics (dependent on power output and burnup, etc.) which has degraded Pu and bears high Am content
- Confirmation of characteristics on thermo-hydraulic dynamics of FBR reactor core
- Verification of core design method containing degraded Pu and Am by actual plant data
- Establishing core management technology as a power plant, such as refueling pattern and Pu enrichment adjustment including the reactivity management of the core with fuels bearing Pu-241 (half-life: 14 year)
- Completing the mission that I stated above, in the initial core (averaged burnup of apx. 50GWd/t) and aiming in the higher burn-up (apx. 80GWd/t, ditto)

2) Irradiation behavior of fuel assemblies for power-generating plant

- Irradiation behavior and soundness confirmation of fuels (averaged burn-up of apx. 50GWd/t) by post irradiation examination

3) Irradiation test on MOX fuel assembly, and degraded Pu-MOX and MA (Minor Actinide) bearing MOX fuel

to confirm the irradiation behaviors of:

- Am-241 in industrial-scale MOX fuel assemblies (test M1)
- MOX fuel with degraded Pu (test M2)
- MA bearing MOX fuel (test M3 and M4)

2. Components and System (1/5)

1) Plant system for loop type reactor

- Confirmation of plant performance at SSTs : plant heat balance, electricity generation efficiency, heat exchange performance, plant performance at continuous rated power operation, etc.
- Obtaining the data on:
  - Response and control characteristics of plant control system with variable controlled flow rate depending on power, in the conditions of step power response test etc.
  - Axial temperature distribution concerning to thermal stratification of the upper plenum region in reactor vessel after emergency shutdown etc.
  - Plant response and control characteristics of sodium cooling loop type reactor, such as transients of temperature/flowrate at primary/secondary system in the conditions of emergency shutdown, load rejection etc.

  → Verification of evaluation method of plant dynamics in loop type reactor with SSTs data
- Measurement of radiation dosimetry at surrounding of reactor and primary components

  → Verification of evaluation method of radiation shielding with SSTs data

2. Components and System (2/5)

2) Hot-vessel type RV, etc.

- Design verification based on actual plant data such as SSTs and rated operation
  - Reactor vessel : temperature distribution of upper plenum region of RV, temperature fluctuation of core outlet, etc.
  - Shielding plug: heat/radiation shielding performance
  - Control rod drive mechanism: drive characteristics, scram characteristics, etc.

3) Large sodium equipment for loop type

- Design verification based on the data of SSTs and rated operation for sodium pumps: operating characteristics, control characteristics, etc., and for IHXs: heat exchanging performances, etc.

4) Fuel handling system

- Performance verification for Japan's unique simplified fuel handling system with horizontally/vertically moving fuel transfer and exchange machines through fuel handling work including refueling, cleaning, storage, etc. (pulling load of fuels, operation time, and confirmation of sodium vapor measures, etc.)
- System is verified in fuel handlings of initial core fuel (averaged burn-up of apx. 50GWd/t) and high burn-up fuel (apx. 80GWd/t ditto)

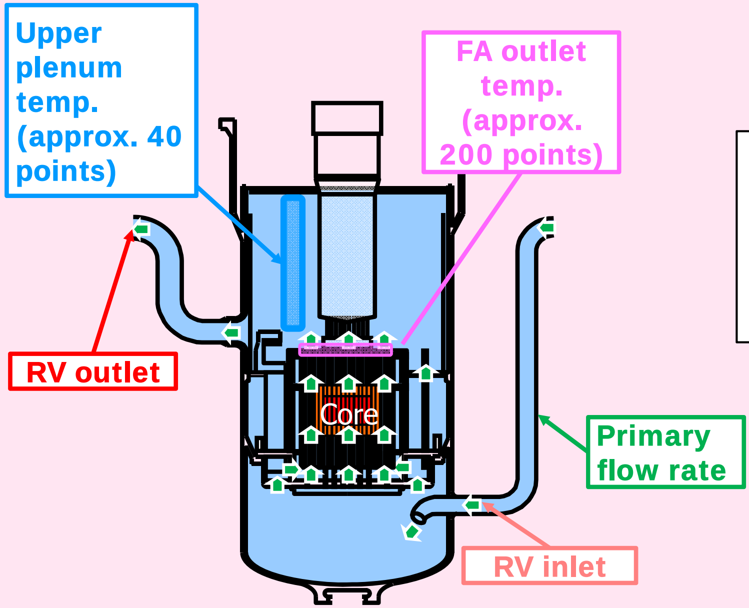
2. Components and System (3/5)

5) Instrumentation

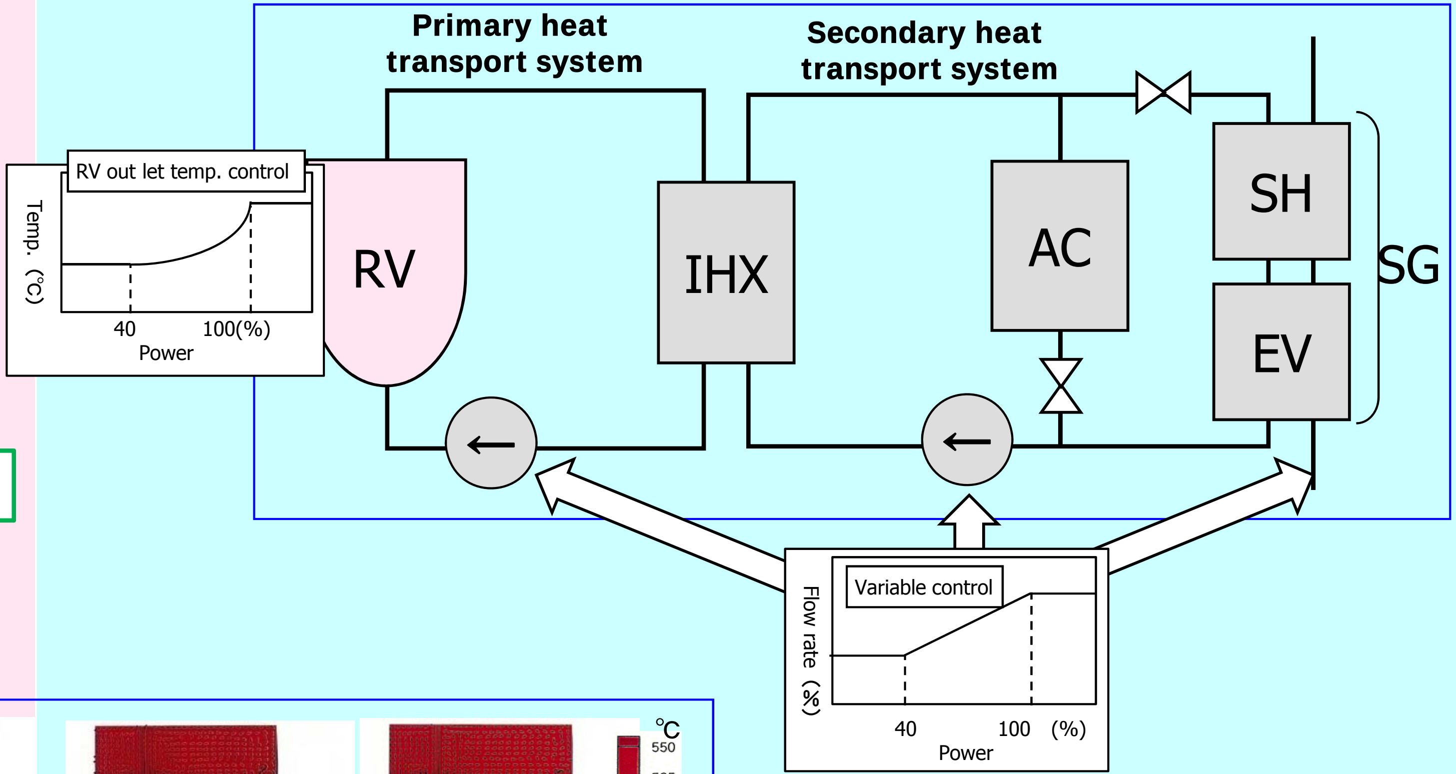
- Design verification based on the data of SSTs and rated power operation
  - Ex-core nuclear instrumentation equipment: core monitoring function
  - Failed fuel detection system by delayed neutron method: background value confirmation, etc.
  - failed fuel detection and location system using tagging gas : background value confirmation, etc.
  - Water leak detection system : background value confirmation, hydrogen transmission rate from SG, etc.
  - Sodium leak detection system
- Development of advanced instrumentation technology: ultrasonic temperature sensor

2. Components and System (4/5)

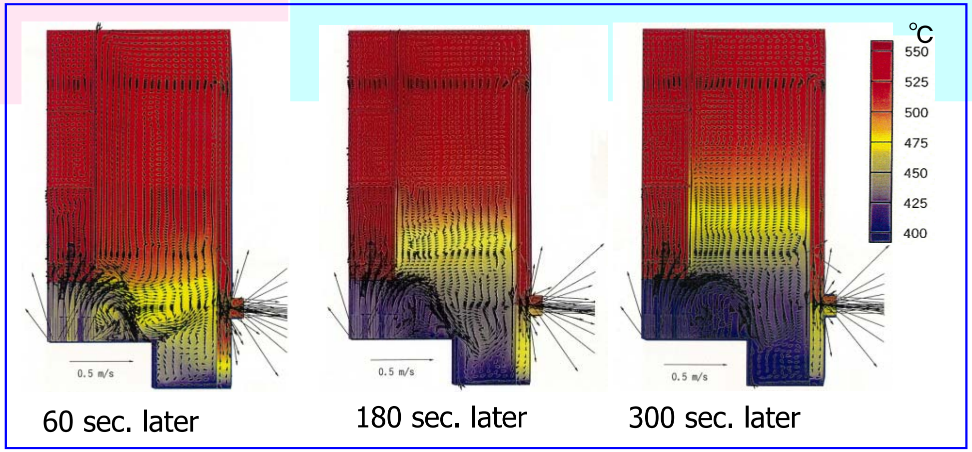
Measurement point around reactor vessel



Plant control system of Monju



<< Detailed calculation>>



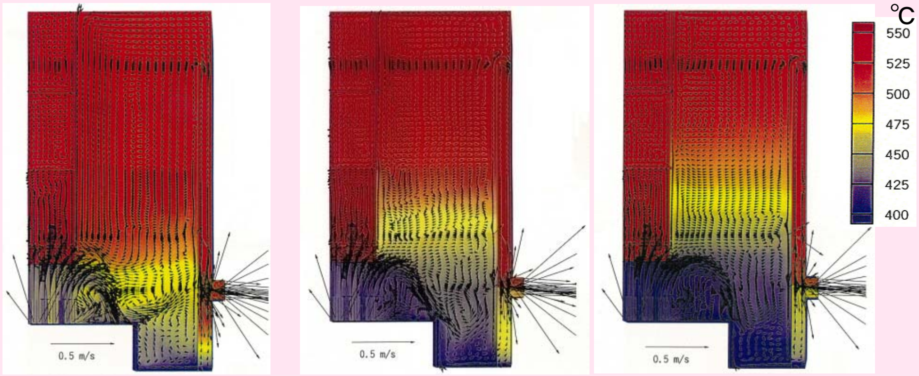
2. Components and System (5/5)

1) Loop type reactor plant system

Establish technological foundation of fast reactor with verification of evaluation code using SST data

(1) Detailed analysis of upper plenum thermal hydraulics

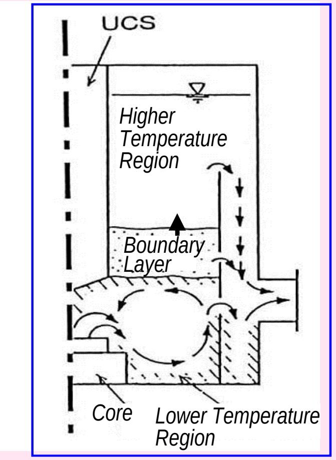
<< Detailed calculation>>

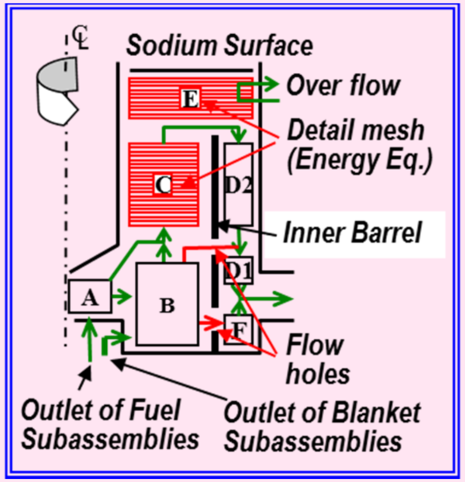


60 sec. later | 180 sec. later | 300 sec. later

Flowrate and temperature

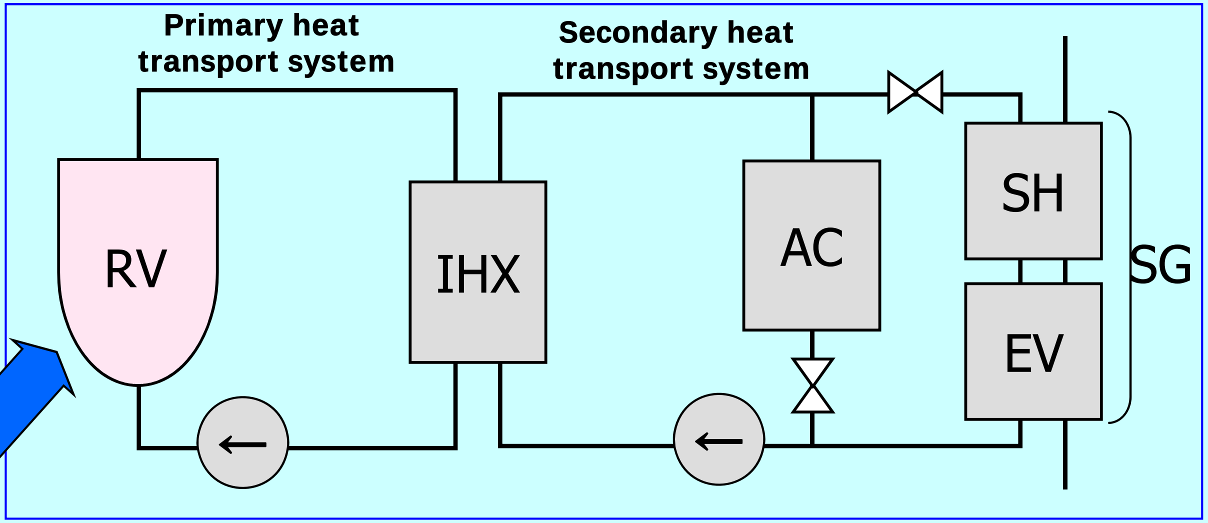
Integrate into Whole-plant Dynamics Analysis Model





(2) Construction of Flow Network Model (FNM)

<< Whole plant dynamics analysis model>>



(3) Modeling of other main components

- Construct FNM of other components (IHX, AC, SG, etc.)

(4) Whole plant dynamics simulation

- Analyze whole plant dynamics based on the FNM
- Verification by SST results
  (temperature change of inlet/outlet of major equipment, flow coast down characteristics, plant operation control characteristics)

3. Sodium Handling (1/2)

Specific features on Sodium Handling Technology of Monju

[1] In-Service-Inspection (ISI) technology for reactor vessel and primary main pipes using in high temperature/radiation environment.
ISI technology for heat transfer tubes in SGs with complex shape of helical coils made of a ferromagnetic material (low alloy steel)

[2] Data related to sodium chemistry, activated products, sodium cleaning, etc. will be available.

1) Inspection technology

- Demonstration through application to Monju on inspection technologies: ISI technology for reactor vessel, primary main pipes, SG heat transfer tubes
- Technological improvements based on actual application data: improvement of inspection performance, reduction of inspection time and costs
- Accumulation of experiences from ISI of Monju : accuracy, inspection time, cost, technological key points, etc.

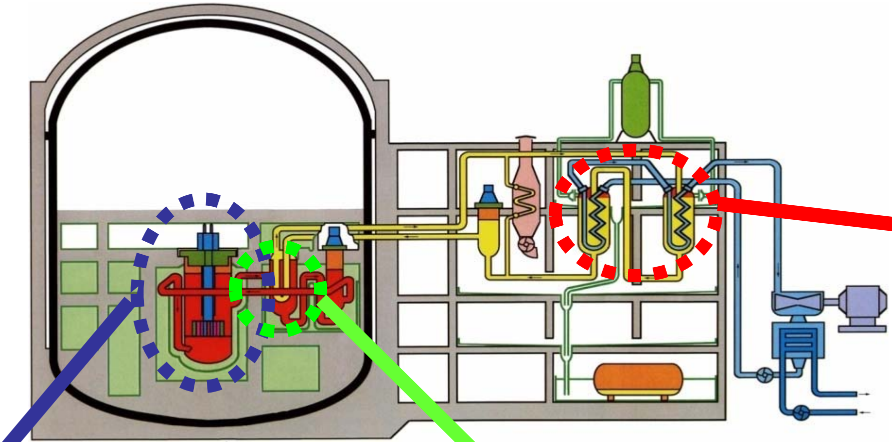
2) Sodium management technology

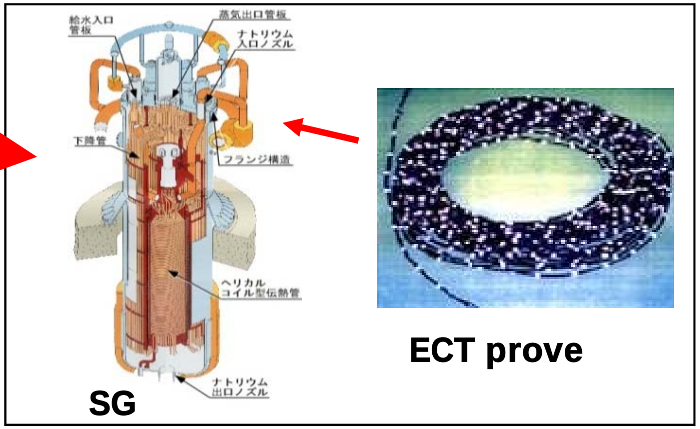
- Establishment of sodium purity management technology: cold trap design, hydrogen migration behavior, purity management standards, etc.
- Obtaining behavior characteristics data of activated products such as corrosion products (CP) which adhere to primary pipes, equipment. Validation of behavior evaluation method based on actual plant data
- Validation of sodium cleaning technology based on experience: Cleaning method of fuel handling facility and large-scale sodium equipment, remote automatic operability, waste liquid treatment including sodium and CP, etc.

3. Sodium Handling (2/2)

In-Service-Inspection (ISI) technology

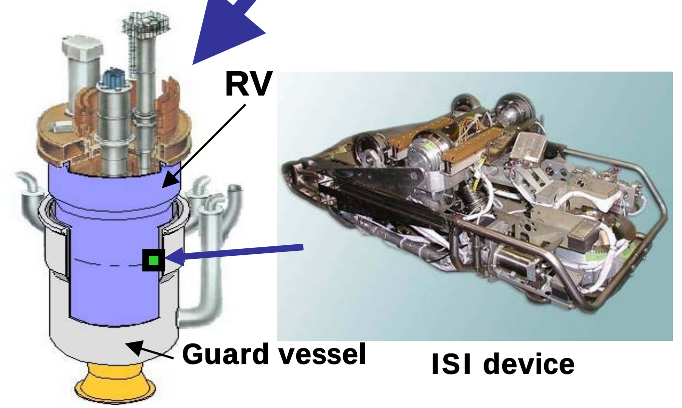
Development aiming at reliability and certainty improvements is conducted with applying inspection technology currently in development to Monju





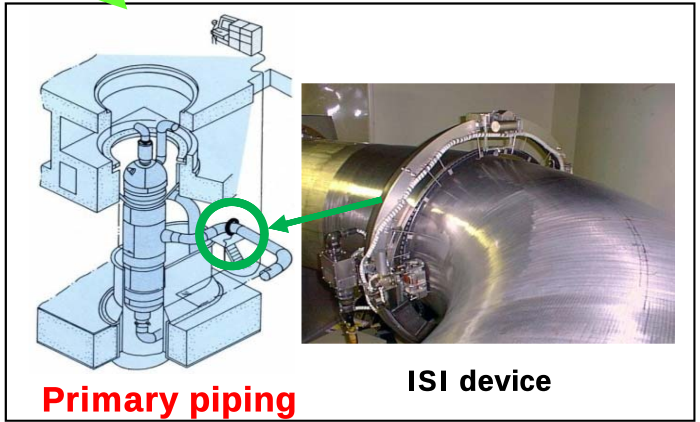
ISI device for SG tubes

- Ferromagnetic material (SG)
- Tube thickness (more than 3.5mm)
- 90 m long, workable with complex configuration heat transfer tubes



ISI device for RV

- Temperature: 200 C
- Radiation activity: 10 Sv/hr
- Self-driving in trackless narrow space (width: 30cm)



ISI device for primary piping

- Self-driving/inspection in narrow space with obstacles
- Work under high radiation field (allowable working time of set-up device: within 5 min)
- Wheel ultrasonic prove with no couplant

4. Plant Operation and Maintenance

<u>Specific features on Plant Operation and Maintenance of Monju</u>

[1] Loop type reactor which has an excellent maintainability

1) Maintenance and operation management technology

- Maintenance management technology for loop type fast reactor based on experiences of operation and maintenance of Monju:

Establishment of degradation mechanism of fast reactor equipment based on operation experience, Monju maintenance plan which inspection frequencies are optimized based on actual inspection results, examination of maintenance management for successor of fast reactor based on the Monju optimized maintenance plan, etc.

- Obtaining maintenance management data on peculiar equipment of sodium cooled reactor, which is accumulated through SSTs and rated operation:

soundness verification data, equipment failure data, etc.

- Development of preventive maintenance technology for fast reactor using actual plant
- Maintenance of operation manuals based on experience of SSTs and rated operation
- Plant diagnosis technology for fast reactor will be developed using the SSTs data

2) Accumulation of lessons learned from plant troubles and failures

5. Confirmation & Evaluation on Safety Function (1/2)

1) Severe accident evaluation technology

- Research on safety enhancement for severe accidents (SA) based on Fukushima Daiichi accidents is conducted using Monju
- Establishment of PSA evaluation technology which is peculiar to fast reactor
- Identification of measures for safety improvements using PSA and its evaluation

2) Improvement of measures for severe accident management (SAM), and its confirmation and demonstration by training and operation

- PSA including wide-ranging external events such as tsunami and earthquake is performed in Monju and establish SA evaluation technology for fast reactor
- Consideration and improvement of measures for SAM considering characteristics of sodium cooled fast reactor is carried out through training and operation

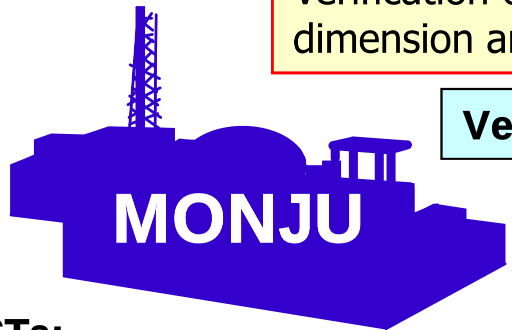
3) Verification of decay heat removal capability at FBR by natural circulation tests

- Demonstration of decay heat removal by high ability of natural circulation which sodium cooled reactor features
- Verification of safety evaluation analysis code (plant dynamic characteristics analysis code) by actual plant data (SSTs data)

5. Confirmation & Evaluation on Safety Function (2/2)

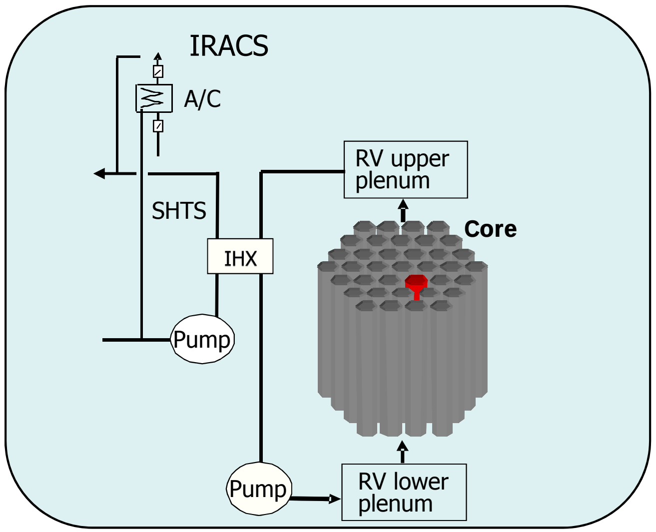
Verification of analytical method for decay heat removal by natural circulation

Verification of safety evaluation analysis code (optimized dynamics analysis model) and multiple dimension analysis is carried out using actual scale data of Monju



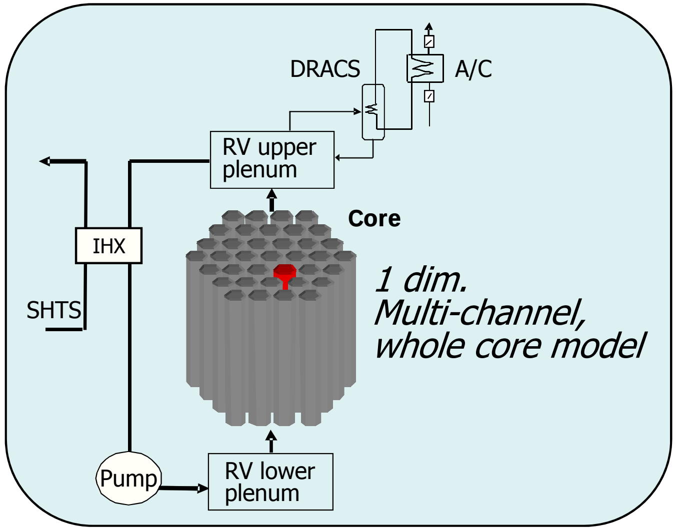
SSTs;
- Natural circulation test,
- Plant trip transient test, etc.

Verification



Natural Circulation Analysis Model of Monju

Verification of analytical method for decay heat removal by natural circulation



Natural Circulation Analysis Model of Large scale Fast Reactor

Evaluation

Establishment of safety evaluation method against severe accidents of fast reactor

 # Applicability of Monju natural circulation

- Non-dimensional number describing heat transfer;
  $Pe_{Monju} \approx (4/5) Pe_{Large\ scale\ FR}$
- Scaled experiments enable to predict natural circulation of Large scale Fast Reactor; $Pe_{W} > Pe_{Monju} > Pe_{Na}$
  (Pe_{W} : 1/10 scaled water experiment, Pe_{Na}: 1/5 scaled sodium experiment)

- Same performance level of natural circulation characteristics as large-scale reactor
- Predictable results with verified analysis code