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
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.

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

CDA Event Progression

Initiation of CDA

Initiating Phase

(Conservative)

(Best estimate)

Transition Phase

(Conservative)

(Best estimate)

Material relocation
Decay heat removal

TREAT tests
CABRI tests

Axial fuel expansion during transient
Fuel dispersal by FP gas
Cladding failure mechanism based on temperature

SAS4A development and alidation

TRAN tests
CABRI, SCARABEE tests
Out-of-pile tests

Understanding of dominant phenomena
Improvement of numerical scheme

Debris cooling exp. in ACRR
Out-of-pile experiments
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
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

Mass and temperature

Nuclear Power

Disrupted core in CDA of SFR

Space, time-dependent neutron transport and kinetics

Multi-phase, multi-component fluid dynamics module

Mass and heat transfer

Pin and wrapper tube melting and mechanical failure

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

Pin : fuel pin
C : cladding
RS : right can wall
RC : right crust
LS : left can wall
LC : fuel crust
P : solid particles
L : liquids
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.

Comparison of the frequency and amplitude of pool surface oscillation

<table>
<thead>
<tr>
<th></th>
<th>BF2 experiment</th>
<th>SIMMER-III</th>
<th>SIMMER-II</th>
</tr>
</thead>
<tbody>
<tr>
<td>amplitude</td>
<td>10 cm</td>
<td>10 cm</td>
<td>30 cm</td>
</tr>
<tr>
<td>frequency</td>
<td>0.8 Hz</td>
<td>0.9 Hz</td>
<td>1.2 Hz</td>
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</table>
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.

[Diagram of 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

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

Study on NCDHR -Validations using Sodium Experiments-

Three-dimensional analysis model

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

Temperature distribution in IHX upper plenum at 1000s

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: \( \phi \) 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/m²) 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.
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.