R&D on Severe Accidents in SFR

- Approach toward GEN-IV SFR in JAEA -

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Outline of Presentation

- R&D on SFR safety and achievement
  - Study on CDA (Core Disruptive Accidents)
  - Study on Plant Safety (Natural Circulation, Chemical Reactions)
- R&D for realization of GEN-IV SFR
  - Requirement from Safety Design Criteria (SDC)
  - Gap toward Gen-IV Reactors satisfying SDC
  - Counter Measures and R&D
    - Prevention
    - Mitigation

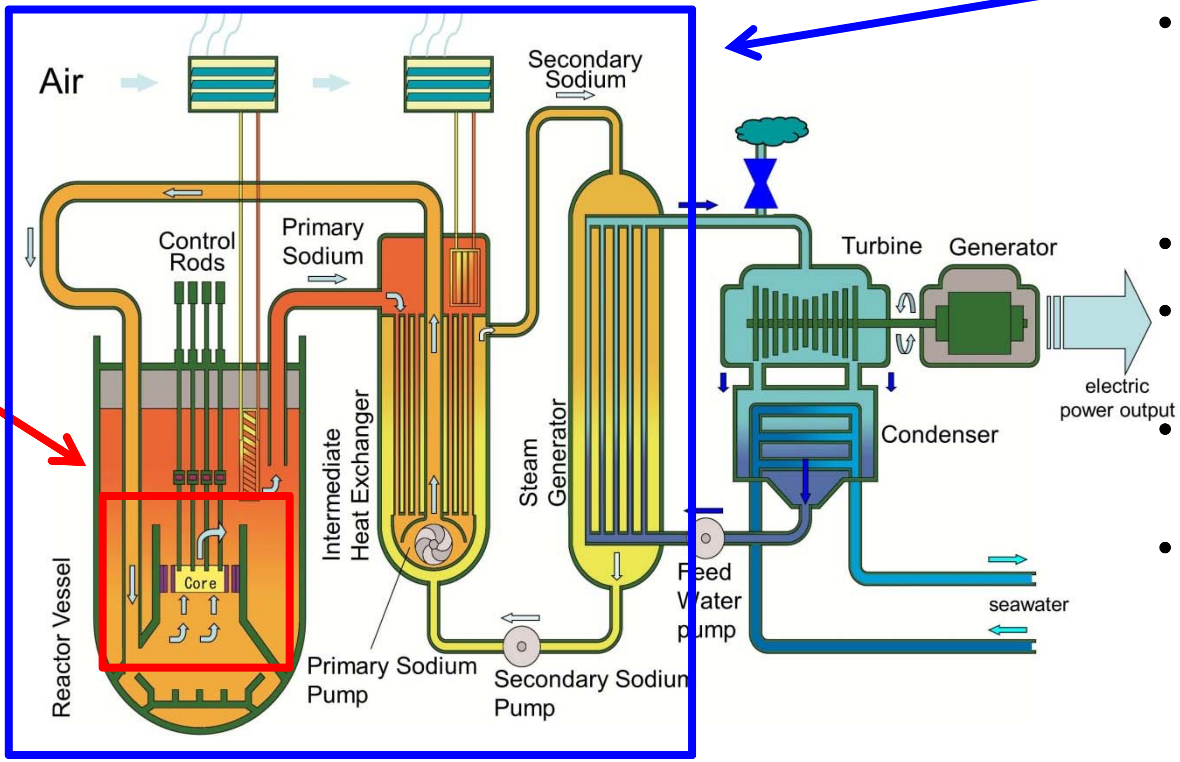
R&D on SFR Safety and Achievement

R&D on SFR Safety and Achievement

Safety study in various aspects accordingly to the characteristics of SFR has been implemented for decades, a certain accomplishment has been made and they are incorporated into the design evaluation for "Monju."

Core Safety

- Fuel pin failure
- Sodium boiling
- Fuel melting and relocation
- Fuel and sodium thermal interaction
- Fuel fragmentation and cooling
- Transfer of radioactive materials etc.



Sodium-cooled Fast Reactor

Plant Safety

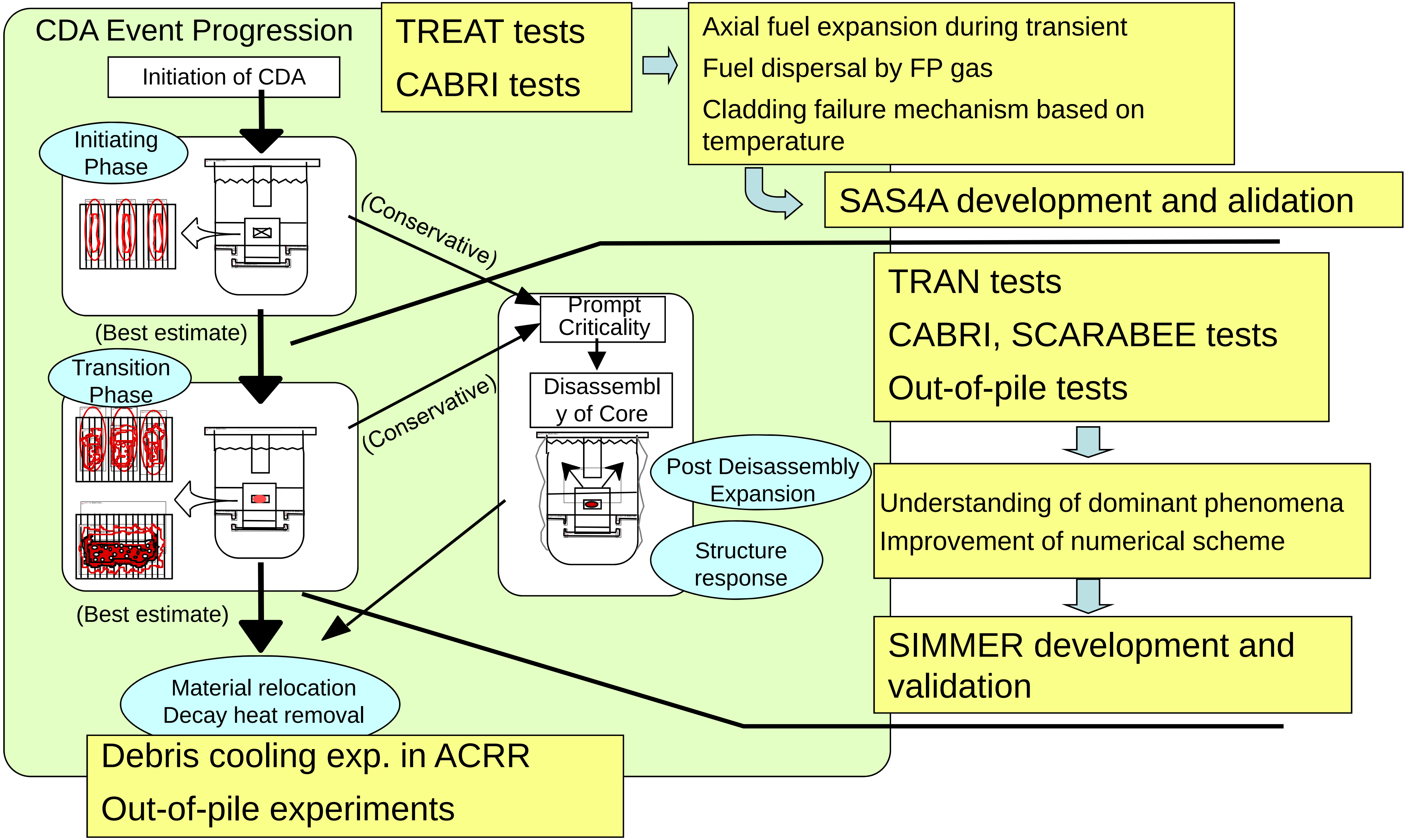
- Plant dynamics including natural circulation decay heat removal
- Sodium leak and fire
- Sodium-water reaction
- Sodium-concrete reaction
- Debris-concrete interaction etc.

Study on CDA (1/6)

- Core Disruptive Accident (CDA) has been one of the major safety issues of Sodium-cooled Fast Reactor (SFR) from the beginning of its development history.
- Extensive studies have been made to develop safety assessment technologies under international cooperation, which enabled mechanistic analysis of CDA event progression based on experimental knowledge.
- The application of these technologies confirmed the safety of Monju against CDA, demonstrating that the integrity of reactor vessel was safely kept against CDA in Monju.

Study on CDA (2/6)

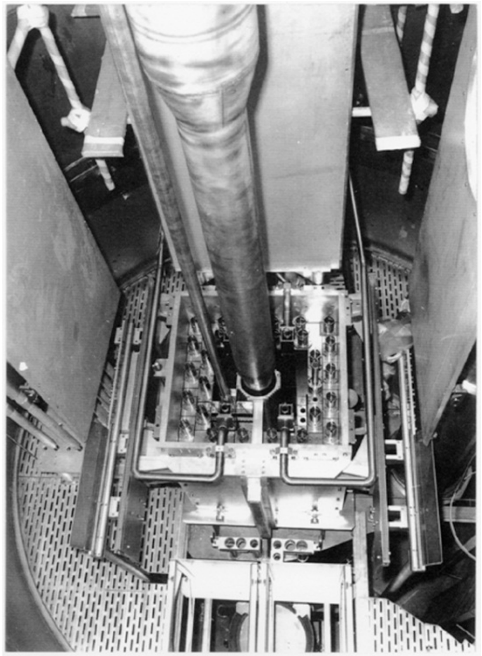
R&Ds on CDA in SFR



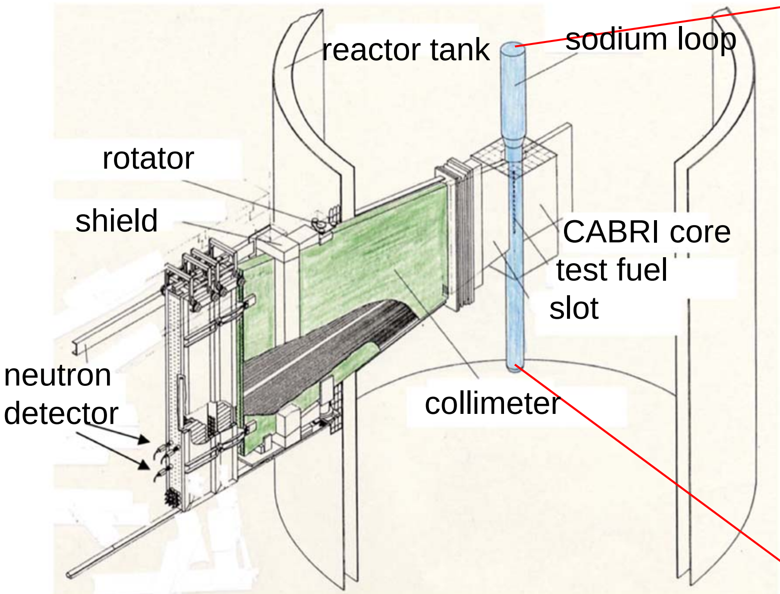
Study on CDA (3/6)

Safety experiments for CDA in SFR by CABRI reactor

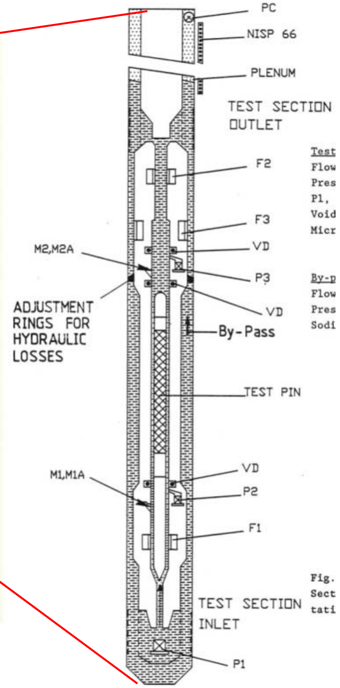
- Many safety experiments mainly for the early accident phase of ULOF in SFR in 4 programs from 1978 to 2002
- The failure of test fuel pin in sodium loop by pulse/ramp power transient was realized.
- Real time measurement of fuel motion by neutron hodoscope



Birdview of CABRI reactor



Neutron hodoscope

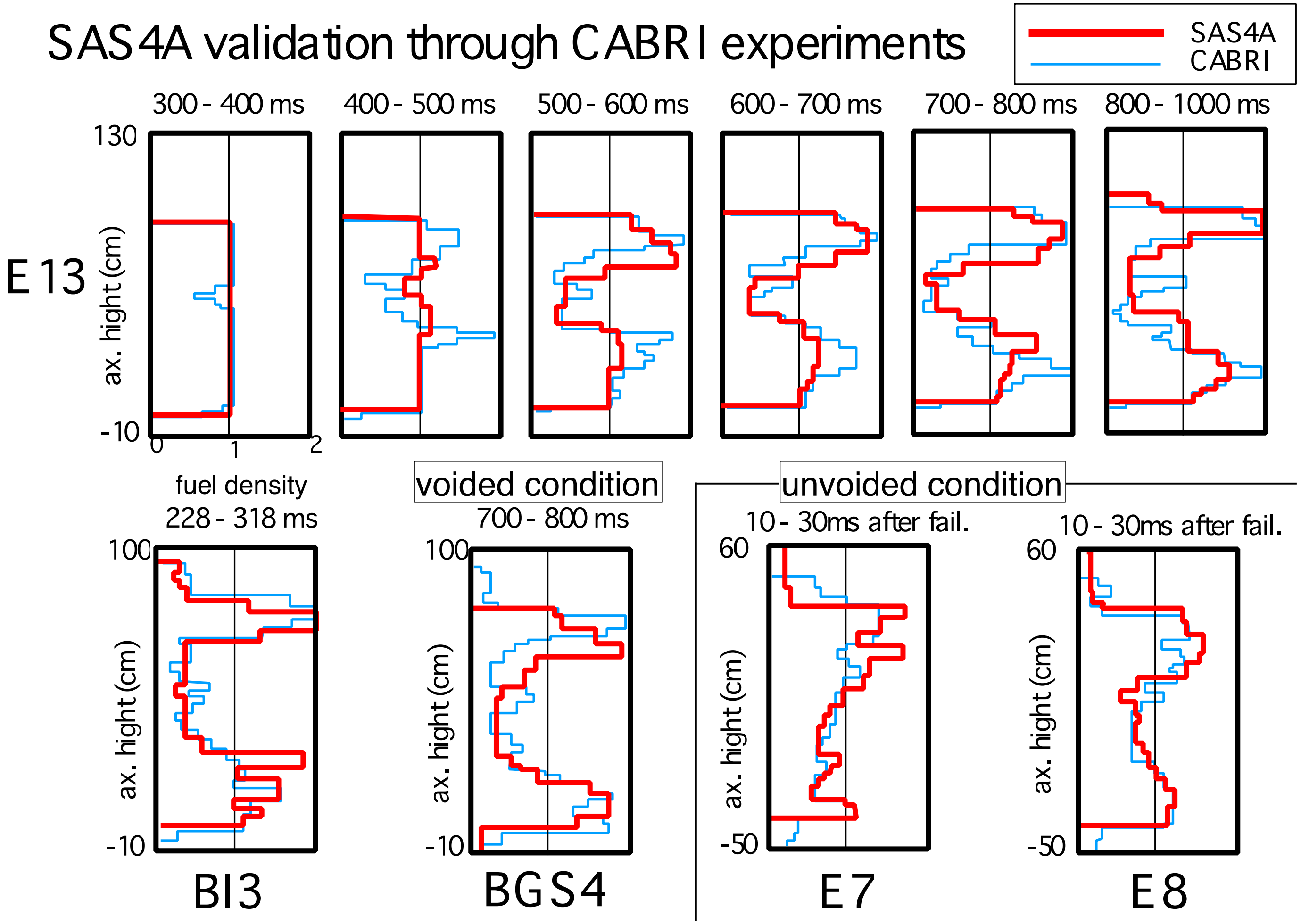


CABRI test section

Study on CDA (4/6)

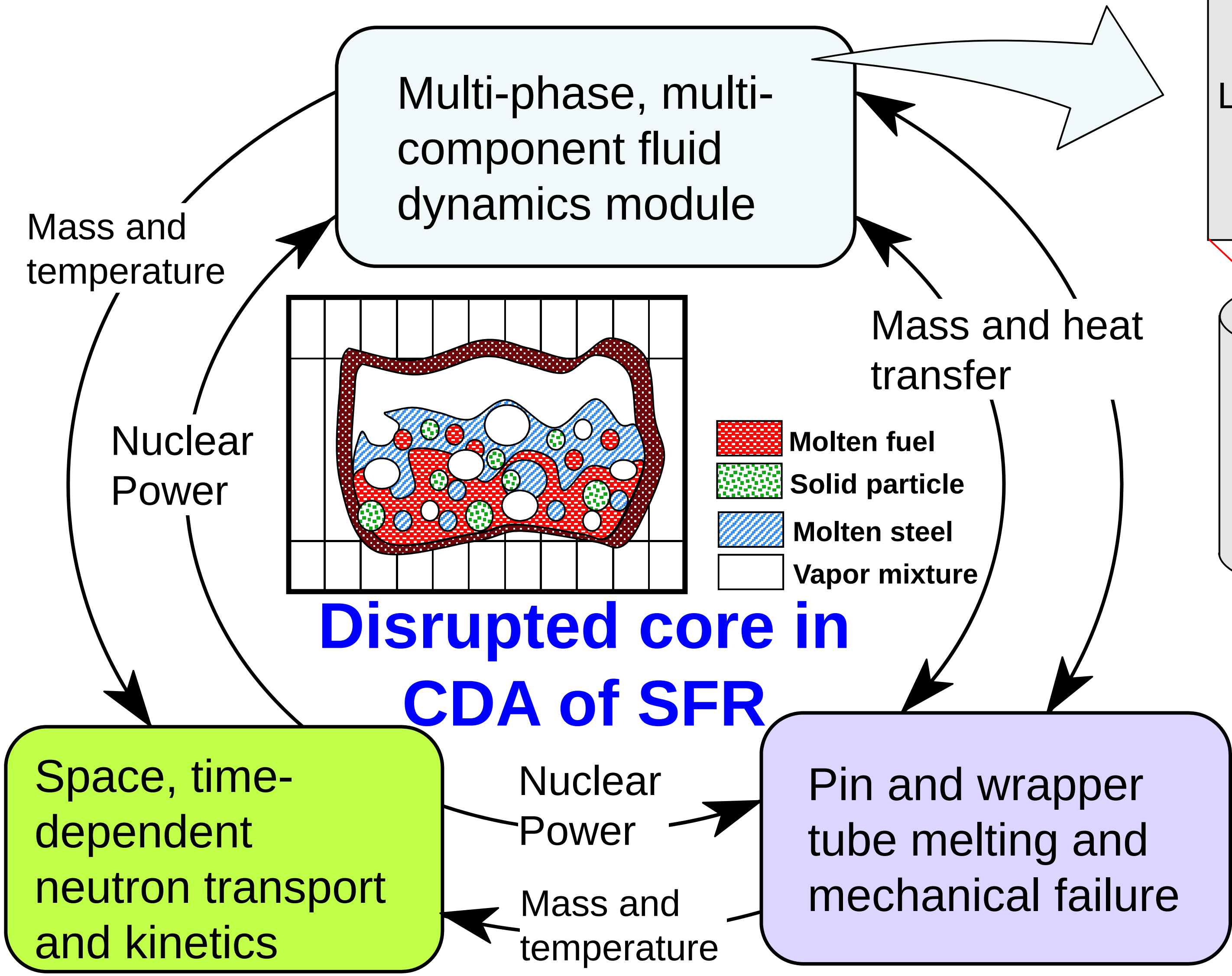
Validation of SAS4A code by CABRI tests: fuel dispersal

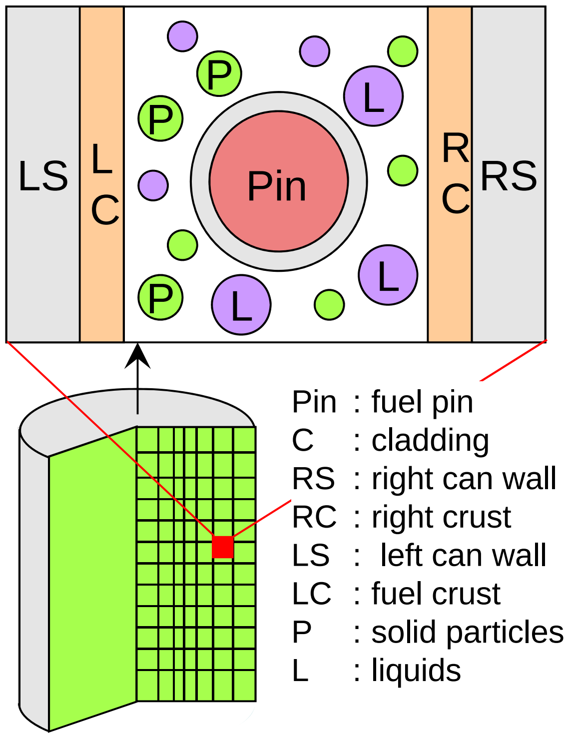
The fuel dispersal after fuel pin failure decreases reactivity and is one of the important phenomena in the early phase of CDA in SFR.
SAS4A model has been validated against CABRI tests with variety of fuel condition and transient condition.



Study on CDA (5/6)

Overall Framework of SIMMER Code



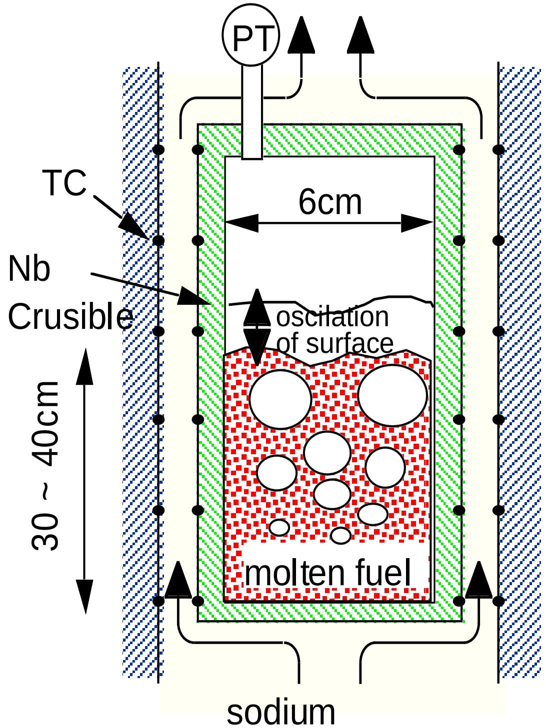


- 2-D (SIMMER-III, r-z or x-z), 3-D (SIMMER-IV, x-y-z)
- 8-velocity fields
- Multi-phase
  solid, liquid, vapor
- Multi-components
  5 material components,
  27 density components,
  16 energy components

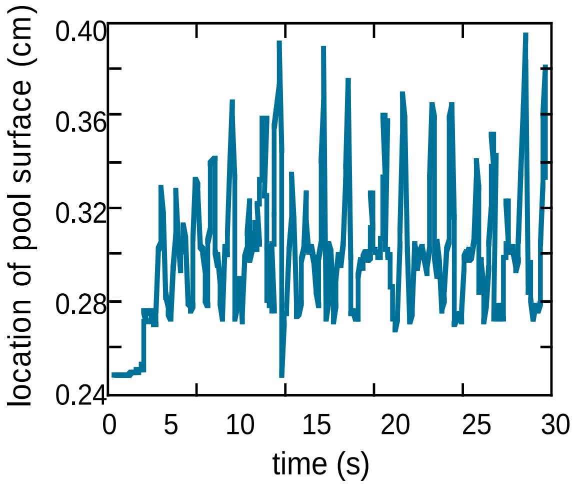
Study on CDA (6/6)

Validation of SIMMER code : Boiling of core material

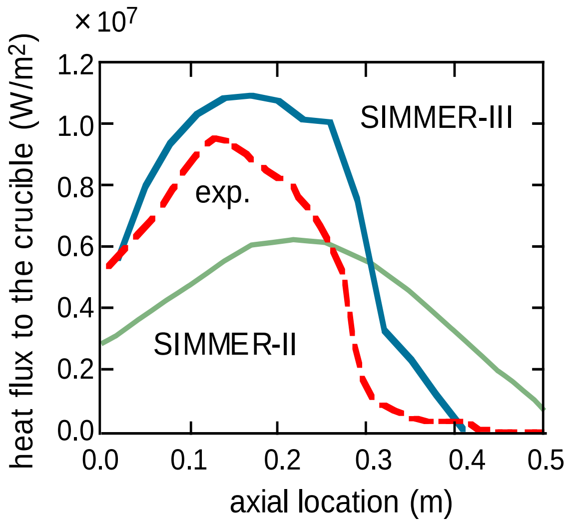
The boiling behavior of molten core material dominates the motion of fuel in the core of SFR and affects the recriticality magnitude.



SCARABEE BF2 experiment



calculated location of pool surface



heat flux distribution tohe cruciblt wall

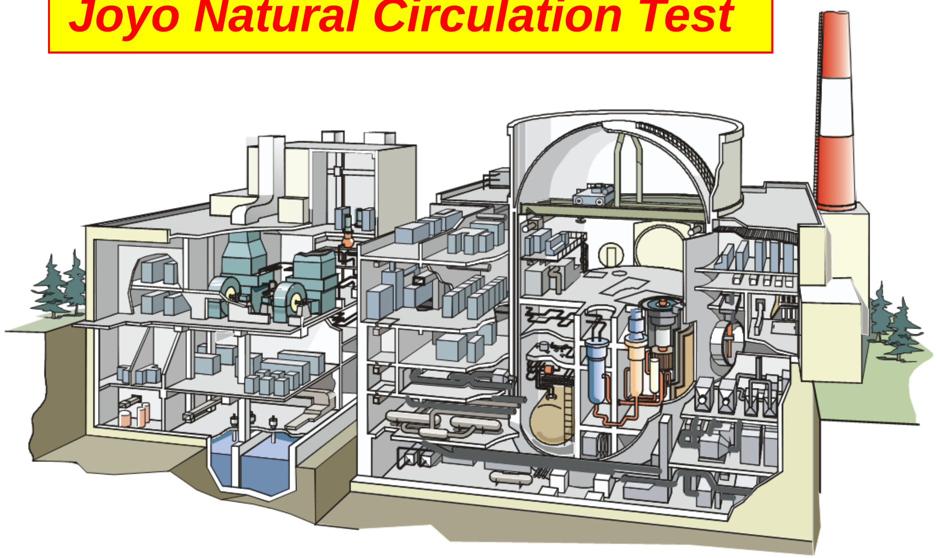
Comparison of the frequency and amplitude of pool surface oscilation

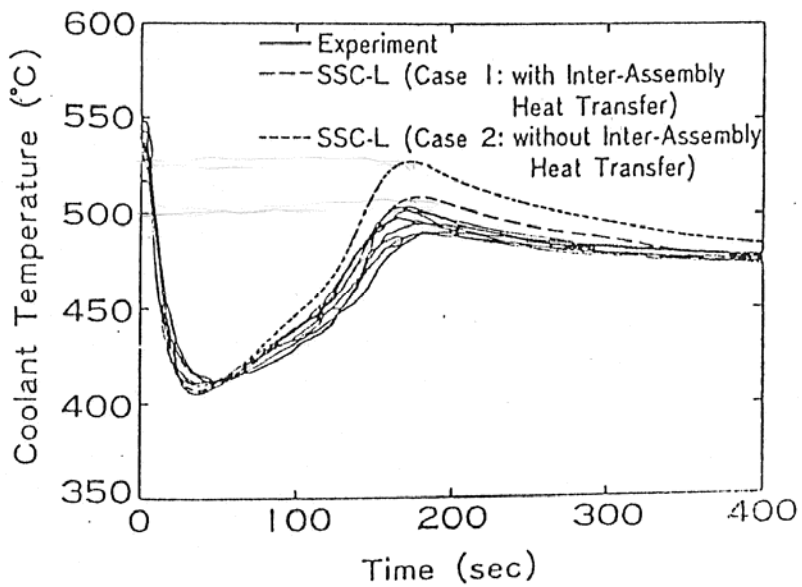
	BF2 experiment	SIMMER-III	SIMMER-II
amplitude	10 cm	10 cm	30 cm
frequency	0.8 Hz	0.9 Hz	1.2 Hz

Study on Plant Safety: Plant dynamics

- Natural circulation heat removal behavior has been demonstrated in actual SFRs such as Joyo, Rapsodie, Phenix, SPX-1, FFTF, EBR-II, PFR, KNK-II
- Plant dynamics codes have been developed and validated based on these results.
- Auxiliary core cooling system of Monju is designed with these technologies.

Joyo Natural Circulation Test

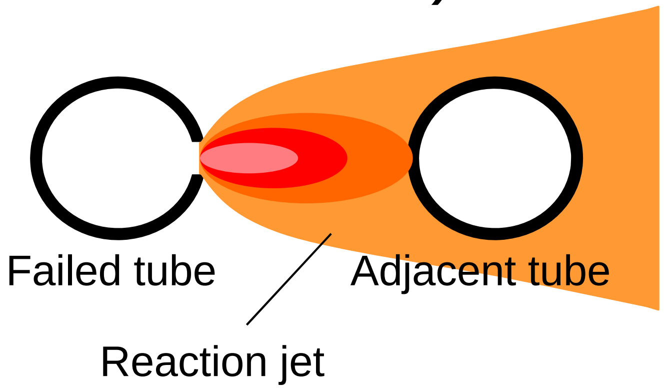




Core Outlet Temperatures and Analysis Results

Chemical reaction issues

- Sodium-Water Reaction (steam leak accident in SG)
  - SERAPHIM code
    - Multi-fluid model considering compressibility
    - Advection-diffusion model for chemical species
    - Chemical reaction model



- Sodium Leak and Fire
  - Various experimental data accumulated (>100 runs)
  - Computational code system already developed, in the status of systematic V&V activities
    - Zone model code SPHINCS
    - Field model (3-dimensional) code AQUA-SF

Example of initial behavior for sodium leak and fire

- Ex-Vessel Accident Evaluation
  - Improvement and validation of CONTAIN/LMR code
    - Zone model code: sodium fire, sodium-concrete reaction, debris-concrete interaction and FP/aerosol behavior, etc.

R&D on Safety toward GEN-IV SFR

R&D on Safety for Realization of GEN-Ⅳ SFR

- JSFR is a large reactor aiming for safety enhancement with eliminating excess conservatism while pursuing economical competitiveness for commercialization.
- Utilizing the acquired knowledge and experience, measures to enhance its safety are incorporated into the design. Related R&Ds are conducted for acquiring data to quantify the effect of the design measures and for developing the analytical tools.
- Feasible safety enhancement measures for SFR can be realized through such R&Ds. This will contribute establishing SDC for GEN-Ⅳ SFR.
- A stress shall be put upon R&Ds for passive reactor shutdown and decay heat removal, and for event termination for postulated core damage.

Requirement from Safety Design Criteria (SDC)

- For a Compact and Large-power SFR
  - Large inventory of core material
    - Reactivity
    - Heat removal
  - Small volume of containment vessel
- Enhancement of Defense in Depth
  - Level 4 (DEC): Control of SA including
    - Prevention of core damage
    - Mitigation of consequences of SA
  - Built-in measures for Prevention and Mitigation of SA

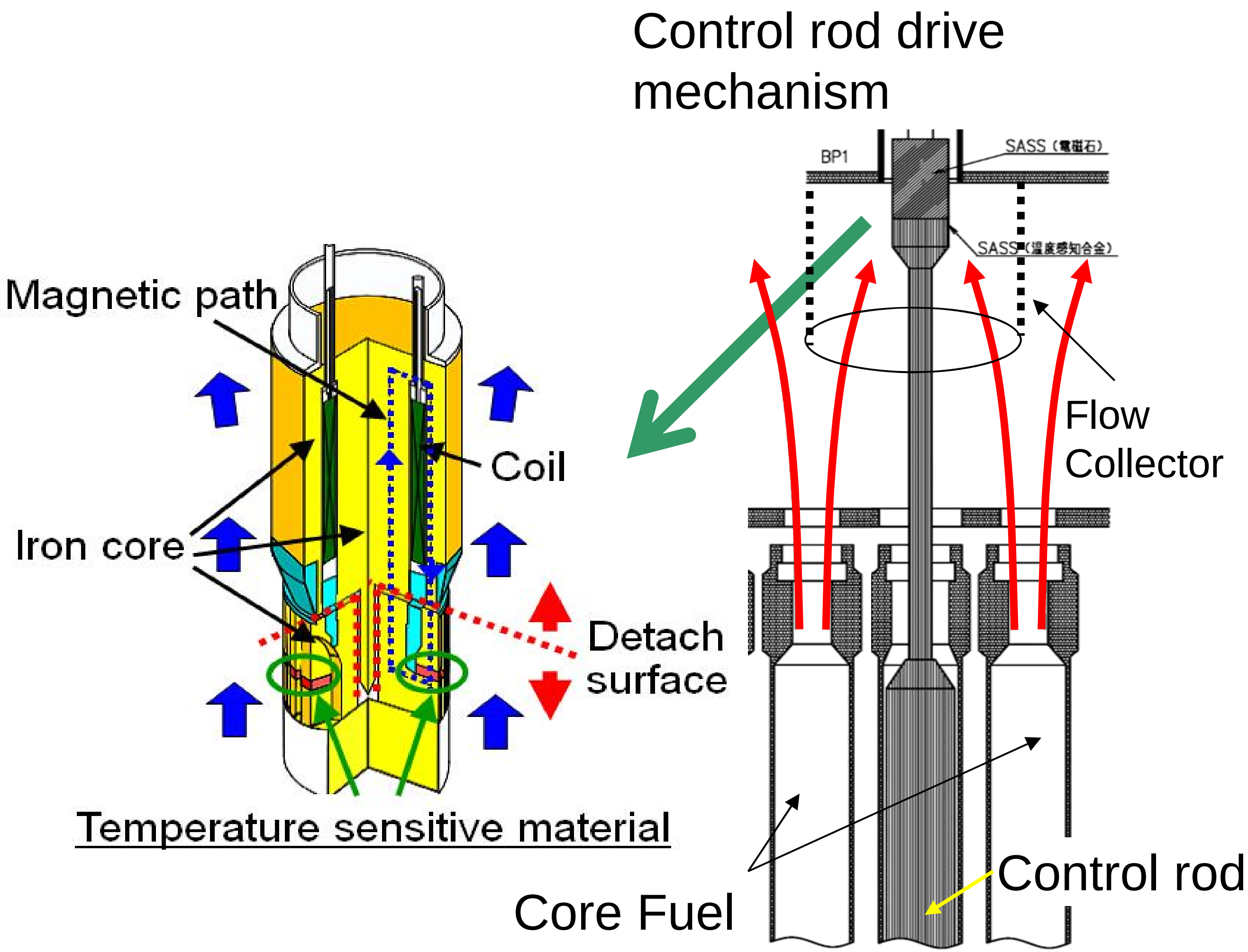
Gap toward Gen-IV Reactors satisfying SDC

- Built-in Measures with Passive and Diverse Features
  - Reactivity Control
  - Decay Heat Removal
- Prevention of Core Damage
  - Passive Reactor Shutdown
  - Passive Decay Heat Removal Systems
    - Long term SBO
    - Survive in wide variation of external events
- Mitigation of Core Damage
  - Control of Core Material Relocation in CDA
    - Release paths of core material
  - Diversity of Decay Heat Removal after Core Damage

R&D on Counter Measures for Gen-IV Reactors

- SASS for passive shutdown
  - Curie-point detach mechanism
  - Stability of holding mechanism under reactor condition
    - Reactor experiment using Joyo
  - Response of detach mechanism
- Natural Circulation Decay Heat Removal (NCDHR)
  - Temperature flattening in core
  - Uncertainty of flow and core temperature
  - Multi-dimensional flow in large components
- Mitigation Measures
  - Control of core material relocation
  - RV cooling for in-vessel retention

Study on SASS



The control rod (C/R) is passively inserted by using a magnetic change at the Curie point temperature of the temperature sensing alloy.

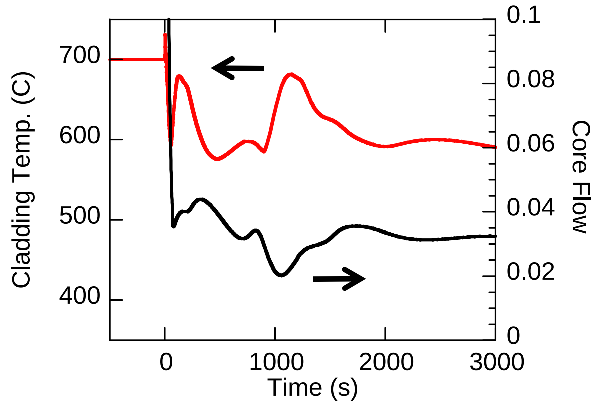
Hot fluid transportation from core fuel subassemblies to SASS magnet in TOP, LOF and LOHS incidents

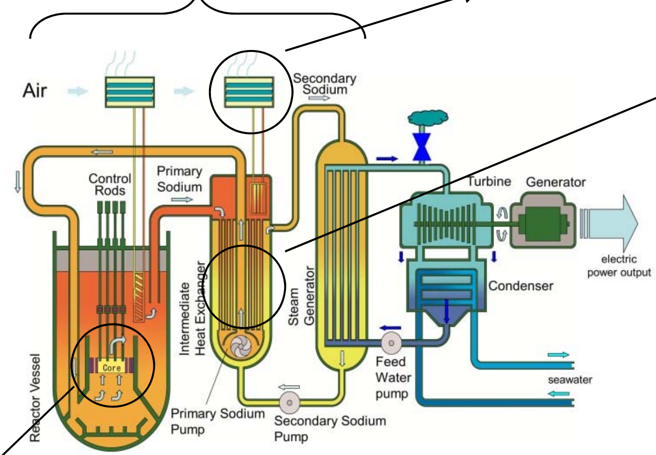
Influence of Buoyancy Force in LOF (Loss of Flow) Events

Evaluation of Transient 3D Flow and Temperature Fields in RV upper plenum

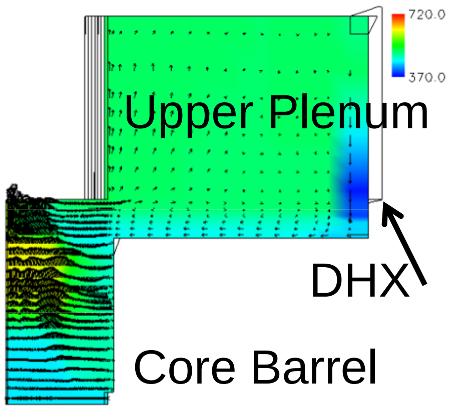
Study on NCDHR

- System Code Analysis
  - ✓ The highest temperature in core
  - ✓ Flow network model
  - ✓ Uncertainty evaluation
- 3D flows in Components
  - ✓Thermal stratification
  - ✓Biased flow in mixed convection
- Freezing in A/C tubes



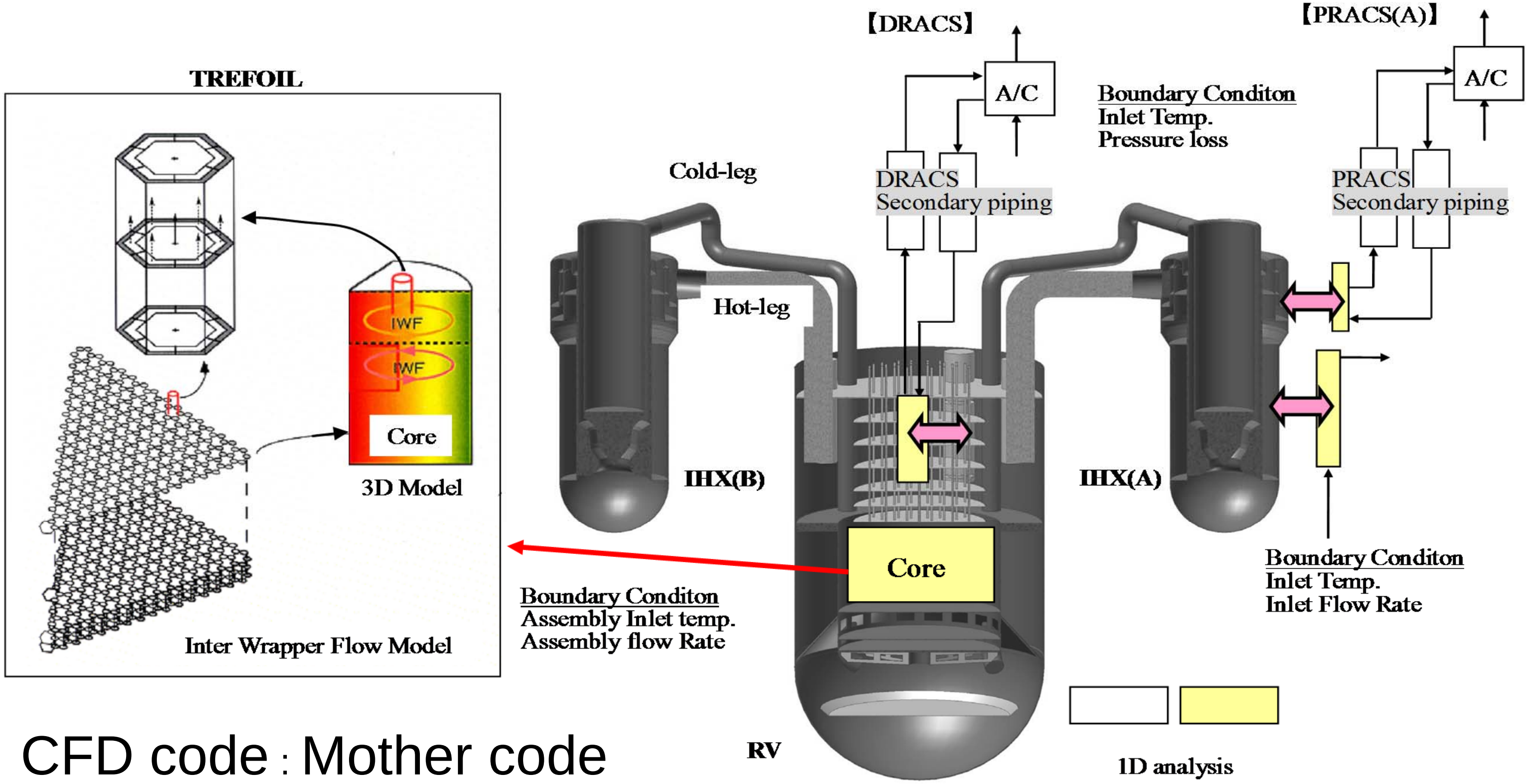


- Complex flow in Core
  - ✓ Temperature flattening due to flow redistributions intra- and inter- subassembly
  - ✓ Natural convection in core barrel (inter-wrapper flow)



- ➢ Precise Core Model
  - All subassemblies are modeled
  - IWF model
  - Subchannel Model of subassembly
- ➢ Full 3D simulation of primary loop
- ➢ Statistic Approach of uncertainty evaluation

Study on NCDHR -Full 3D model of Primary System-



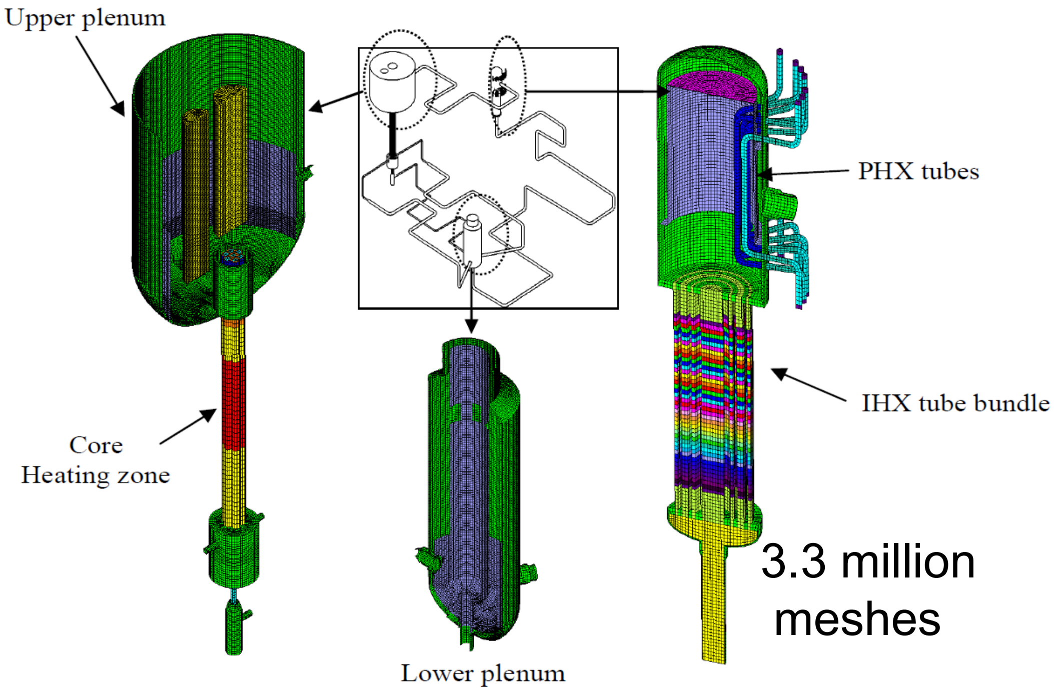
CFD code : Mother code

TREFOIL : Core thermal-hydraulics

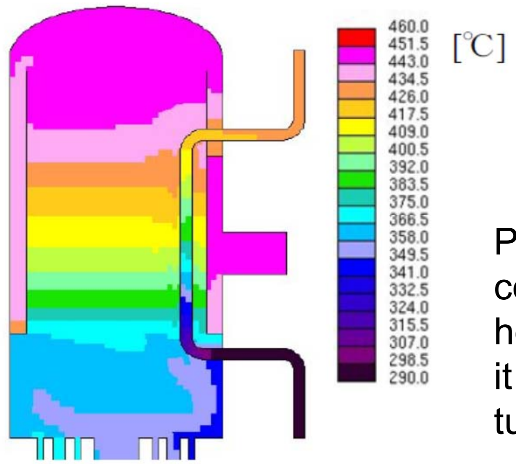
1-D code : Heat exchangers and secondary side of DHRS

O. Watanabe, et al., Development of Numerical Analysis Methods for Natural Circulation Decay Heat Removal System Applied to a Large Scale JSFR, NURETH14-258 (2011).

Study on NCDHR -Validations using Sodium Experiments-

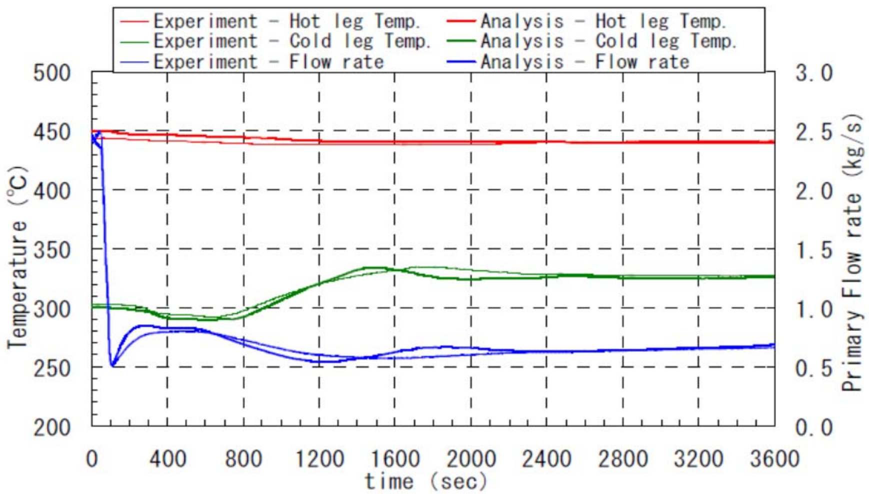


Three-dimensional analysis model

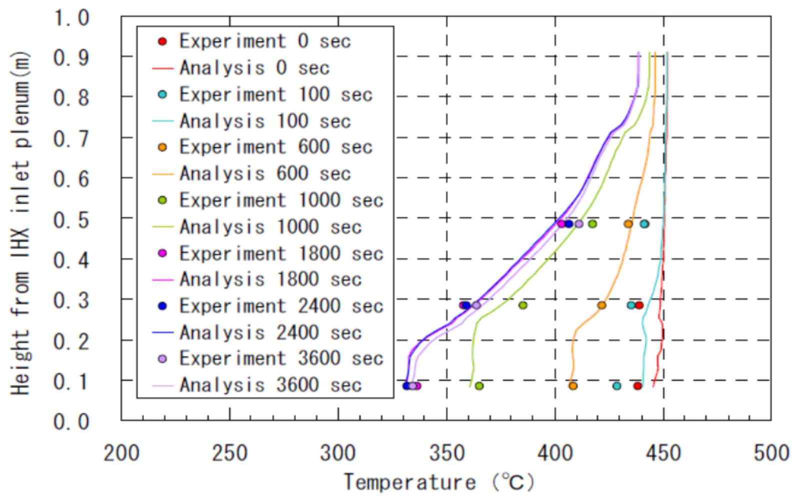


Temperature distribution
in IHX upper plenum at 1000s

Primary-side sodium cooled by PRACS tube horizontally diffuses, while it locally goes down along tubes by buoyancy force.



Transient in primary system



Vertical temperature distributions in IHX upper plenum

Study on NCDHR -Validation of System Code-

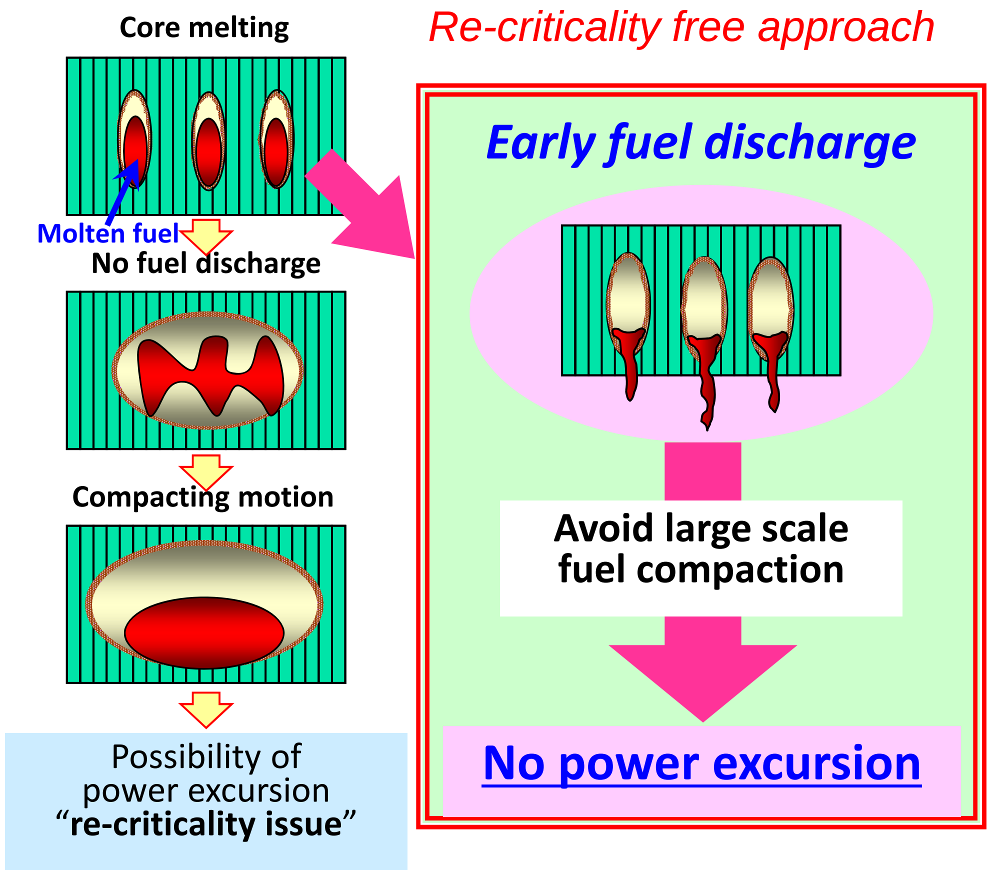
- Test data to be obtained at Monju is significant for confirmation of the scale effect.

  - Scale Effect
    - Reactor experiments
      - Joyo, Phenix, EBR-II
      - Monju
  - Monju Reactor
    - Reactor vessel: ϕ 8m
    - Core: 714MWth, Core Fuel Subassembly: 198
    - Air cooler and stack: 15MW x 3 units, Stack Height 10m
    - Natural circulation experiment is planned.

Study on CDA Mitigation for Gen-IV Reactors

Design concepts: In-Vessel Retention of CDA

- ***Re-criticality free* concept**
  - Avoidance of power excursion, by early molten-fuel discharge
    - The formation of whole-core molten pool might bring a power excursion due to the compacting motion.
    - Molten fuel should be early discharged from the core before the failure of fuel-assembly can-wall.
  - This enables the employment of compact CV.



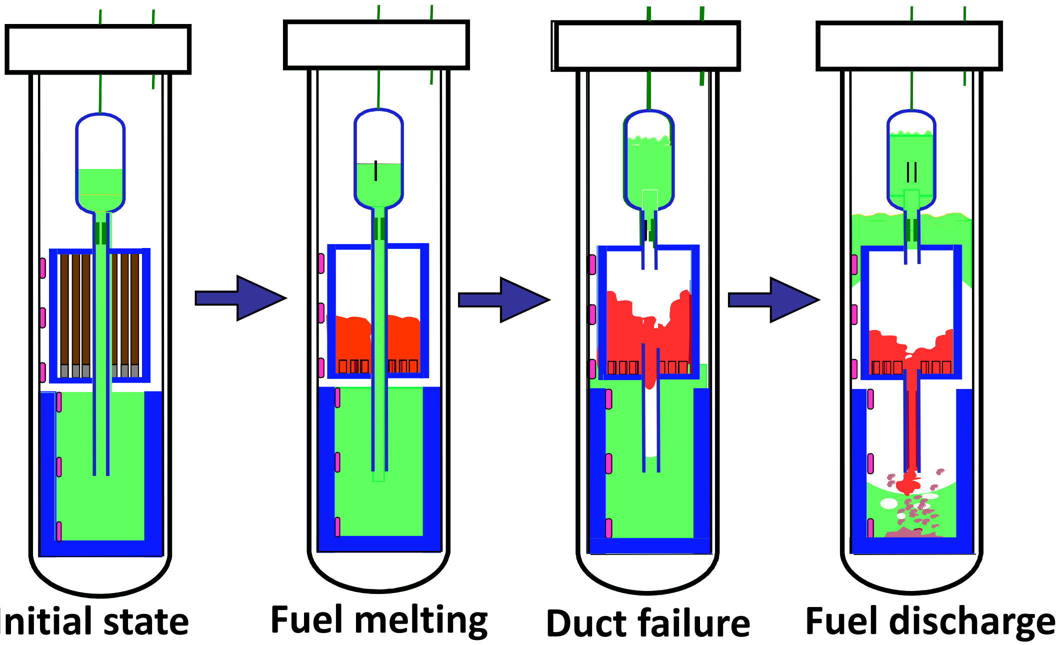
- ***In-vessel debris cooling***
  - Avoidance of thermal boundary-failure, by stable cooling
    - Fuel inventory to be discharged from the core is comparatively large in the respect of reactor-vessel diameter.
    - Discharged fuels should be retained/stably-cooled on the multi-layer debris tray by sodium that has excellent heat transfer characteristics.

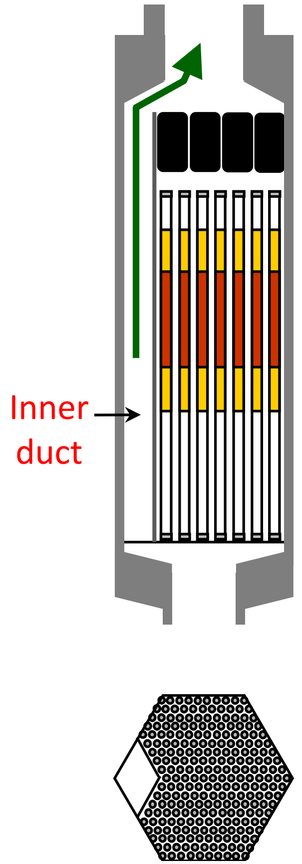
In-pile Experiment on Core Material Relocation

Experimental demonstration of early fuel discharge by EAGLE program

- High (> 8 MW/m^2) heat flux from the molten core materials to duct.
  - The wall heat up was dominated by its thermal inertia.
  - The inner duct of FAIDUS would break earlier than the wrapper tube.
- Rapid fuel discharge even with a low pressure difference (ca. 0.1MPa).
  - High pressure difference expected in the reactor condition (~1MPa) would result in rather rapid fuel discharge.

IGR for in-pile tests in EAGLE program



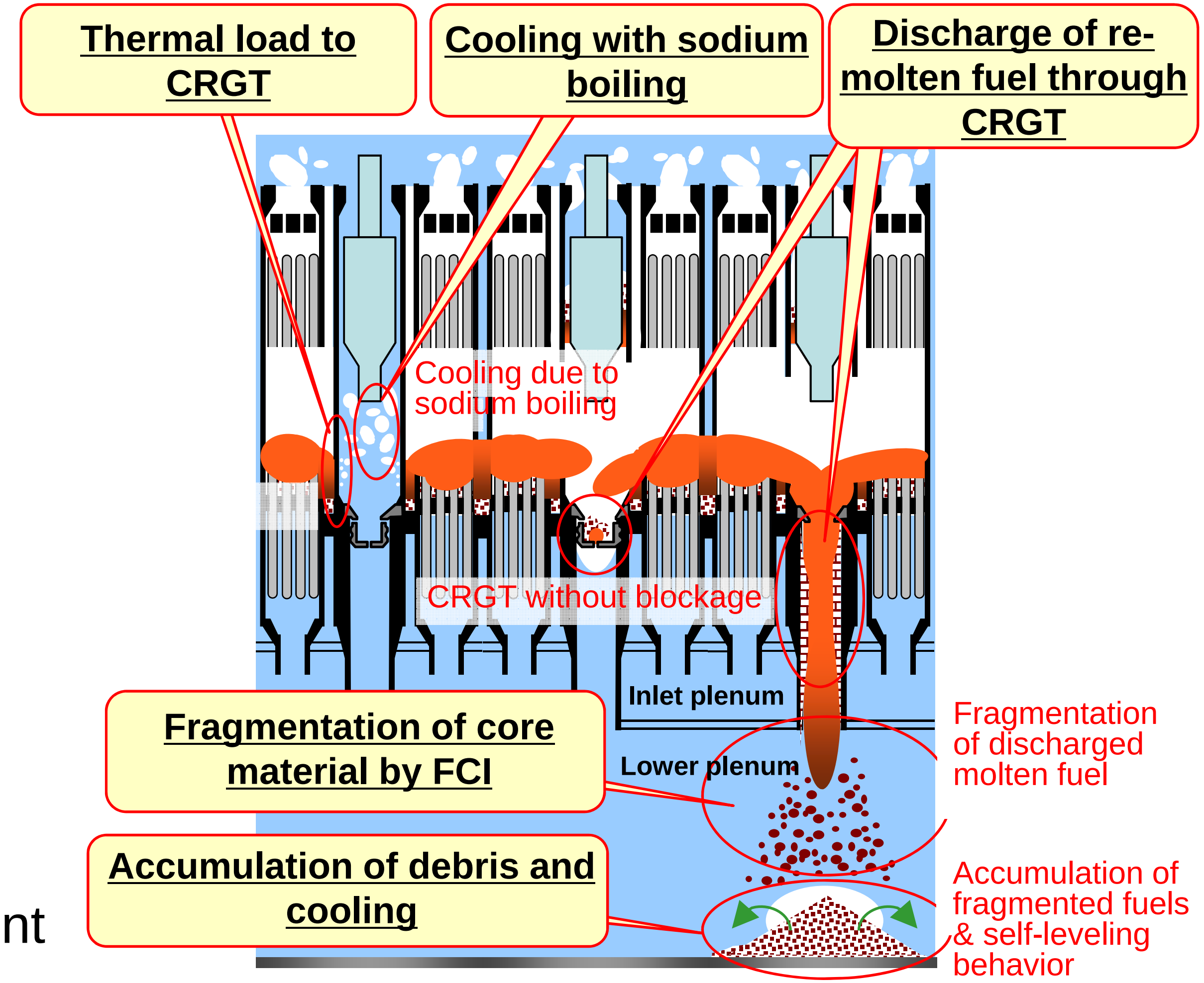


FAIDUS

R&D issues for mitigation of core damage (1)

Key phenomena in achieving IVR in CDA of SFR

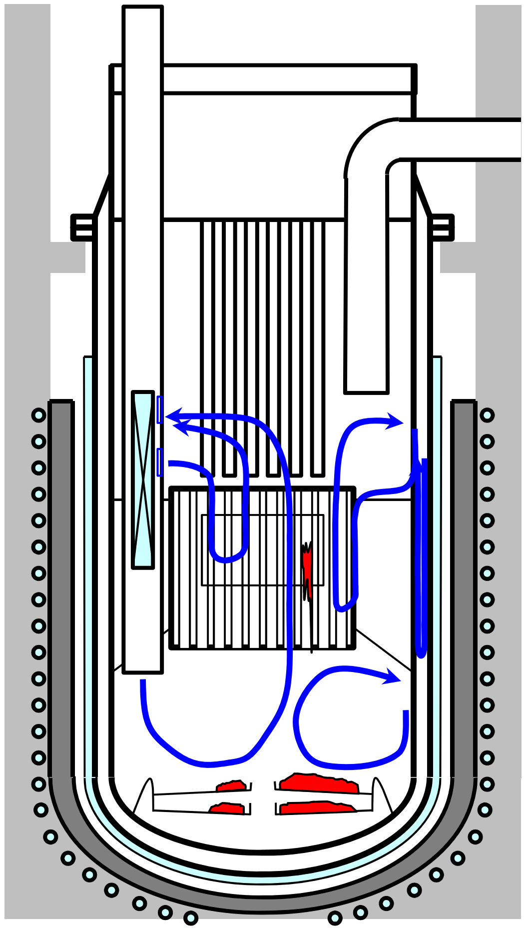
- The perspective for the elimination of the severe recriticality events has been obtained by the analysis of event progression reflecting the knowledge of EAGLE experiments.
- The remaining issues to achieve IVR are the decay heat removal and stable retention of degraded core materials.
- The key phenomena which require future experimental study and R&D of assessment measures are shown.



R&D issues for mitigation of core damage (2)

- For cooling of degraded core, considering the lessons learned from the TEPCO's accident, diversification of cooling measures is essential and a related test program using large scale sodium test facility is under planning.

- Decay Heat Removal
  - Disrupted core
    - Flow path in core
  - Debris in core catcher and reactor upper plenum
  - Sodium level above the core
- Dipped Heat Exchanger (DHX) in RV
- Other measures for variety of DHR system
  - Heat removal through RV wall



Conclusions

- Safety study in various aspects accordingly to the characteristics of SFR has been implemented for decades, as a result, a certain accomplishment has been made and they are incorporated into the design evaluation for “Monju”.
- Taking the lessons learned from the TEPCO’s Fukushima Dai-ichi NPS accident, continuous efforts should be made to enhance safety for SFR, and it is important to advance the safety study utilizing existing reactor such as Monju and its related research facilities for SA countermeasures.
- Based on the achievement of such R&D efforts, presenting feasible safety design measures for GEN-Ⅳ SFR is significant to contributes to establish SDC for the SFR to be the future international standard.