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Severe Accidents in Sodium Fast Reactors: Safety Study Approach, Prevention and Mitigation by Design

Dr. Nathalie Seiler, Prof. Frederic Bertrand, and Mr. Shigenobu Kubo

05 November 2025

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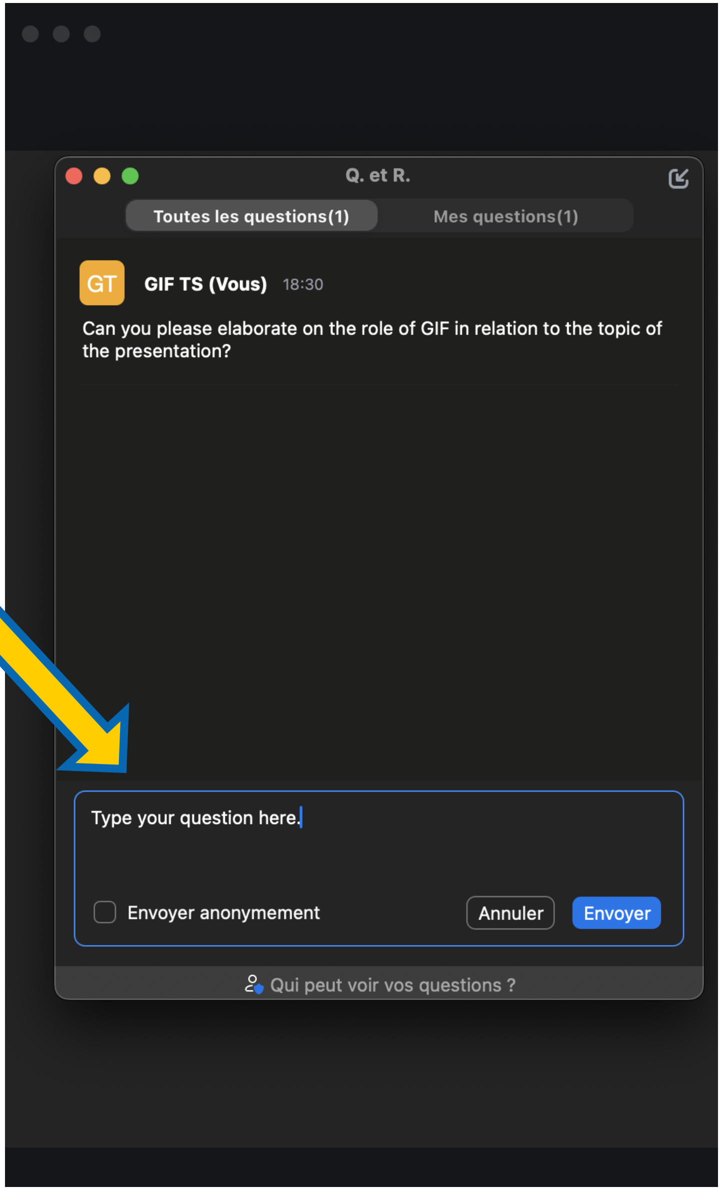

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In Memoriam of Joël Guidez

The GIF Education and Training Working Group expresses its sadness and infinite gratitude to Mr. Joël Guidez, a fervent supporter of fast neutron reactors with two major French Milestones Phénix and Superphénix, and his contribution to the advanced molten salt reactor community.

We have captured a little of Joël's brilliant scientific, and professional legacy through the following webinars:

29 NOV 2017 – "Phénix and Superphénix – Feedback Experience"

https://www.gen-4.org/resources/webinars/education-and-training-series-15-phenix-and-superphenix-feedback-experience

27 JAN 2022 – "ESFR SMART European SFR project"

https://www.gen-4.org/resources/webinars/education-and-training-series-61-esfr-smart-european-sodium-fast-reactor-concept

21 JUN 2023 – "International Knowledge Management and Preservation of SFRs"

https://www.gen-4.org/resources/webinars/education-and-training-series-78-international-knowledge-management-and

Thank you Joël for your enthusiasm, dedication, and joie de vivre!

Meet the Presenters

Prof. Frédéric Bertrand (Editorial Board Manager of the book "*"Severe Accidents in Sodium Fast Reactors: Safety Study Approach, Prevention and Mitigation by Design*) is a Research Director, a CEA Senior fellow and a Professor at the Institut National des Sciences et Techniques du Nucléaire (INSTN). He has worked for about 20 years at CEA in safety and conceptual design of innovative nuclear reactor (Gen IV and space reactors).

Dr. Nathalie Seiler (Editorial Board Member, responsible for chap 5) is a research director, fellow expert at the CEA. She is a thermal-hydraulics engineer and researcher in the field of severe accidents in third- and fourth- Generation nuclear reactors and has supervised the development of physical simulation tools, coupled with advanced statistical tools, for assessing the safety of sodium Fast Reactors under various accident transients.

Mr. Shigenobu Kubo (chapter editor for Japanese SFR's projects) is a deputy Director of Fast Reactor Research and Development Department at the Japan Atomic Energy Agency (JAEA). He has been engaged in sodium-cooled fast reactor development since 1989 in Japan. His specialties are SFR system design, safety design, and related R&Ds. He was involved in the France-Japan ASTRID collaboration as design task leader and severe accident task leader.

Email: frederic.bertrand@cea.fr; nathalie.seiler@cea.fr; kubo.shigenobu@jaea.go.jp

CHAPTER 1 (F. Bertrand)

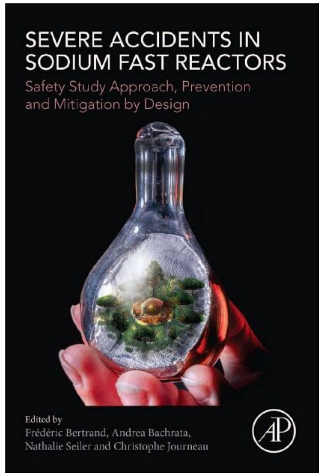
Introduction

Frédéric Bertrand
CEA, DES, IRESNE, Reactor Studies Department, Saint-Paul-lez-Durance, France

- **Objectives of the Book**

  Capitalize on knowledge and results concerning ASTRID's design in relation to Severe Accidents (SA).

  ASTRID (acronym for Advanced Sodium Technological Reactor for Industrial Demonstration) was the French fourth-generation nuclear reactor prototype project - sodium-cooled fast reactor- developed from the 2010s to 2019.
  This book is devoted to engineers and researchers seeking knowledge about SAs in sodium-cooled fast reactors (SFRs) and about the design of key components aimed at limiting their consequences. French and international past experiences are summarized.



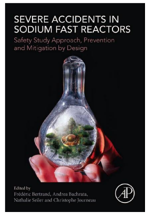
- **Mitigation of Severe Accidents: A Crucial Issue for SFRs**

  Mitigation = Limiting the consequences in terms of radiological releases

  - Given:
    - The reactivity potential of an SFR core that is highly enriched in fissile material
    - The chemical reactivity of sodium: prevention of large sodium fires in the event of a severe accident

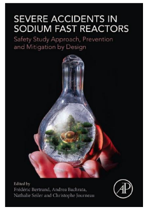
→ Mitigation of severe accidents has been considered from the very first designs of sodium-cooled fast reactors, unlike in pressurized water reactors (PWRs).

CHAPTER 1



- **Chapters of the Book (568 pages)**
  - **1 –** Introduction
  - **2 –** Dynamic and historical vision of the severe accident approach in previous projects
  - **3 –** Design elements of **ASTRID** and **JSFR**
  - **4 –** Considerations on **severe accident (SA)** scenarios and study methodologies integrating mitigation from the design stage
  - **5 –** Calculation tools used for SA studies
  - **6 –** Mitigation studies carried out for **ASTRID**
  - **7 –** Experimental studies supporting the design of mitigation systems
  - **8 –** Mitigation approaches and studies for **JSFR** and other recent **SFR** concepts
  - **9 –** Additional **R&D** needs
  - **10 –** Conclusion

CHAPTER 1

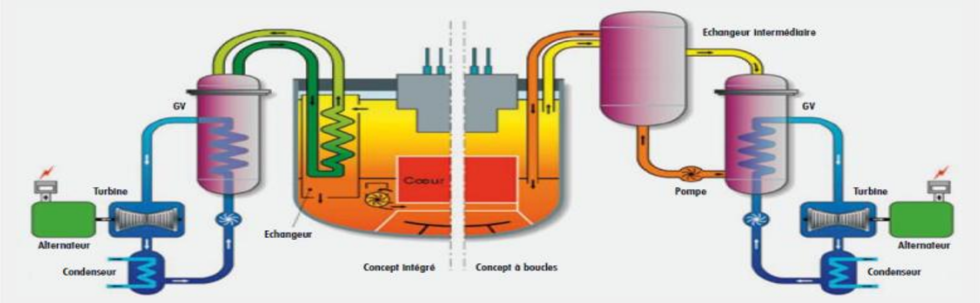


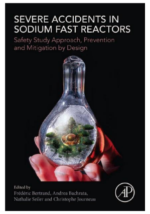
- **Editorial Committee**

  – **Book Editors and Chapter Editors:**
  F. Bertrand (**Chief Editor**), A. Bachrata (**Editor**), N. Seiler (**Editor**),
  C. Journeau (**Editor**), S. Kubo (**Chapter Editor**), B. Carluec (**Chapter Editor**)

- **Contributions as Co-authors:**
  Engineers and researchers from **JAEA**, **Framatome**, **EDF**, **IRESNE** (DER, DTN, DEC), CEA/**ISAS** (DM2S and DRMP) – *32 participants*

CHAPTER 2 (B. Carluec)

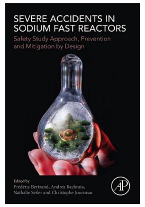




Dynamic and historical vision of severe accident approach in SFR former projects

Bernard Carluec[1,a], Shigenobu Kubo[2], Fabienne Audubert[5], Nathalie Seiler[3], Frédéric Bertrand[4], Andrea Bachrata[3], Pierre Sciora[4], Christophe Journeau[3] and Magali Zabiégo[3]
[1]FRAMATOME, Technical & Engineering Division, Lyon, France; [2]Oarai Research and Development Institute, Japan Atomic Energy Agency, Oarai, Ibaraki, Japan; [3]CEA, DES, IRESNE, Nuclear Technology Department, Saint-Paul-lez-Durance, France; [4]CEA, DES, IRESNE, Reactor Studies Department, Saint-Paul-lez-Durance, France; [5]CEA, DES, IRESNE, Fuel Studies Department, Saint-Paul-lez-Durance, France

- **Chapter Content (117 pages)**
  - General concepts on the safety approach for sodium-cooled fast reactors (SFRs) with respect to severe accidents (SAs) *(Superphénix, EFR, ASTRID)*
  - Main physical characteristics of a severe accident in an SFR
  - Aspects related to core reactivity
  - Aspects related to sodium
  - Historical overview of design options concerning SFR severe accidents *(France, Japan, and Europe)*
  - Options for prevention of core melting
  - Options for prevention and mitigation of mechanical energy release
  - Long-term cooling and containment



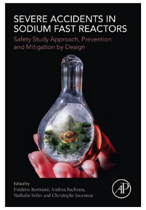
CHAPTER 2

Safety features of a SFR (compared to PWR): core aspects

- At their nominal operating point, **sodium-cooled fast reactor (SFR) cores** are far from their most reactive configuration (high enrichment in fissile material).
  - The **positive reactivity contribution** from sodium voiding in the core can reach several dollars (about **7 $ for SPX**).
  - A change in geometry can lead to **reactivity insertion**:
    - **Core compaction** resulting from core melting → several tens of dollars
    - **Cladding relocation** → around ten dollars
  - A **1% radial compaction** of the core leads to a **reactivity insertion > 1 $**.

- The **core kinetics parameters** and **stabilizing effects** are **less favorable** than those of **PWRs**.

Concept	SFR	PWR
Delayed neutron fraction	~ 350 pcm	~ 650 pcm
Neutrons lifetime duration	~10^{-7} s	~10^{-4} s
Doppler reactivity feedback	~ - 1 pcm/K	~ - 2 à 3 pcm/K

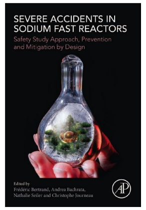


Safety features of a SFR (compared to PWR): coolant aspects

- **Chemical Reactivity, sodium is a powerful reducing agent that reacts with:**
  - **Air** (sodium fires)
  - **Water** (energetic reaction producing hydrogen)
  - **Oxide fuel**, to a lesser extent

- **Thermal-hydraulics**
  - Tendency to generate **hot or cold thermal shocks** (high heat transfer coefficient)
  - **Thermal fatigue** (particularly in the core cover plug)
  - **Thermodynamic interaction** with molten fuel and steel (steam explosion)

- **However, sodium does not only have drawbacks in terms of safety — it also offers advantages:**
  - The reactor operates at **atmospheric pressure**
  - **High heat removal capability**, including under **natural convection**
  - Possibility of using the **atmosphere as a cold sink** to remove residual power
  - **Ability to retain certain fission products**, such as iodine and cesium
  - **Post-accident cooling** and **in-vessel retention**



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CHAPTER 2

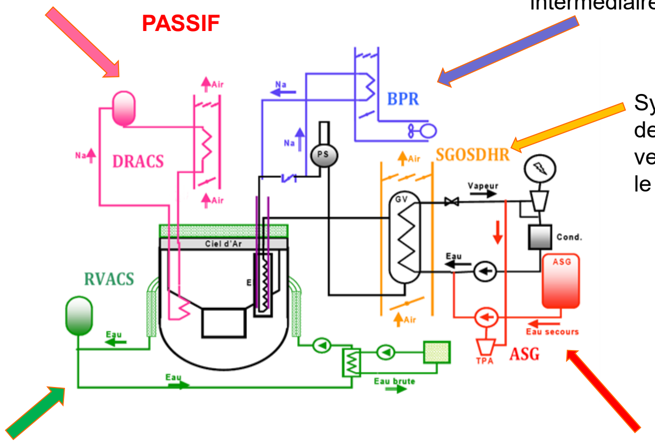
Safety approach for SFRs

- **Prevention of Severe Accidents**
  - Improving the reliability of reactor shutdown systems (normal, diversified, and passive)
  - Inertia of pumps and favorable natural behavior
  - Enhancing the reliability of the decay heat removal function
  - Demonstration using a defense-in-depth approach and a simplified probabilistic safety assessment (PSA)

Boucle **Na- Air** dans le circuit primaire

PASSIF

Boucle **Na- Air** en dérivation du circuit intermédiaire

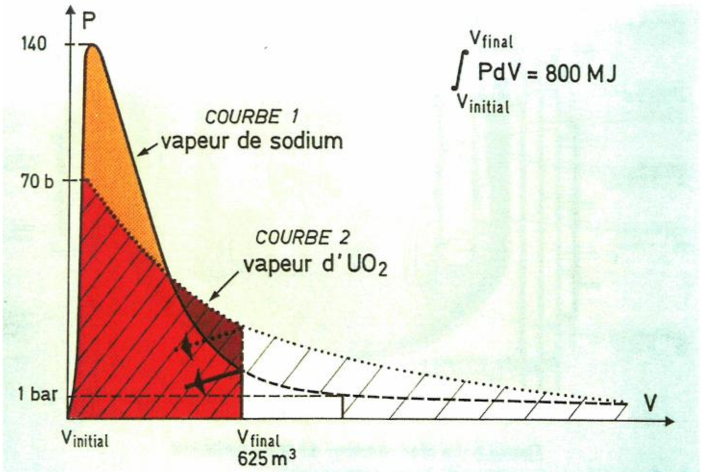


Système de refroidissement des GV par ouverture des ventelles et circulation **d'air** le long du GV

Refroidissement de l'enceinte primaire (cuve + dalle) par serpentins d'**eau**

ASG : Alimentation de secours en **eau** des GV par Turbo Pompes

- **Study of Consequences and Justification of Reactor Vessel Design**
  - **Assessment of mechanical energy**
  - **Reactivity ramp** associated with the **compaction of two half-cores**
  - Followed by analyses based on **scenarios for larger reactors (SPX)**



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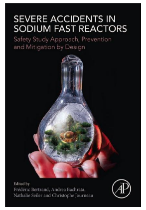
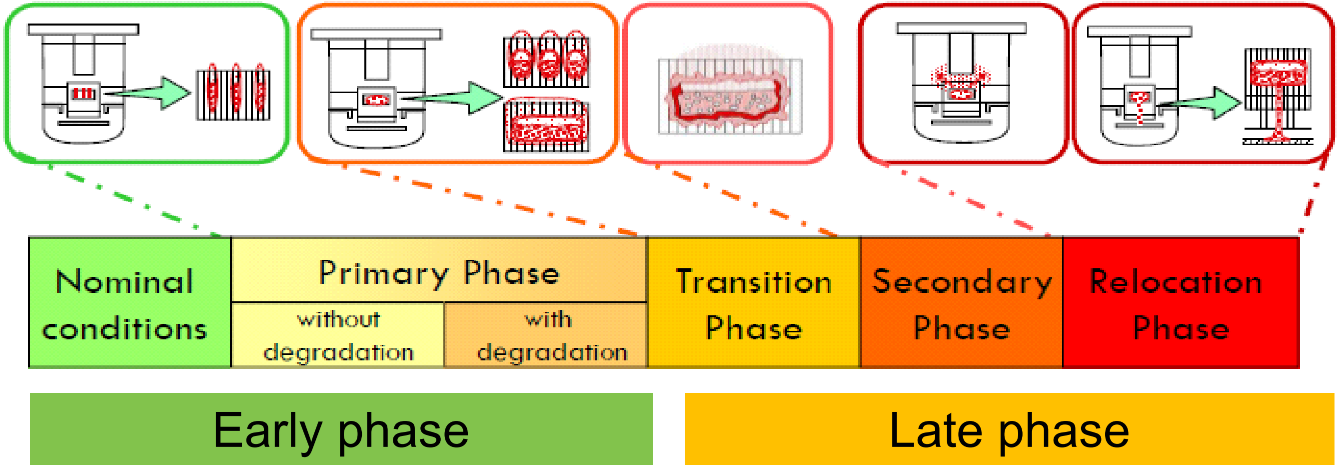


Severe accident initiating sequences and accident progression

- **Unprotected transients**: shutdown systems not efficient enough, failed, or activated too late
  - **Reactivity insertion** (Unprotected Transient OverPower, UTOP)
  - **Loss of primary flow** (Unprotected Loss of Flow, ULOF)
  - **Local melt accidents leading to generalized melting** (Unprotected Sub-Assembly Fault, USAF)
- *Note: the loss of the cold source, given the practical elimination of the DHR function loss, is not an initiating sequence of a core meltdown accident.*

- **Accident progression**
  - Mainly governed by core reactivity effects
  - Short time scale



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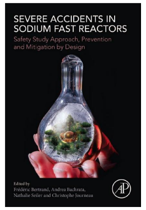

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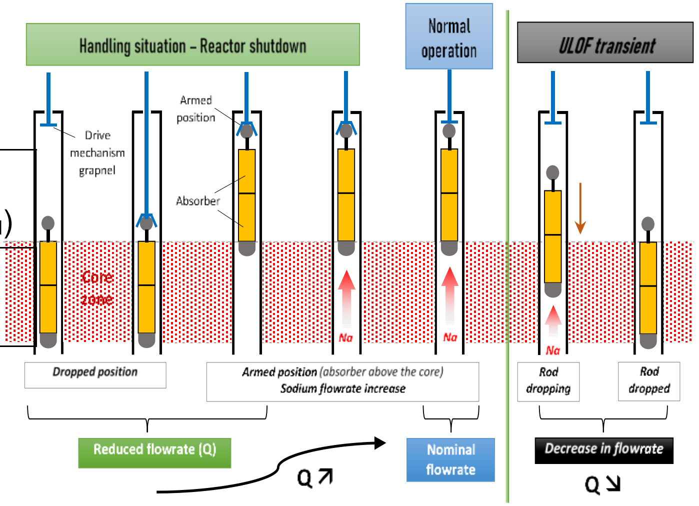
Design options in various projects for severe accident prevention

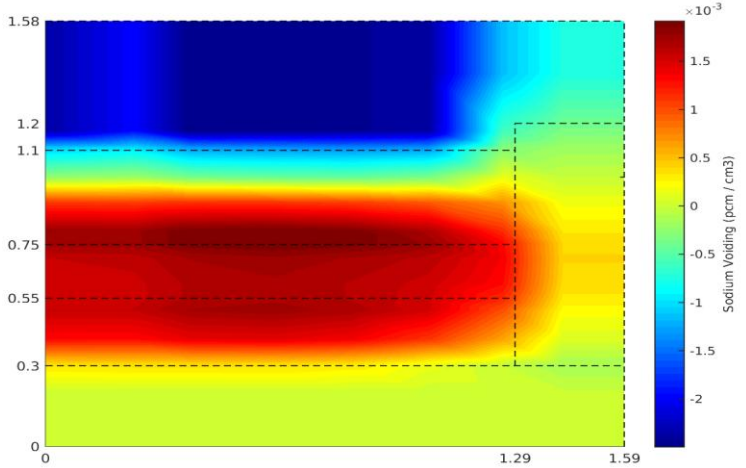
- After **Superphénix**, up to **Astrid**, reactor concepts were improved with respect to the **prevention and mitigation of severe accidents (SA)**.
- An **additional lines of defense** (some already existed, such as the **pump inertia** of SPX, which helped prevent boiling, for example) have been implemented to **prevent severe accidents** (e.g., **hydraulic or Curie-point control rods**).
- **Core stability near its operating point:**

$$\delta\rho = K\delta T_i + G\delta\Delta T_c + H\delta P_c + \delta\rho_{ext}$$

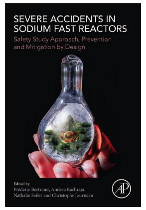
	K (pcm/K)	G (pcm/K)	H (pcm/%P_N)
ASTRID core – (end of equilibrium cycle)	-1.68	-0.69	-4.27



- The **core is designed to minimize reactivity potential** (for example, Astrid's **low-void and low reactivity swing core**).



- Severe accident scenarios that could lead to **major, uncontrollable consequences** not manageable by design are **identified during the design phase** and are **subject to design provisions** allowing their **practical elimination** (for example, ingress **of a large quantity of gas through the core**).



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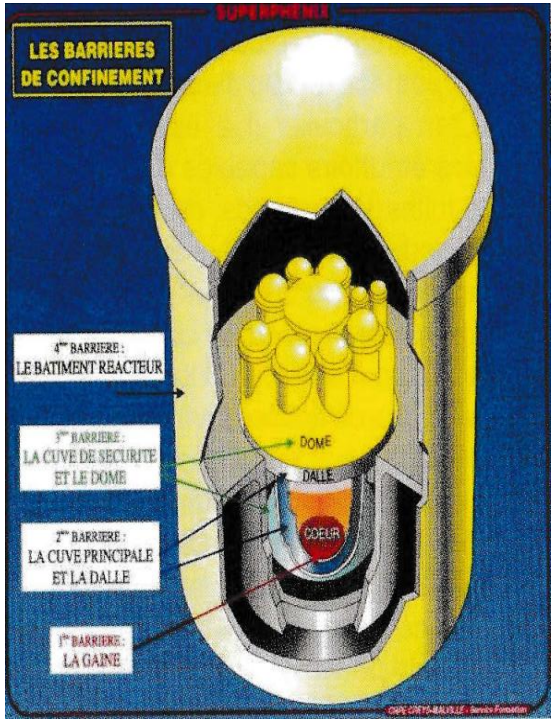
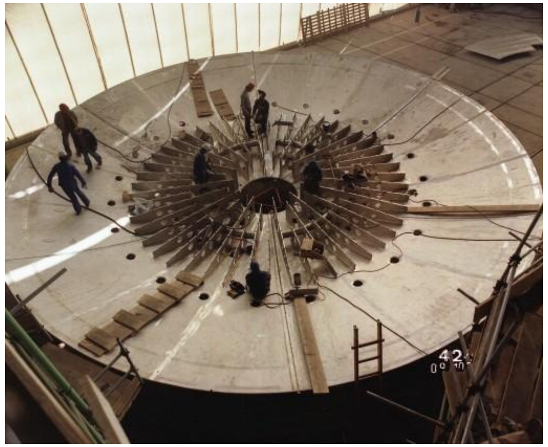
Design options in various projects for severe accident mitigation

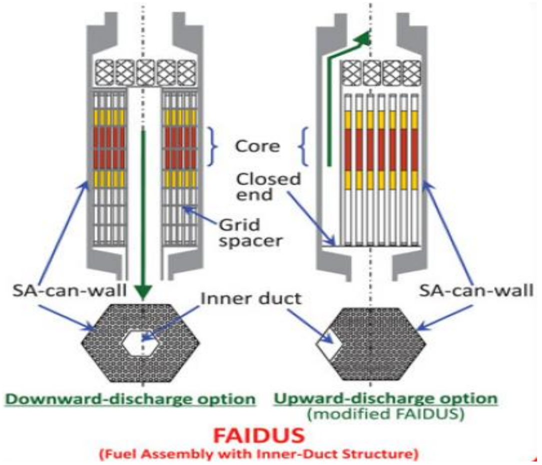
- **Confinement barriers**, and in particular the **main vessel**, tested with **MARS**

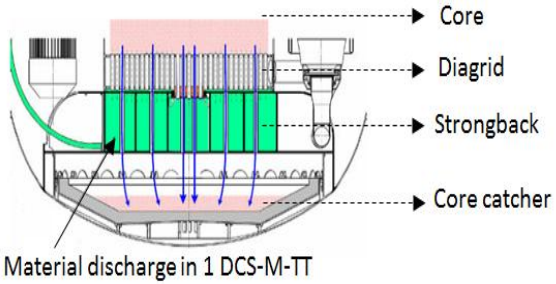


- **Core catcher** to ensure **retention of materials within the vessel**



- **Fuel relocation devices** outside the core zone to **reduce reactivity**





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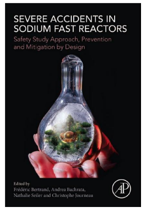

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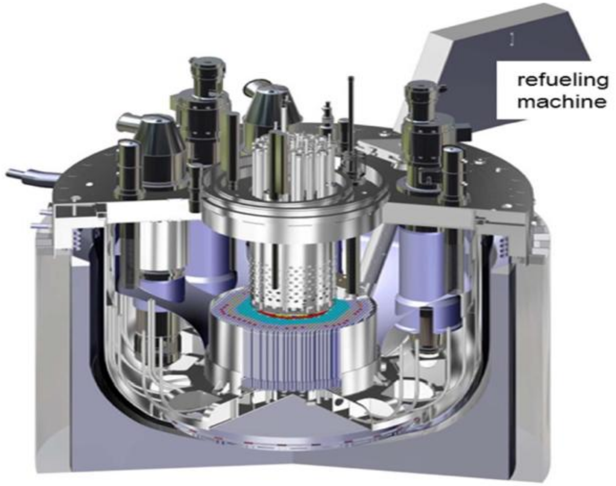
CHAPTER 3 (F. Bertrand, S. Kubo)

Design of ASTRID and JSFR



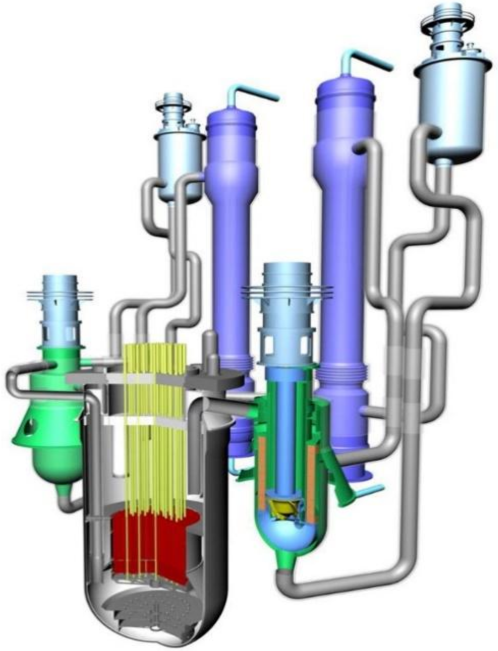
Overview of ASTRID and JSFR design

Frédéric Bertrand[1], Pierre Sciora[1], Christophe Journeau[2], Thiery Beck[3], Bernard Carluec[4,a] and Shigenobu Kubo[5]

[1]CEA, DES, IRESNE, Reactor Studies Department, Saint-Paul-lez-Durance, France; [2]CEA, DES, IRESNE, Nuclear Technology Department, Saint-Paul-lez-Durance, France; [3]CEA, DES, IRESNE, Fuel Studies Department, Saint-Paul-lez-Durance, France; [4]FRAMATOME, Technical & Engineering Division, Lyon, France; [5]Oarai Research and Development Institute, Japan Atomic Energy Agency, Oarai, Ibaraki, Japan

- **Chapter contents (43 pages):** For **ASTRID** and **JSFR** design:

- The core and its in-core prevention and mitigation systems
- The reactor block, the decay heat removal systems (EPuR), and the core catcher
- The intermediate circuit and the energy conversion system
- The reactor building



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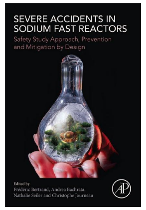
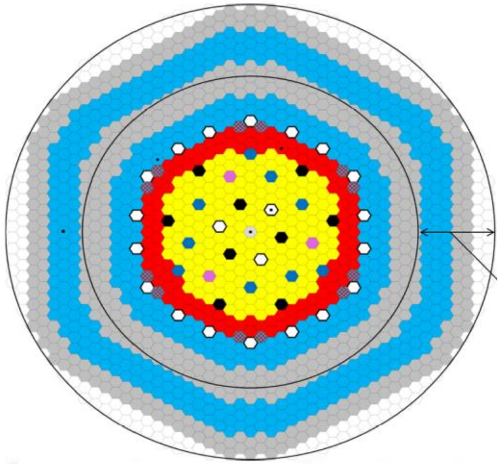
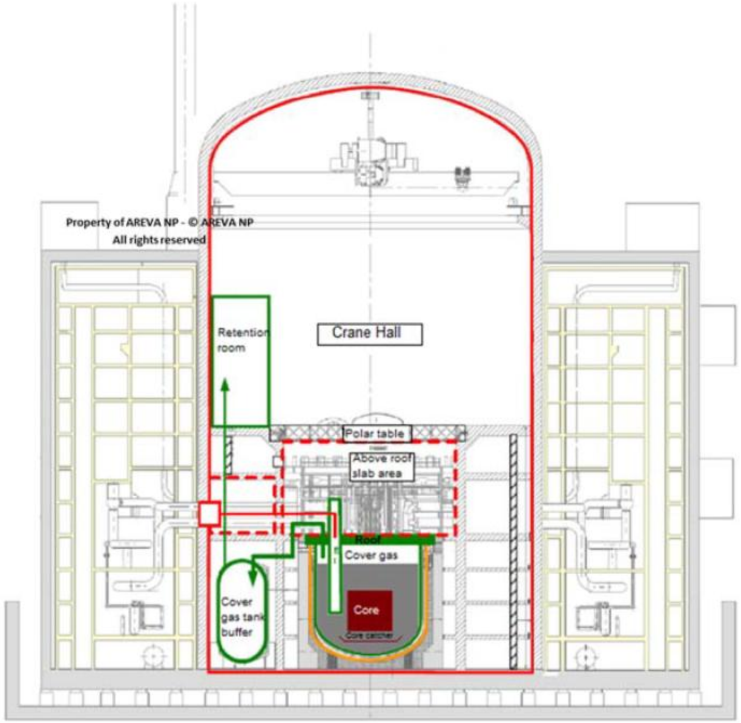
Overview of ASTRID design



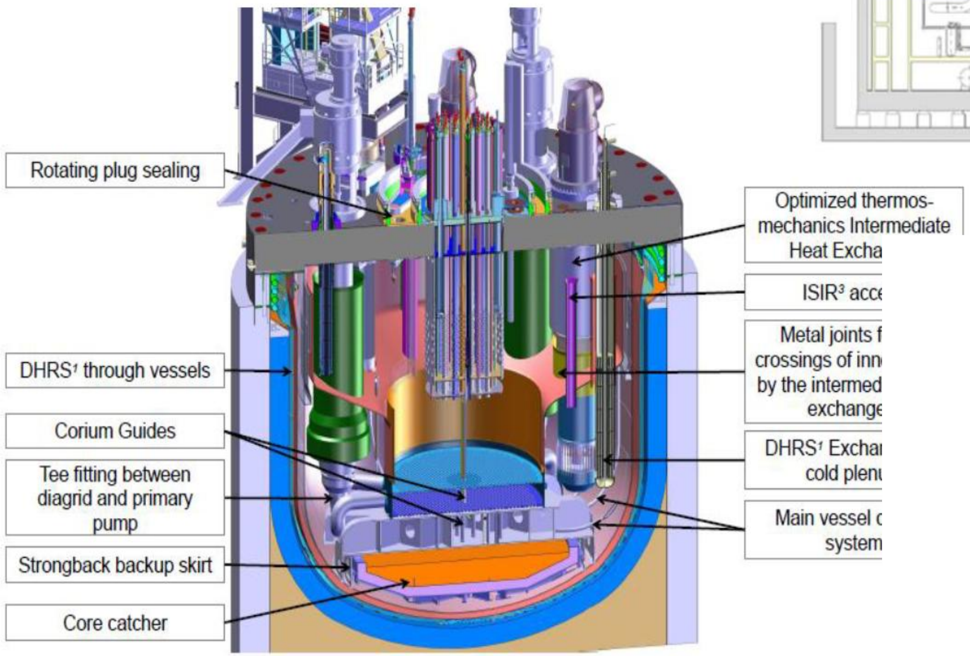
- Low void worth core



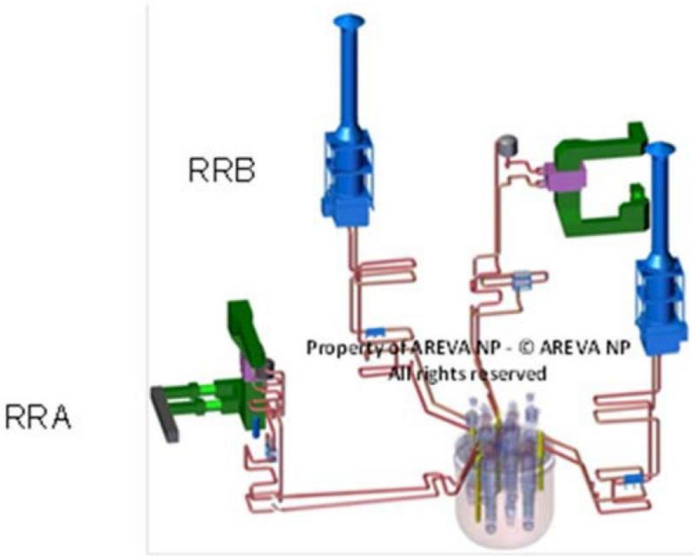
- Reactor building



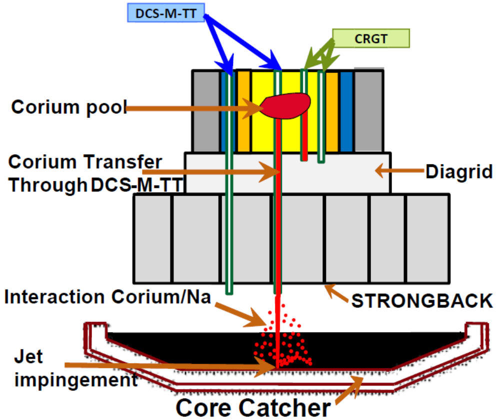
- Reactor vessel



- DHR systems



- Core catcher and transfer tubes (DCS-M-TT)



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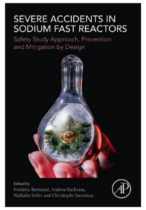
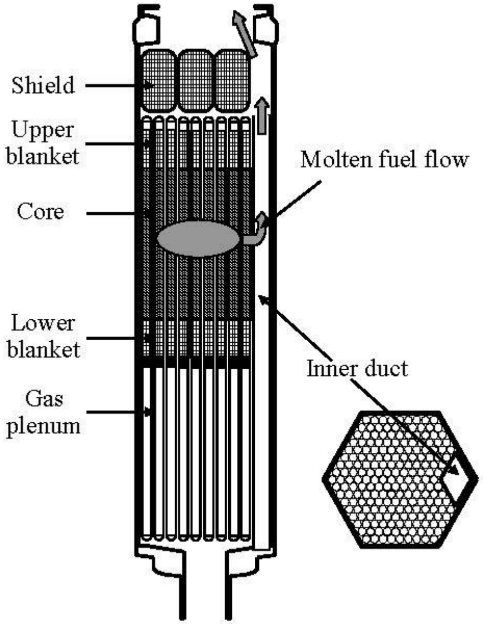
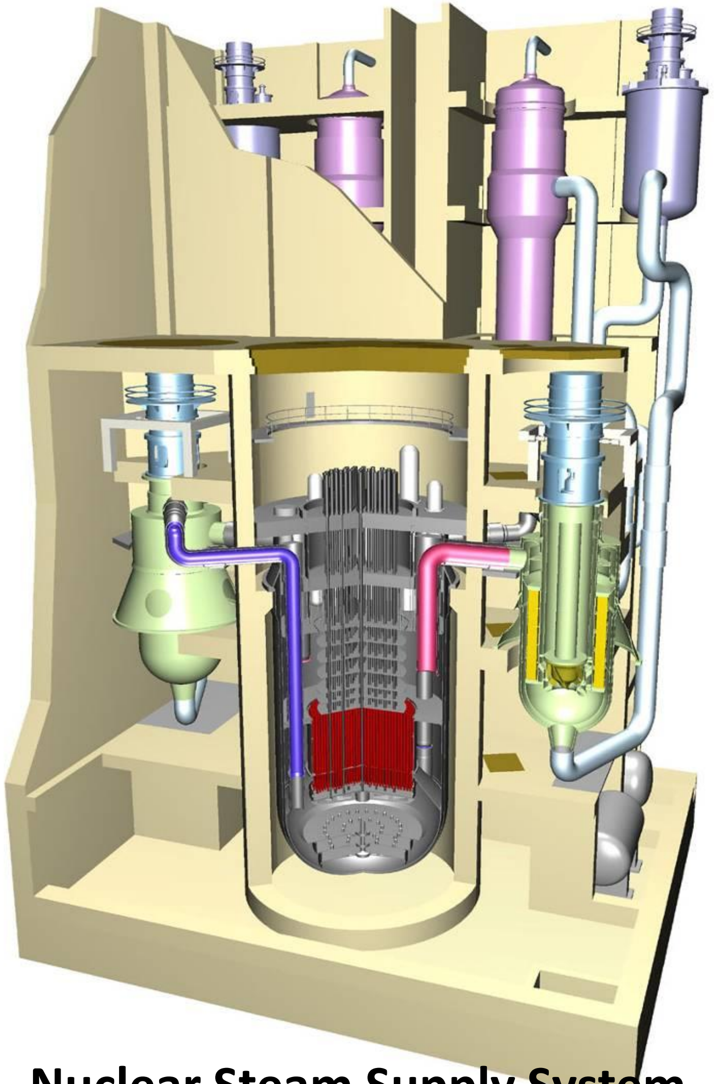
Overview of JSFR design

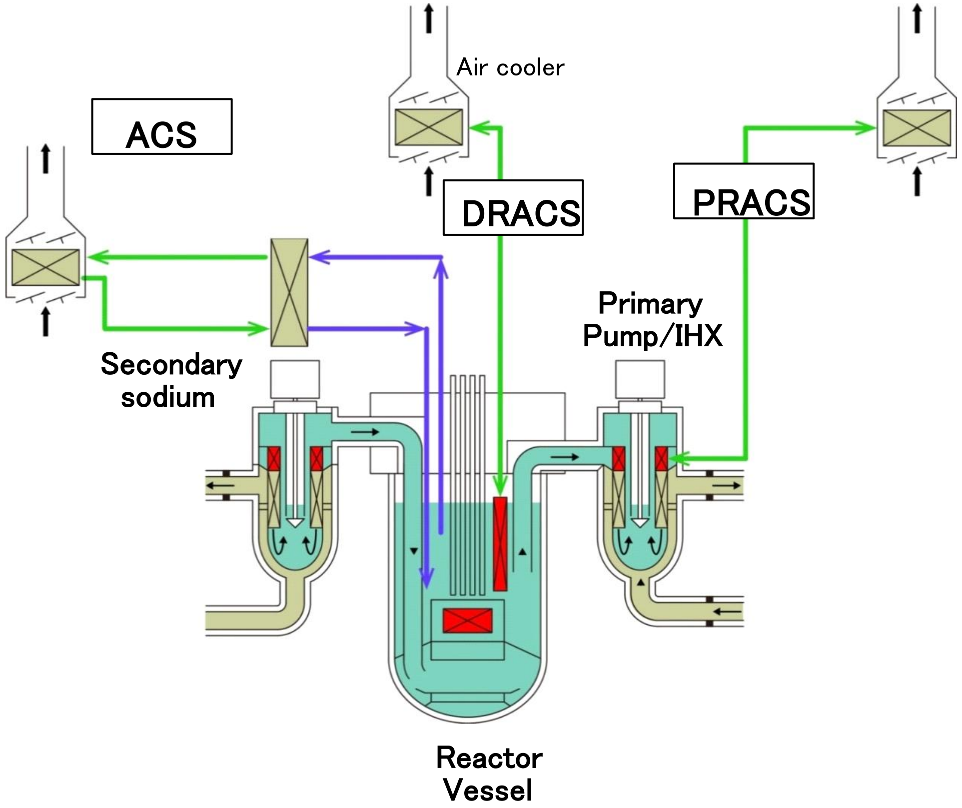




FAIDUS:
Fuel Assembly
with Inner DUct Structure



Nuclear Steam Supply System



DHR Systems

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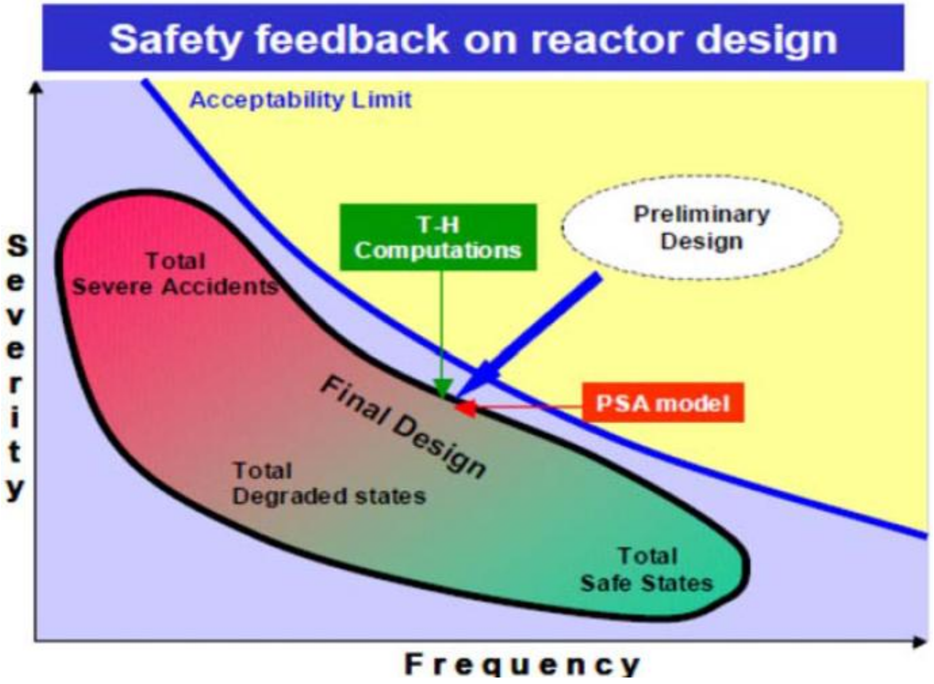
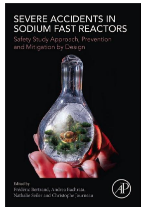
CHAPTER 4 (F. bertrand)

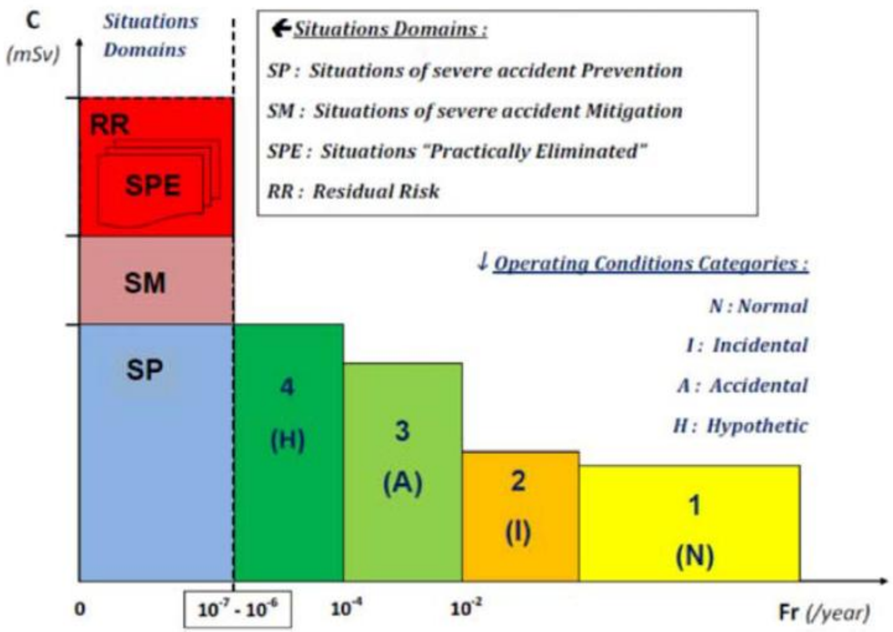
Approach for study and design of mitigation devices



Severe accident scenario consideration and study methodology for mitigation implementation by design proposed by CEA

Frédéric Bertrand
CEA, DES, IRESNE, Reactor Studies Department, Saint-Paul-Lez-Durance, France





Chapter contents (18 pages):

- General elements on the safety approach and description of severe accident sequences
- Study methodology supporting the prevention and mitigation approach
- Approach based on natural behavior studies, degraded states, and scenarios
- Feedback on design

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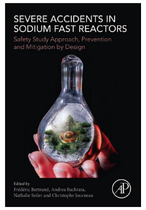
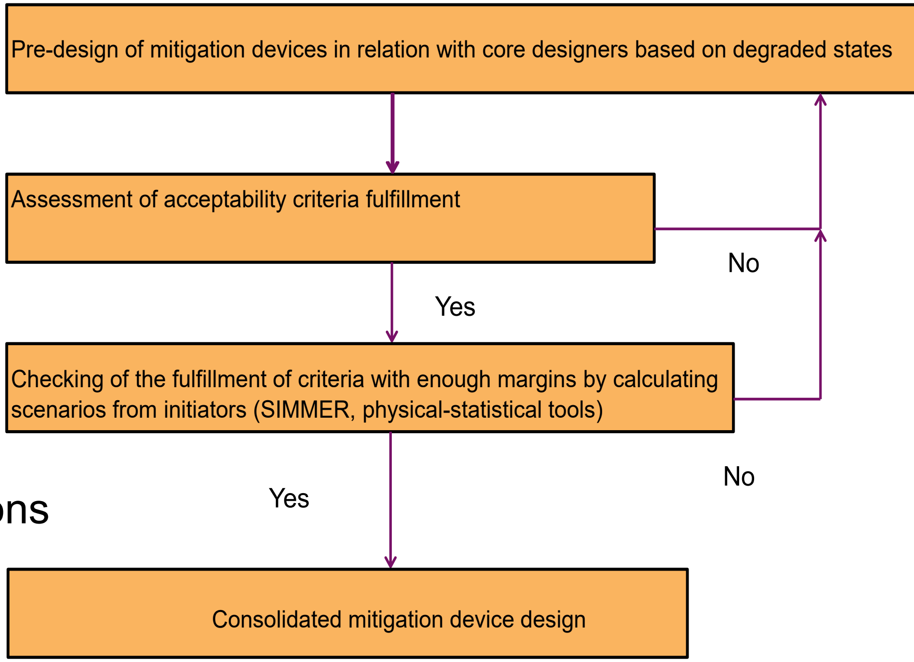
CHAPTER 4



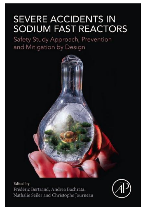
Design and study approach

- **Selection of mitigation options and principles:**

- **Identification of the nature and magnitude of effects** induced by the **accident under natural behavior:**
  - Energy, power
  - Relocated mass and distribution in the vessel
  - Etc.
- **Design of mitigation devices:**
  - Calculation of accident progression consequences starting from unfavorable degraded core states (pessimistic consideration of uncertainties through initial assumptions)
  - Parametric studies within the available design space (no dependence on the accidental sequence)
  - Pre-design based on the effects to be mitigated in unfavorable configurations
- **Design verification** based on scenarios starting from the initiator, to assess design margins while considering model uncertainties
- **Feedback on design?**



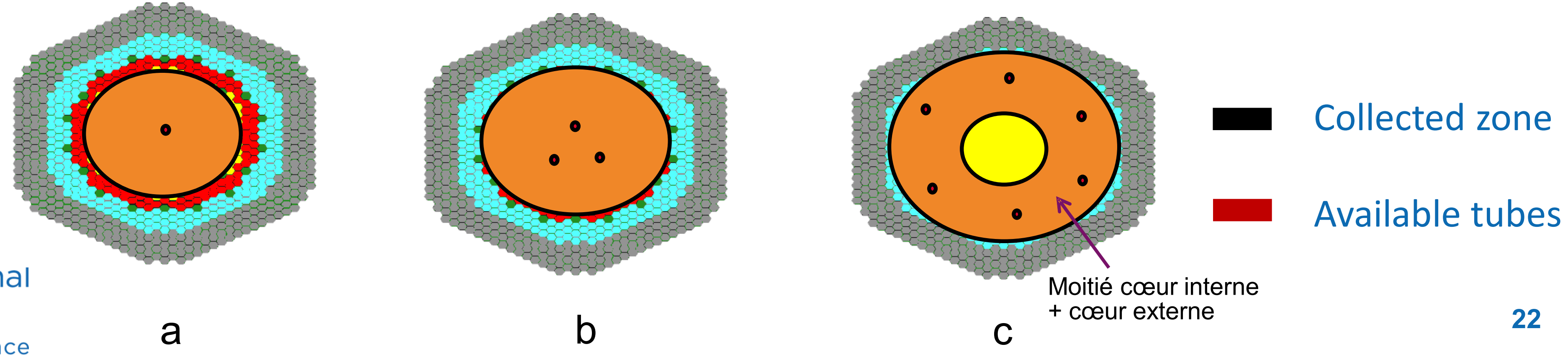
CHAPTER 4



Mitigation options and objectives

- **Transfer tubes** → limit the amount of fissile material in the core during the secondary phase and prevent or limit thermal energy deposition from a power excursion.
  - *Example of design evaluation criteria:* mechanical energy to assess the effectiveness of the transfer tubes.
- **Reactor vessel, decay heat exchangers (EPuR), and reinforced slab** → increase the load/rupture margin.
- **No propagation of the accident** toward the internal storage (sufficiently fine tube network).
- **Molten material catcher** → subcriticality, cooling, and in-vessel retention.

Examples of degraded states considered for calculating the thermal erosion of the core catcher



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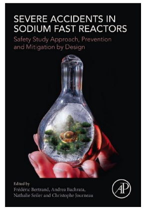
CHAPTER 5 (N. Seiler)

Calculation tools



Severe accident calculation tools used for severe accident and mitigation studies

Nathalie Seiler[1], Pierre Gubernatis[1], Laurent Trotignon[1], Emmanuelle Dufour[1], Rémi Clavier[1], Elena Martin-Lopez[2], Alberto Beccantini[3], Florence Drui[3], Andrea Bachrata[1], Pierre Sciora[2], Andrea Quaini[2], Frederic Daude[4], Serguei Potapov[4], Yoshitaka Fukano[5], Yoshiharu Tobita[6], Hidemasa Yamano[5] and Jean-Baptiste Droin[2]

[1]CEA, DES, IRESNE, Nuclear Technology Department, Saint-Paul-lez-Durance, France; [2]CEA, DES, IRESNE, Reactor Studies Department, Saint-Paul-lez-Durance, France; [3]CEA, DES, ISAS, Systems and Structures Modeling Department, Gif sur Yvette, France; [4]EDF R&D, ERMES, Palaiseau, France; [5]Oarai Research and Development Institute, Japan Atomic Energy Agency, Oarai, Ibaraki, Japan; [6]Research and Development Institute, Japan Atomic Energy Agency, Oarai, Ibaraki, Japan

Chapter contents (91 pages):

- Description of the **study methodology** using different types of tools (**estimators**, **physico-statistical**, **mechanistic**)
  - Description of **mechanistic** tools (for the study of steady-states and transients)
  - Description of **physico-statistical** tools

  For each tool are described:
  - Main characteristics, modeling assumptions, solved equations, numerical methods, etc.
  - Summary of its **validation**
- R&D perspectives in terms of digital platform development

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→ **The results of studies presented in Chapter 6 were obtained using these tools.**

Description of the study methodology involving different types of tools

Incorporate safety considerations during reactor design:

Phase 1: Preliminary design; Major developments possible

→ Order-of-magnitude estimates based on safety criteria estimators (Bertrand et al. 2021)

→ Identification of the strengths and weaknesses of each design option,

→ Selection of the main options in light of safety considerations,

Phase 2: Advanced design;

→ Fewer possibilities for change,

→ Numerous calculations to study the variability of results of interest (sensibility studies) with respect to a safety criterion for a given design:

Rapid physical SCT + advanced statistics + uncertainties on models, initial conditions & design → physico-statistical SCT.

Phase 3:Finalized design;

No further evolution; Design verification with highly validated SCT, including all physical phenomena, long CPU time → **mechanistic SCT**

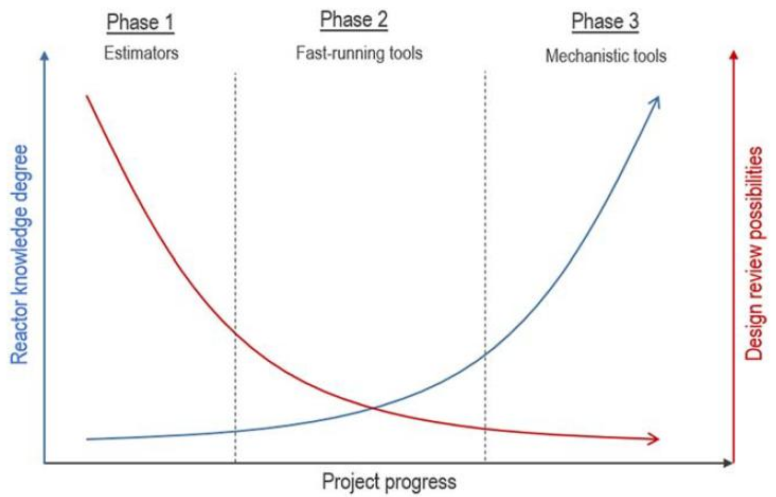

Figure 5.1 Schematic view of the reactor knowledge degree and the design change possibilities as a function of the design project progress and associated severe accident studies strategy.

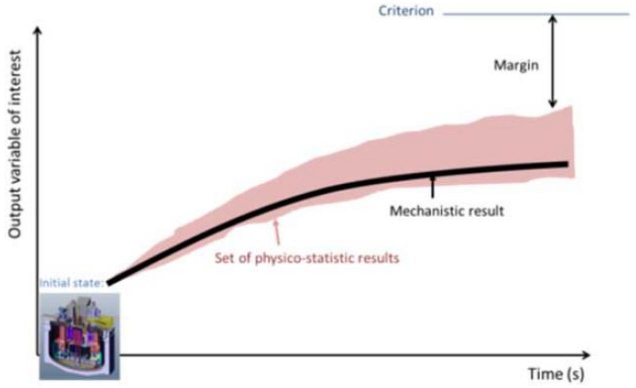

Figure 4.3 Combination of mechanistic and physico-statistical calculations for margin assessment.

SCT: Scientific Calculation Tools (validated)

Mechanical SCT (description and numerous validations)

Characterization of core **Neutronics**:
initial state: CST **TRIPOLI-4** (Monte Carlo) – Best-Estimate simulation
steady- states: CST **ERANOS** (deterministic approach)

Thermochemistry / Thermodynamic equilibrium:
CALPHAD - Thermophysical data for transients

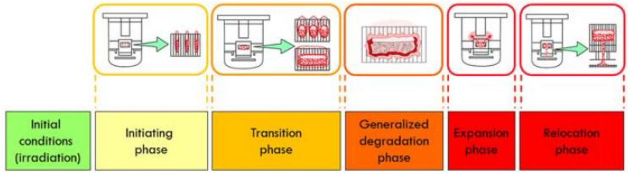

Figure 5.3 Successive stages of a typical severe accident in SFR.

Characterization of **accidental transients**:
'Calculation scheme':

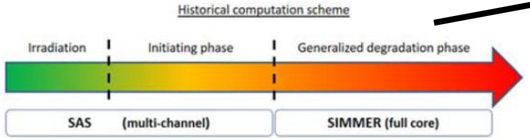

Figure 5.4 Historical scheme for calculating a severe accident transient in an SFR.

+ Coupling with **EUROPLEXUS** or **AUTODYN** (fast dynamics) for vessel deformation (safety criterion),
+ Coupling with **CONTAIN-LMR** for containment pressurization and release of fission products (safety criterion).

SAS: multi-1D, point kinetics

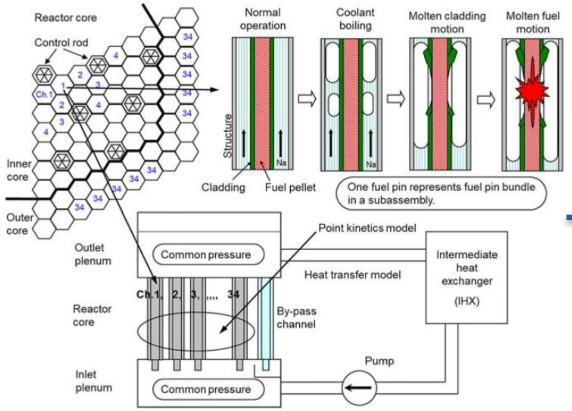

Figure 5.6 Conceptual image of SAS4A code (Fukano, 2015).

Difficult transition

SIMMER: Eulerian, 3D, 3D kinetic, multi-component

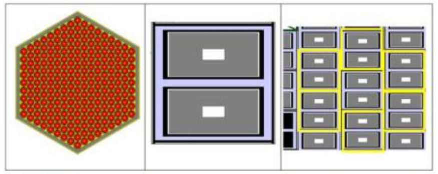
Figure 5.10 Example of a possible core and subassembly discretization approach in SIMMER.

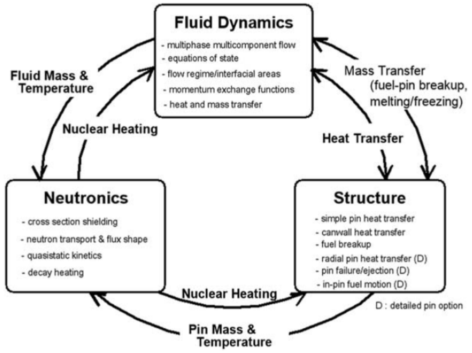

Figure 5.11 Interplay of different phenomenologies simulated by SIMMER in the course of a reactor accident transient (Yamano et al., 2003).

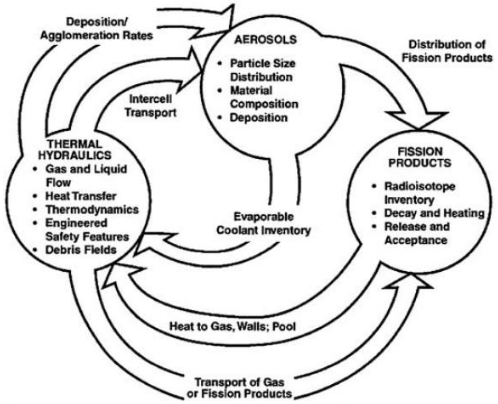

Figure 5.15 Three basic physical blocks treated in CONTAIN and feedback loops among them (Murata et al., 1997).

Physico-statistical SCT (description and validations)

Physico-statistical SCT: **only phenomena relevant to the accident sequence (derived from PIRT)**on the contrary to mechanistic OCS: all accident physics modeled → very fast and remain predictive

- A family of SCT developed by CEA:0D + 1D or even 2D modules + surrogate models & **Validation/experiments &SIMMER**, even taking experimental uncertainties into account
- 1 JAEA SCT: **Super-COPD** (debris beds) - 1D

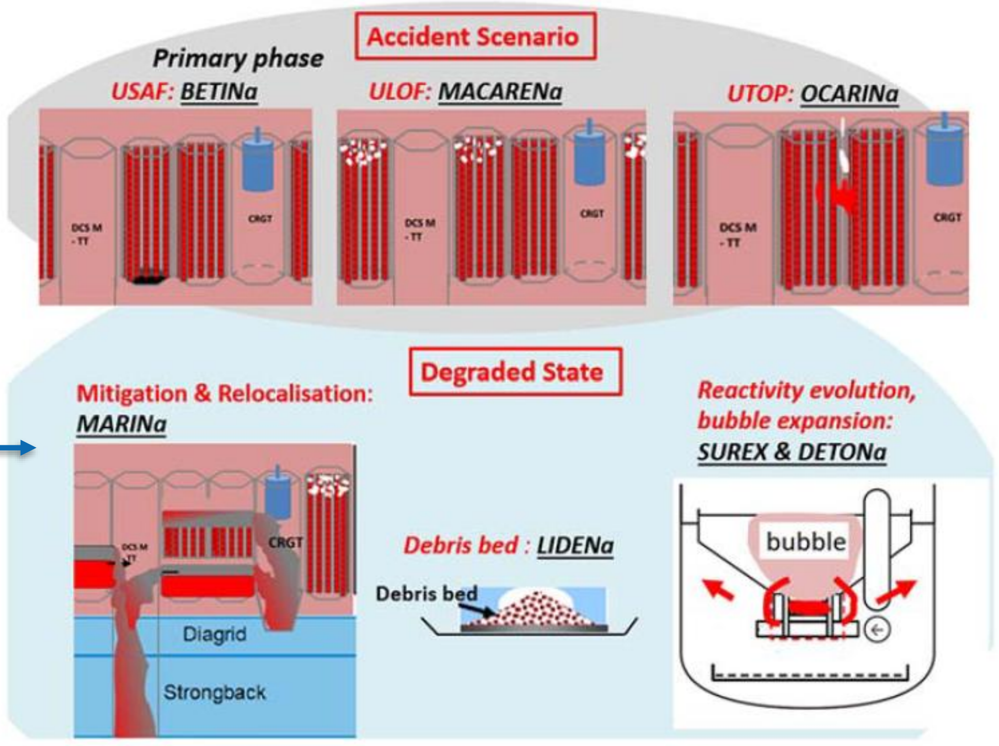

Figure 5.18 Overview of CEA parametrable tools.

Combining these tools to obtain a complete transient

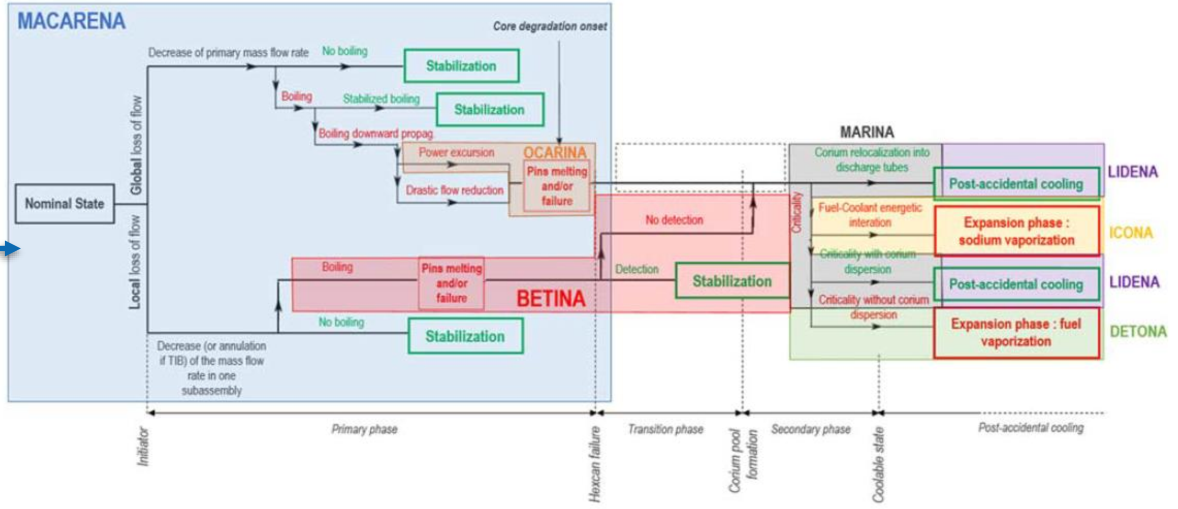

Figure 5.31 Illustration of the fast-running tools coupling for the entire ULOF sequence simulation.

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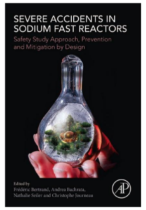
CHAPTER 6 (A. Bachrata)

Mitigation studies for ASTRID

Mitigation studies carried out for the French SFR demonstrator



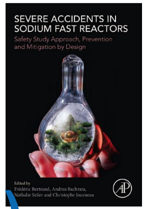
Andrea Bachrata[1], Nathalie Seiler[1], Frédéric Bertrand[2], Christophe Journeau[1], Jean-Baptiste Droin[2], Pierre Sciora[2], Andrea Quaini[2], Christine Guéneau[4], Stéphane Gossé[4], Matthieu Garrigue[4], Magali Zabiégo[1], Bernard Carluec[5,a], David Gentet[3], Lena Andriolo[6], David Lemasson[6], Shigenobu Kubo[7], Delphine Gerardin[6], Yuichi Onoda[7], Hidemasa Yamano[7], Kenichi Matsuba[7] and Kenji Kamiyama[7]

[1]CEA, DES, IRESNE, Nuclear Technology Department, Saint-Paul-lez-Durance, France; [2]CEA, DES, IRESNE, Reactor Studies Department, Saint-Paul-lez-Durance, France; [3]CEA, DES, IRESNE, Fuel Studies Department, Saint-Paul-lez-Durance, France; [4]CEA, DES, ISAS, Physico-chemistry and Material Research Department, Gif sur Yvette, France; [5]FRAMATOME, Technical & Engineering Division, Lyon, France; [6]EDF R&D, ERMES, Palaiseau, France; [7]Oarai Research and Development Institute, Japan Atomic Energy Agency, Oarai, Ibaraki, Japan

Chapter content (106 pages):

- -Natural behavior of the core, prevention and mitigation;
- -Mitigation strategy, accident families;
- -Design of mitigation devices (transfer tubes, recuperator, vessel, “sacrificial box”);
- -Studies on the degraded core (SIMMER, simplified tools), debris bed, materials, recriticality studies, jet, reactor building;
- -Scenario studies (SIMMER, ULOF study, mechanical energy, EUROPLEXUS, AUTODYN).

CHAPTER 6



Mitigation device purpose (resulting from mitigation strategy explained in Chap. 4)

•In most of configurations investigated for

→ transition and secondary phases of ULOFs

→ primary phases of UTOPs

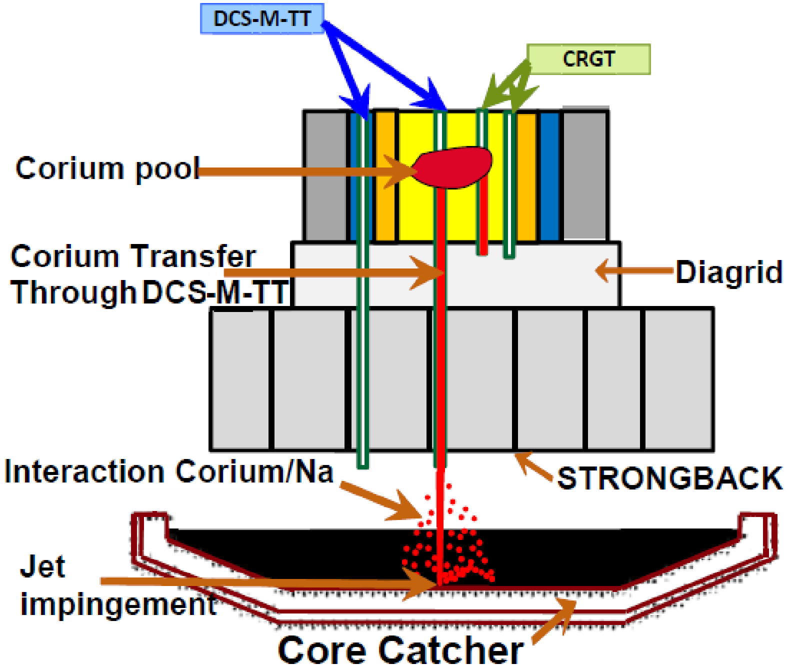
→ *preliminary studies have shown that mechanical energy released by fuel expansion is rather low (~10–40 MJ)*

•In some particular hypothetic cases leading to store more calorific energy in the core

→ *Mechanical energy released by fuel expansion could be higher*

•to cover these kind of unlikely situations, to control the accident scenario, it has been decided to:

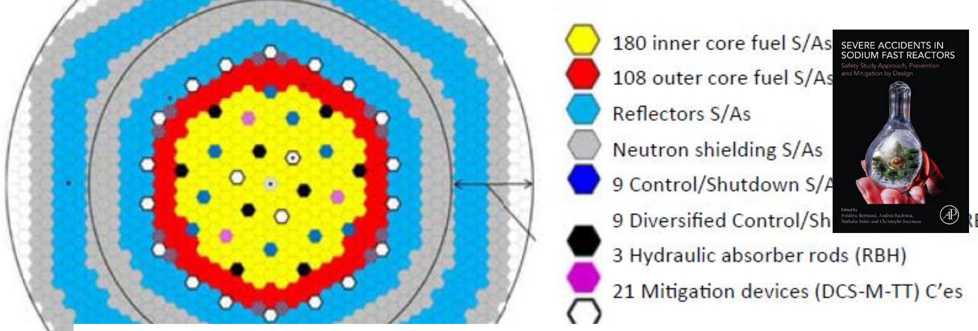
- *Limit the amount of fissile material in the core region thanks to molten material transfer tubes (DCS-M-TT)*
- *Install a core catcher into the primary vessel in order to collect and to cool-down materials relocated by DCS-M-TT*



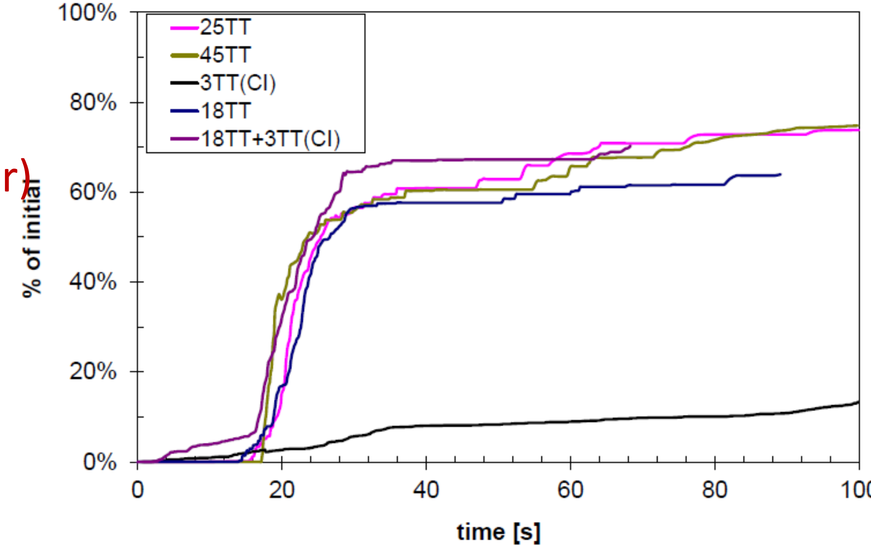
CHAPTER 6

Mitigation device preliminary design

Study of molten material relocation from a <u>core degraded state</u> through a large number of <u>parametrical</u> studies



Relocated mass vs DCS-M-TT number (SIMMER, steady power)



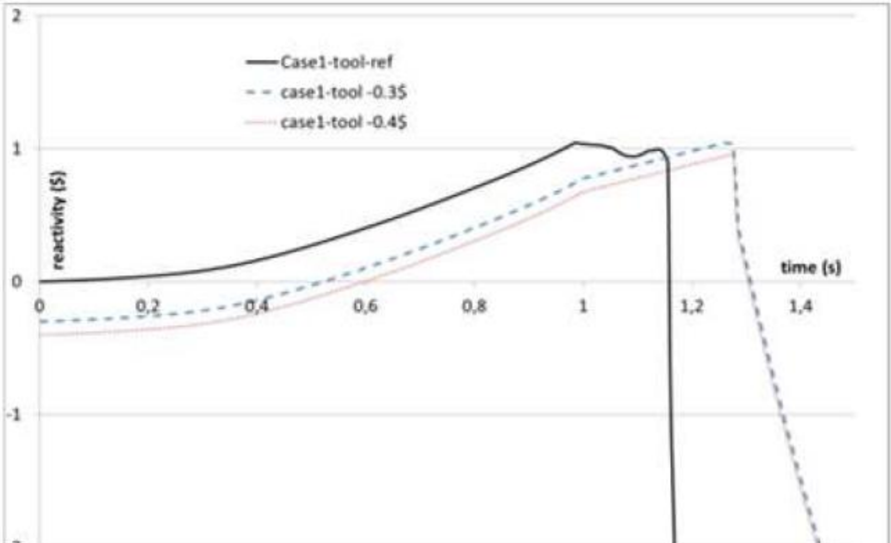
Study of design free parameters

- Number and location of DCS-M-TTs
- Fluid cross section in the upper part of DCS-M-TT
- Upper shielding geometry
- Fluid cross section at the diagrid elevation

Study of scenario evolution parameters (from degraded state)

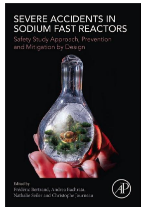
- Leaktightness of upper shielding
- Initial reactivity
- Initial material temperature and power
- Initial material distribution

Reactivity evolution vs. initial reactivity MARINa tool



→ Preliminary core design is settled (with DCS-M-TT)

CHAPTER 6

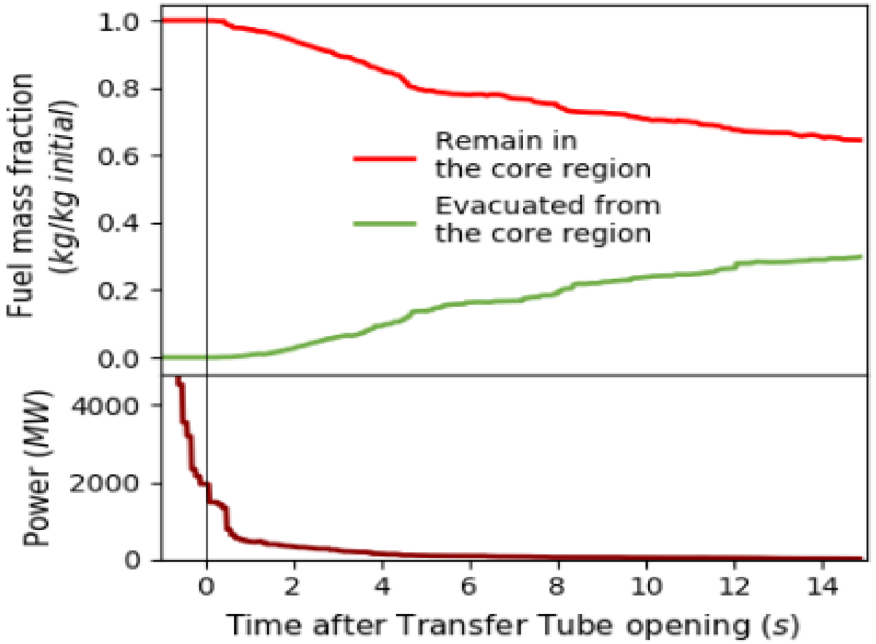


Mitigation device efficiency verification

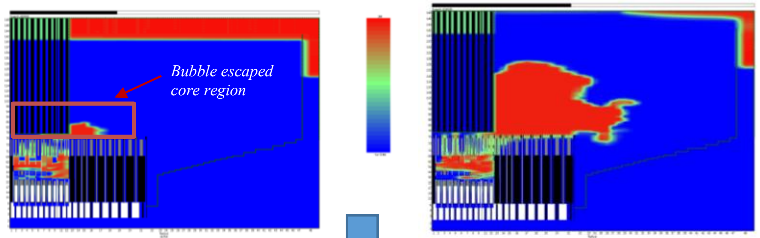
Study of accident sequences from their initiation (SIMMER mechanistic calculation)

Focus on transient after transfer tube opening (SIMMER calculation)

→ *The DCS-M-TTs enable to relocate 40 % of the fuel onto the core catcher in 15 seconds and to cancel the fission power*

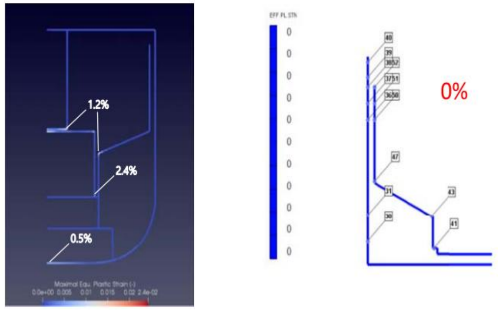


SIMMER



	Mechanical energy (MJ)
Core state from ULOF (SIMMER) evaluated by DETONA	
With DCS-M-TT	Zero
Without DCS-M-TT	~100 MJ

EPX

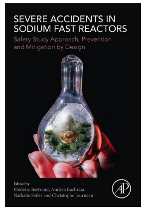


Mechanical energy with and without DCS-M-TT (DETONa, SIMMER/EPX)

→ *The DCS-M-TT reduces from one order of magnitude the calorific energy deposited in the core*
→ *Negligible mechanical energy with DCS-M-TT compared to transient inducing vessel deformation without them (but below safety criterion)*

CHAPTER 6

Mitigation device: to take away



Severe accident studies have confirmed the expected good natural behaviour of CFV core during the primary phase in particular in case of ULOF

→ mechanistic studies (SIMMER)
→ parametric studies and physico-statistical studies with design oriented tools

Material relocation mitigation devices (DCS-M-TT) have been designed in order to cope with a wide range of scenario evolutions (degraded state approach)

→ Range of adjustable design parameters have been investigated to reach a core design including DCS-M-TTs
→ Their efficiency to mitigate secondary excursions has been preliminarily shown

Verification of design margin have been done taking into account scenario variability and uncertainties (both for DCS-M-TT and core catcher)

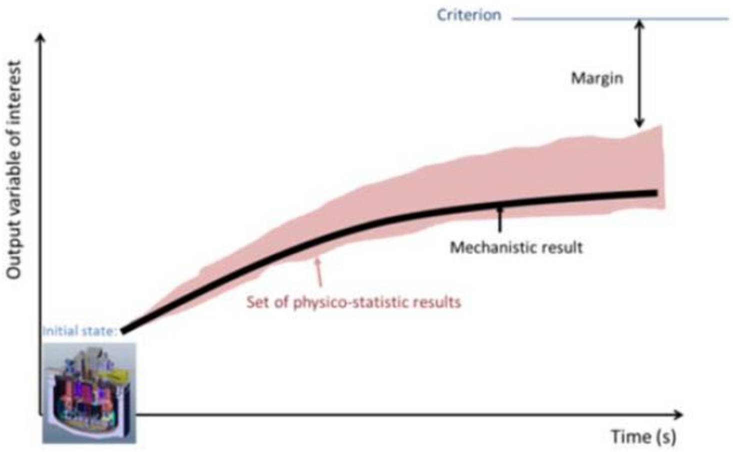
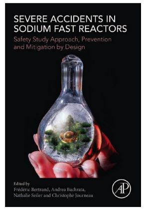


Figure 4.3 Combination of mechanistic and physico-statistical calculations for margin assessment.

CHAPTER 7 (C. Journeau)

Experiments



Experimental support studies for ASTRID mitigation device design

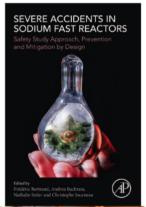
Christophe Journeau[1], Andrea Bachrata[1], Andrea Quaini[2], Rémi Clavier[1], Bernard Carluec[3,a], Shigenobu Kubo[4], Christine Guéneau[5], Matthieu Garrigue[5], Stéphane Gossé[5], Magali Zabiégo[1], Francois Charollais[1], Pierre Gubernatis[1], Emmanuelle Dufour[1] and Elena Martin-Lopez[2]

[1]CEA, DES, IRESNE, Nuclear Technology Department, Saint-Paul-lez-Durance, France; [2]CEA, DES, IRESNE, Reactor Studies Department, Saint-Paul-lez-Durance, France; [3]FRAMATOME, Technical & Engineering Division, Lyon, France; [4]Oarai Research and Development Institute, Japan Atomic Energy Agency, Oarai, Ibaraki, Japan; [5]CEA, DES, ISAS, Physico-chemistry and Material Research Department, Gif sur Yvette, France

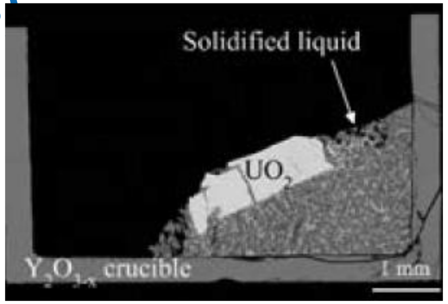
- **Chapter Content (56 pages):**
  - Reminder of the existing experimental database before the start of the ASTRID project (but with different safety objectives)
  - Out-of-pile tests (Thermochemistry / Discharge in transfer tubes / Corium-Sodium interaction / Liquid jet erosion / Thermohydraulics of a bath in the collector).
  - In-pile tests (EAGLE, SAIGA)
  - Lessons from these tests and remaining needs
  - 119 bibliographic references

CHAPTER 7

Out-of-pile experiments (ordered by topics unlike in other chapters)

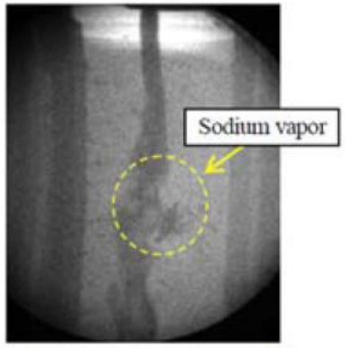


- **Thermochemistry**
  - UO_2-Steel-B_4C interaction (Japan, France, Czech Republic via EU project)
- **Discharge in transfer tubes**
  - Objective: avoid the most energetic sequences in the core
  - MELT tests (Japan) and EAGLE out-of-core tests (Kazakhstan)
- **Corium-Sodium Interaction**
  - Jet fragmentation: MELT tests (+ image processing with SPECTRA)
  - Sodium boiling: SERUA experiment under construction
- **Erosion by hot liquid jet**
  - Interaction of an unfragmented jet with the collector
  - JIMEC tests (KIT, Germany) and HAnSoLO (LEMTA)
- **Thermohydraulics of a bath in the collector**
  - LIVE-CC tests (KIT, Germany; ESFR-SMART and ESFR-SIMPLE projects)



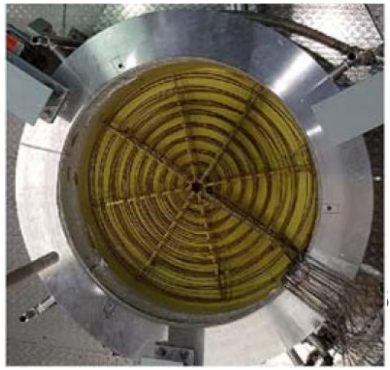
Garrigue et al.

COMETA



MELT

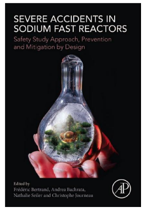
JIMEC



LIVE-CC

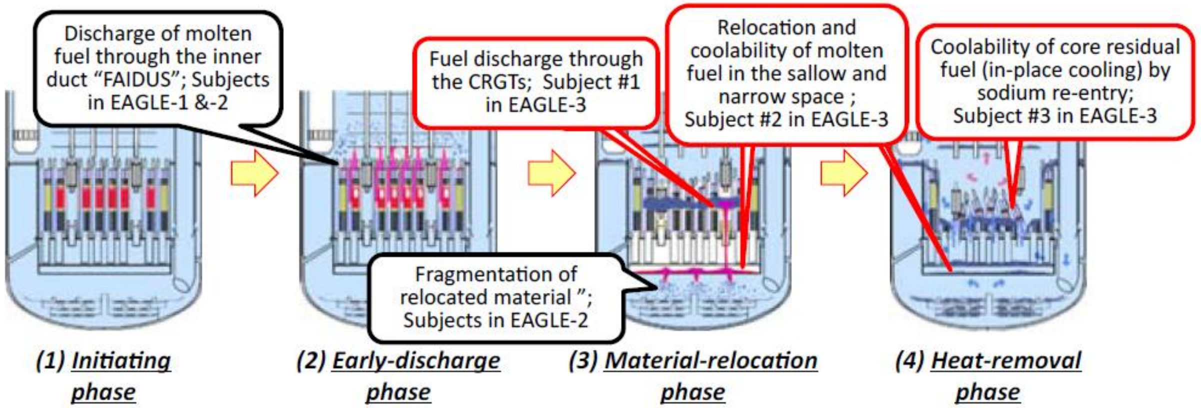
CHAPTER 7

In-pile experiments: 2 programs in the IGR reactor (Kazakhstan)



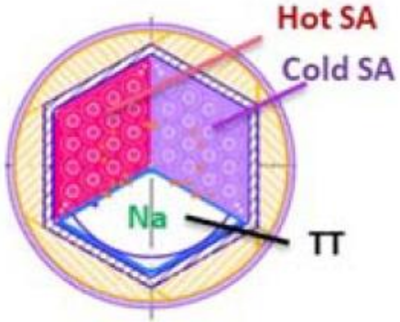
EAGLE (managed by JAEA)

- Validation of the SIMMER code on phases of corium transfer outside the core



SAIGA (managed by CEA)

- In preparation.
- Transfer of corium from a "bundle" of 12 pins to a tube filled with sodium.
- ULOF-type accident with the stop of sodium circulation after a brief steady-state.
- Test duration: 4-40 seconds, with innovative instrumentation.



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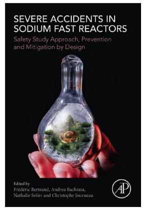
CHAPTER 7
Some lessons from these tests



- **Extensive use of experimental results** to validate the OCS, e.g. SIMMER (and SCONE in the future), and the thermodynamic database TAF-ID.
  - Tests in-pile: SCARABEE and EAGLE;
  - Out-of-pile tests: FARO-TERMOS (test campaign interrupted in the 1990s);
  - Validation of DNS tools for the thermalhydraulics of corium pools in core catchers through the LIVE-CC experiments.
- **Inspiration for mitigation measures**
  For example:
  - transfer tubes were inspired by relocations observed in CABRI and SCARABEE, as well as by lessons from the more recent EAGLE experiments.
  - The JIMEC and HAnSoLO tests initiated R&D on local thickening of core catchers beneath transfer tubes.
- **Contributions to the understanding of phenomena**

  For example:
  – Understanding and modeling of jet ablation phenomena following HAnSoLO;
  – Modeling of jet fragmentation by entrainment and vaporization of liquid sodium, thanks to X-ray visualizations from the MELT experiments.

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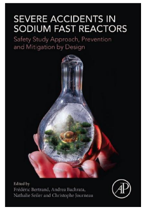
CHAPTER 8 (S. Kubo)

Severe accident mitigation approach and studies for JSFR and other recent SFRs

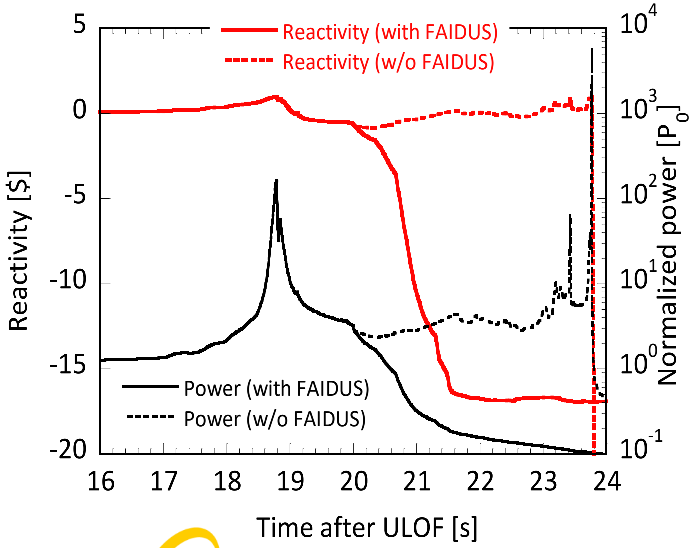
- **Chapter contents (20 pages):**
- JSFR safety study on severe accident mitigation
- BN-1200 (Russia)
- FBR-1&2 (India)

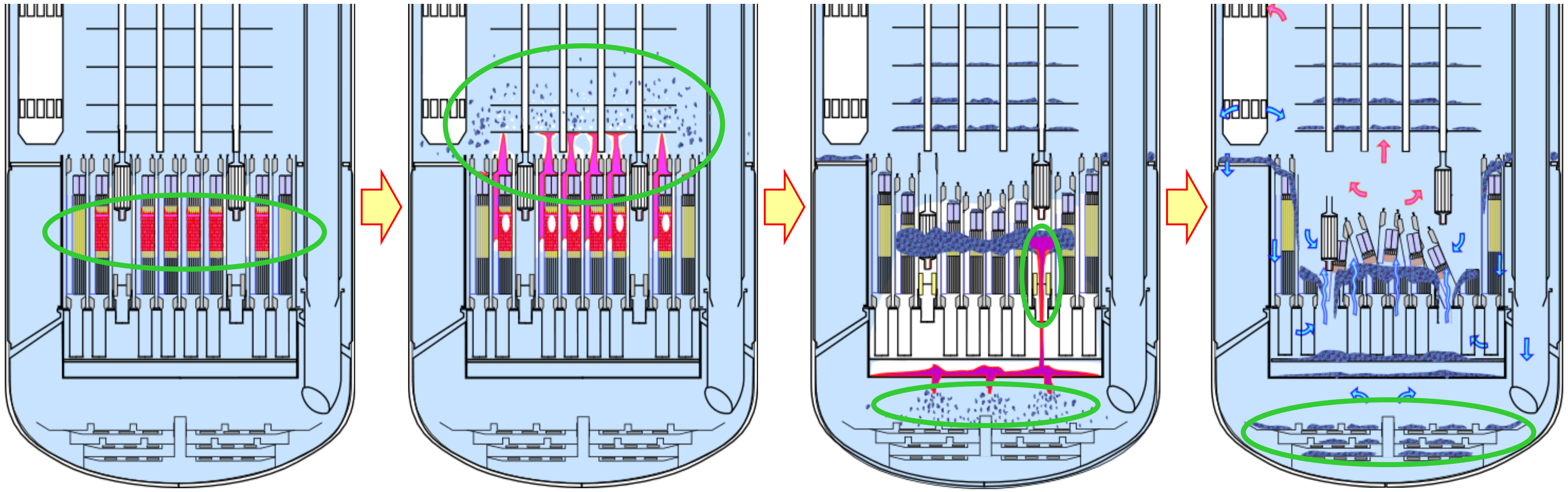
Severe accident mitigation approach and studies for JSFR and other recent SFRs

Shigenobu Kubo, Hidemasa Yamano and Yuichi Onoda
Oarai Research and Development Institute, Japan Atomic Energy Agency, Oarai, Ibaraki, Japan



Various measures are incorporated into design to avoid severe re-criticality and to achieve in-vessel retention.





Initiating Phase	Early-discharge Phase	Material-relocation Phase	Heat-removal Phase
Limit sodium void reactivity	**Molten fuel discharge duct (FAIDUS)**	**Fuel relocation through CRGT**	**In-vessel core catcher**

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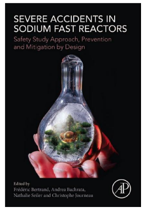
CHAPTER 9 (F. Bertrand)

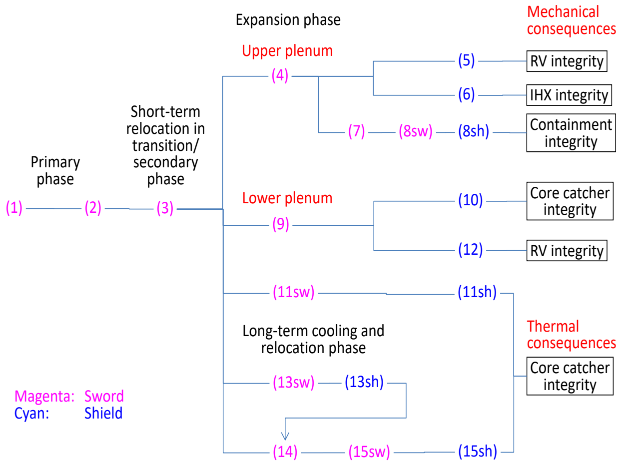
R&D needs

Further R&D needs

Frédéric Bertrand[1], Andrea Bachrata[2], Christophe Journeau[2], Nathalie Seiler[2] and Shigenobu Kubo[3]

[1]CEA, DES, IRESNE, Reactor Studies Department, Saint-Paul-lez-Durance, France; [2]CEA, DES, IRESNE, Nuclear Technology Department, Saint-Paul-lez-Durance, France; [3]Oarai Research and Development Institute, Japan Atomic Energy Agency, Oarai, Ibaraki, Japan





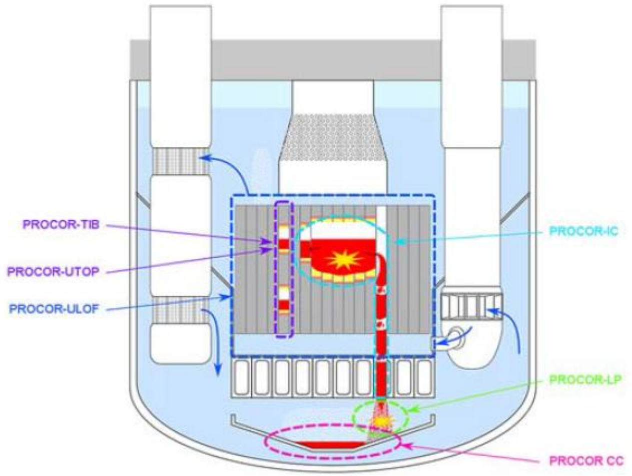
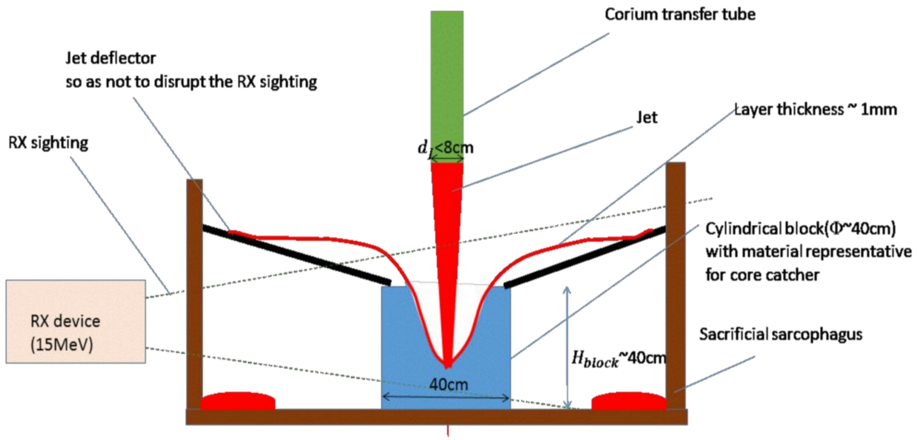


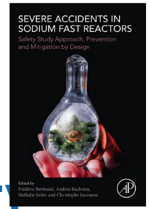
Figure 5.32 PROCOR-Na applications included in the development plan.



- **Chapter contents (31 pages):**
  - Methodology used to carry out a PIRT, presentation of results, and prioritization of R&D
  - R&D needs in simulation
  - R&D needs in experimentation

CHAPTER 9

Identification of additional R&D needs: phenomena identification and ranking tables (PIRT)



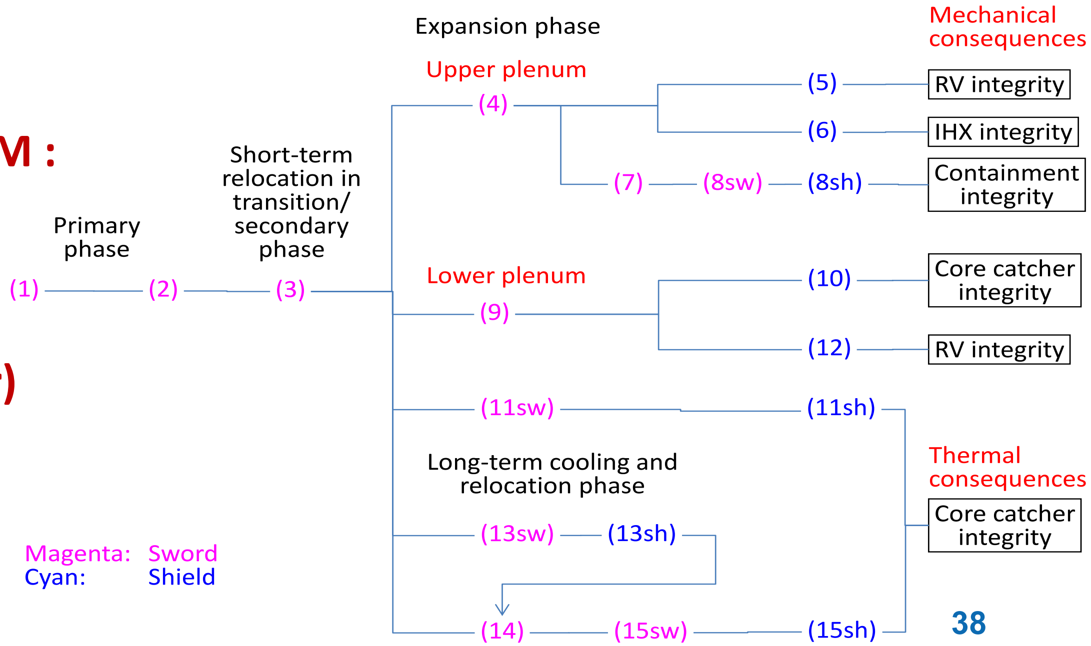
- **PIRT Methodology (PIRT carried out by CEA, JAEA, MFBR, and Framatome)**
  - ULOF, UTOP, and USAF have been addressed
  - Representation of event sequences using generic event trees
  - Hierarchization of elementary phenomena for each phase of the accident scenario according to their **importance (I)** and **knowledge (C)** → Overall score = **I × C**
  - Importance defined based on evaluation criteria called *Figures of Merit (FoM)*

Phase	No	Qualitative FOM in ULOF PIRT
P	(1)	Fuel SA coolability
	(2)	Power excursion
S	(3)	Core stored energy and fuel inventory
E	(4)	Core expansion behavior (upper plenum)
	(5)	Pressure build up behavior due to FCI (lower plenum)
	(6)	To keep function of primary coolant boundary
	(7)	To keep function of DHX
	(8)	Amount of ejected sodium mass and fission products
	(9)	To keep function of core catcher (Mechanical loading on core catcher)
	(10)	To keep function of core catcher (Core catcher integrity versus jet)
	(11)	To keep the in-vessel retention
L	(12)	Fuel inventory which can be retained in core region
	(13)	Criticality (Material retention in core region)
	(14)	Relocation onto the core catcher
	(15)	To keep function of core catcher (Material retention on core catcher)
	(16)	Criticality (Material retention on core catcher)

Two kinds of FoM :

sword/shied

(loading/barrier)

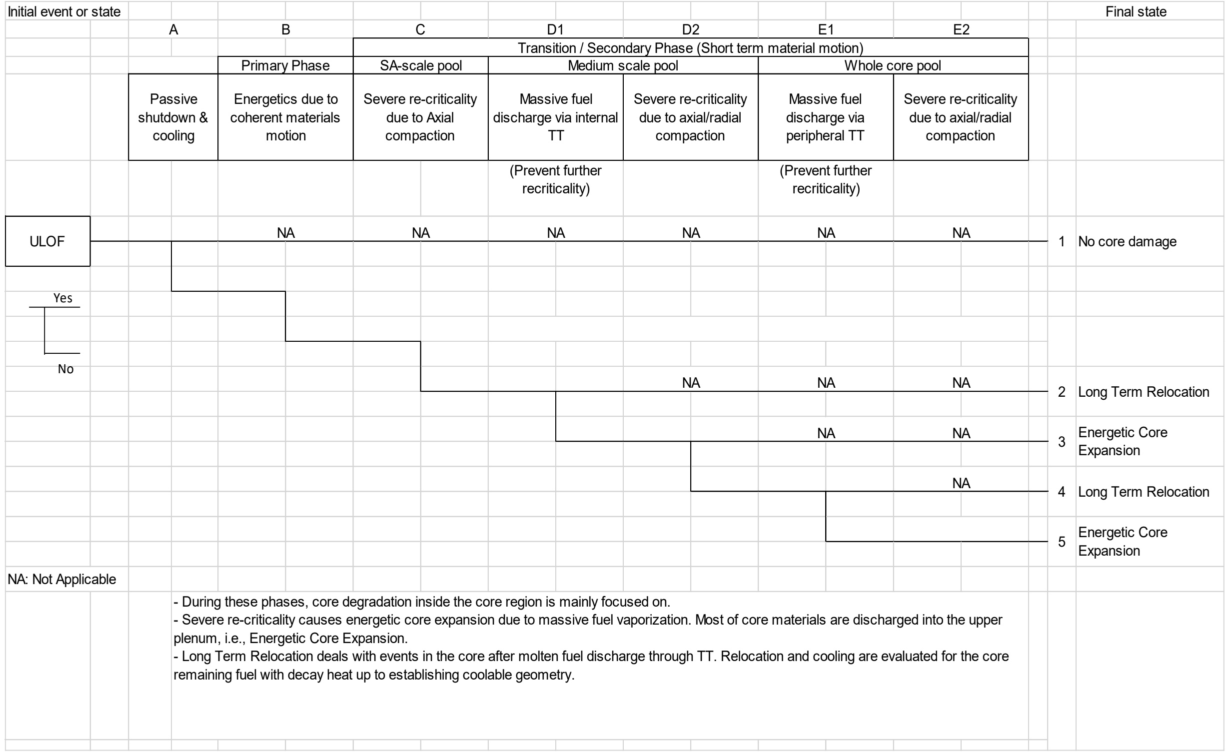


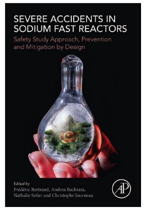
CHAPTER 9

Summary of PIRT Results (R&D Prioritization)

- **ULOF**
  - Axial relocation and compaction of fuel
  - Opening of DCS-M-TT
  - Material transfer within DCS-M-TT
- **UTOP**
  - Failure of fuel pins and fuel discharge
  - Rapid channel drainage due to FCI
- **USAF (less priority)**
  - Propagation to neighboring assemblies for a CFV core
  - Foaming phenomenon for irradiated fuel and its effect on radial propagation

- **Generic ULOF event tree (primary/transition phase)**





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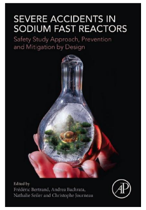
CHAPTER 9

R&D prospects for numerical plateform



Mechanistic SCT: Best-Estimate (BE) simulations

SIMMER V development: initiation and degradation phases

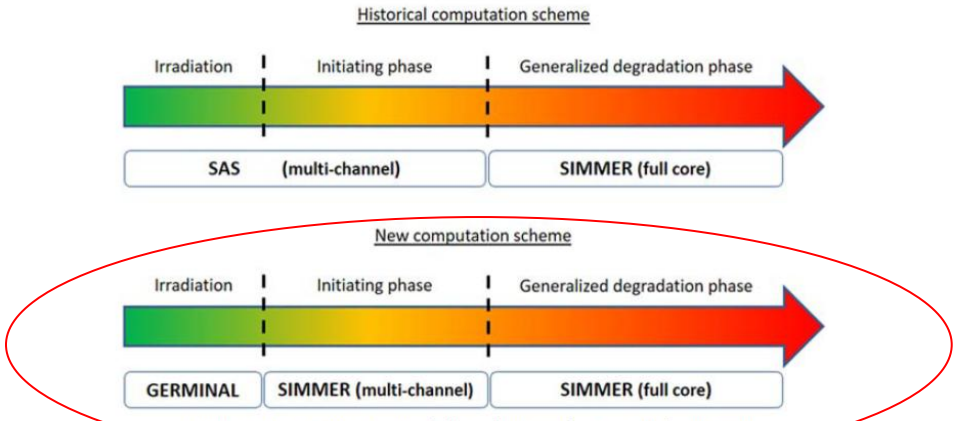


Figure 5.16 Comparison of the schemes, former, and new one.

SEASON : platform based on the coupling of validated multi-physics SCT

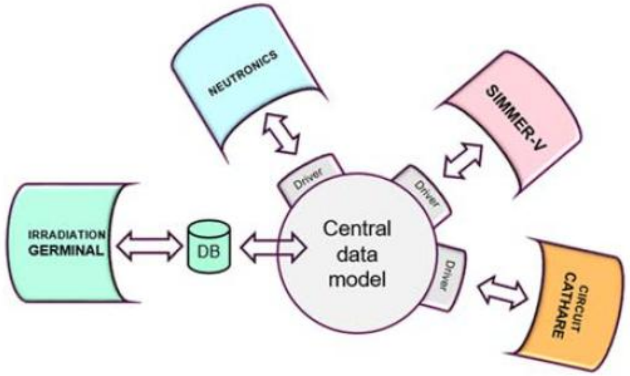


Figure 5.36 A schematic view of the satellite architecture of SEASON.

Physico-statistical SCT tools: BE Plus Uncertainties (BEPU)

Platform **PROCOR_Na** gathers the CEA's Physico-statistical SCT tools

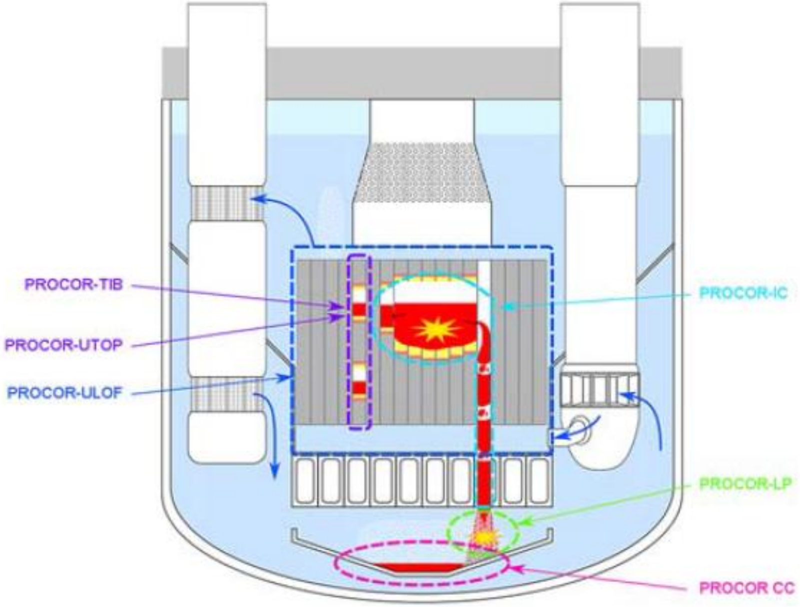


Figure 5.32 PROCOR-Na applications included in the development plan.

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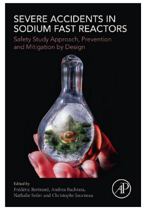
CHAPTER 9

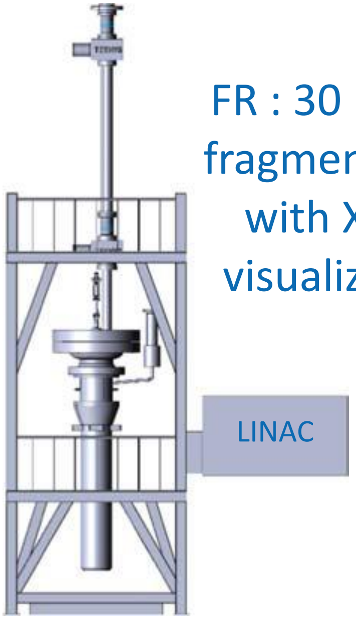
Experimental R&D needs



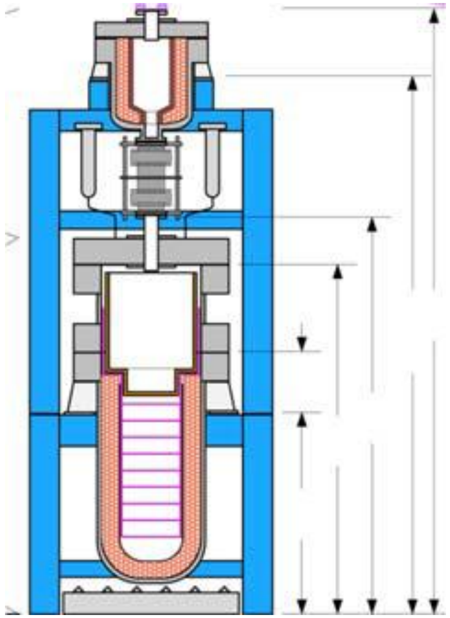
Despite the extensive historical experimental database, there remain unaddressed experimental needs, as safety requirements have evolved significantly since Generation II.

- **In-pile tests on pin degradation** (if, like CFV-type pins, they differ from those studied in SCARABEE)
- **Thermochemistry and thermophysical property tests** *(VITI)*
- **Prototypic corium–sodium interaction tests** to support the development of **SCONE: SAFeTY**

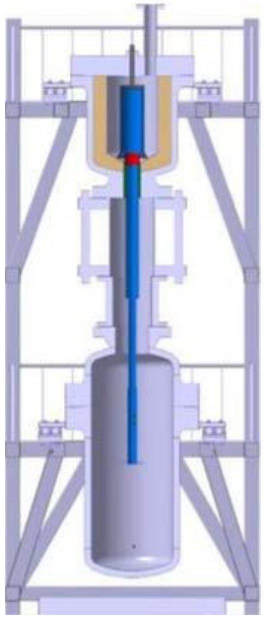
FOURNAISE furnace (ECM Technologies)



FR : 30 kg, Jet fragmentation with X-ray visualization

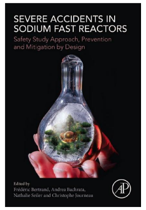


EXPLO : 100 kg vapor explosion



TR : Relocation in transfer tubes.

CONCLUSION (F. Bertrand)



Conclusion

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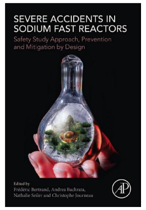
Chapter contents (15 pages):

- Summary of each chapter
- Main achievements and perspectives
- Concluding remarks



CONCLUSION

Concluding elements

Severe accident approach and its evolution

- **Evaluation of mechanical energy envelope** in early projects (core compaction)
- **Superphénix**: verification of design with a representative scenario (primary vessel sized for a scenario-independent mechanical energy), inclusion of a core-catcher
- **EFR**: improved reliability of automatic reactor shutdown to attempt exclusion of large-scale core melting (study of local blockage and ultimately ULOF), inclusion of a core-catcher
- **ASTRID**: enhanced prevention through passive systems, in-core mitigation devices, and a core-catcher designed with a top-down approach and degraded-state methodology, considering all sequence families
- **JSFR**: prevention by passive means, mitigation aiming at avoiding severe re-criticality

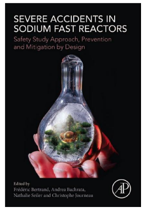
ASTRID and JSFR designs

JSFR:

- Loop-type reactor (750 MWe, later 1500 MWe) with a water/steam SCE
- **Prevention:** Curie point control rods
- **Mitigation:** FAIDUS, Control Rod Guide Tube, multi-stage core-catcher beneath the core

ASTRID:

- Integrated reactor (600 MWe) with a gas SCE
- **Prevention:** CFV core, hydraulic and Curie point control rods
- **Mitigation:** CFV core, DCS-M-TT system, top-down designed vessel, internal core-catcher



CONCLUSION
Concluding elements

Simulation tools

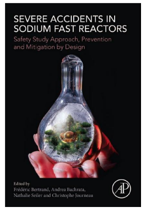
- **Static tools** for preliminary core design and equilibrium states in severe accident (SA) conditions
- **Fast computational tools** for studies supporting design and margin evaluation
- **Mechanistic tools** for design verification

Mitigation and accident scenario studies

- **ASTRID CFV core:** non-energetic behavior
- **Preliminary design of transfer tubes** for highly unfavorable cases and for guiding degradation progression
- **Transport of fission products (FPs)** toward the upper part of the reactor, **FCI** and **core-catcher erosion** to be investigated in greater depth

Experimental studies completed or ongoing

- **Out-of-pile analytical tests:** phase mixture behavior, erosion, FCI, etc.
- **Integral out-of-pile and in-pile tests:** steel jet fragmentation in MELT, relocation process validated in FAIDUS and CRGT
- **SAIGA test:** planned to study material relocation in a DCS-M-TT system



CONCLUSION
Ending words

Concluding remarks

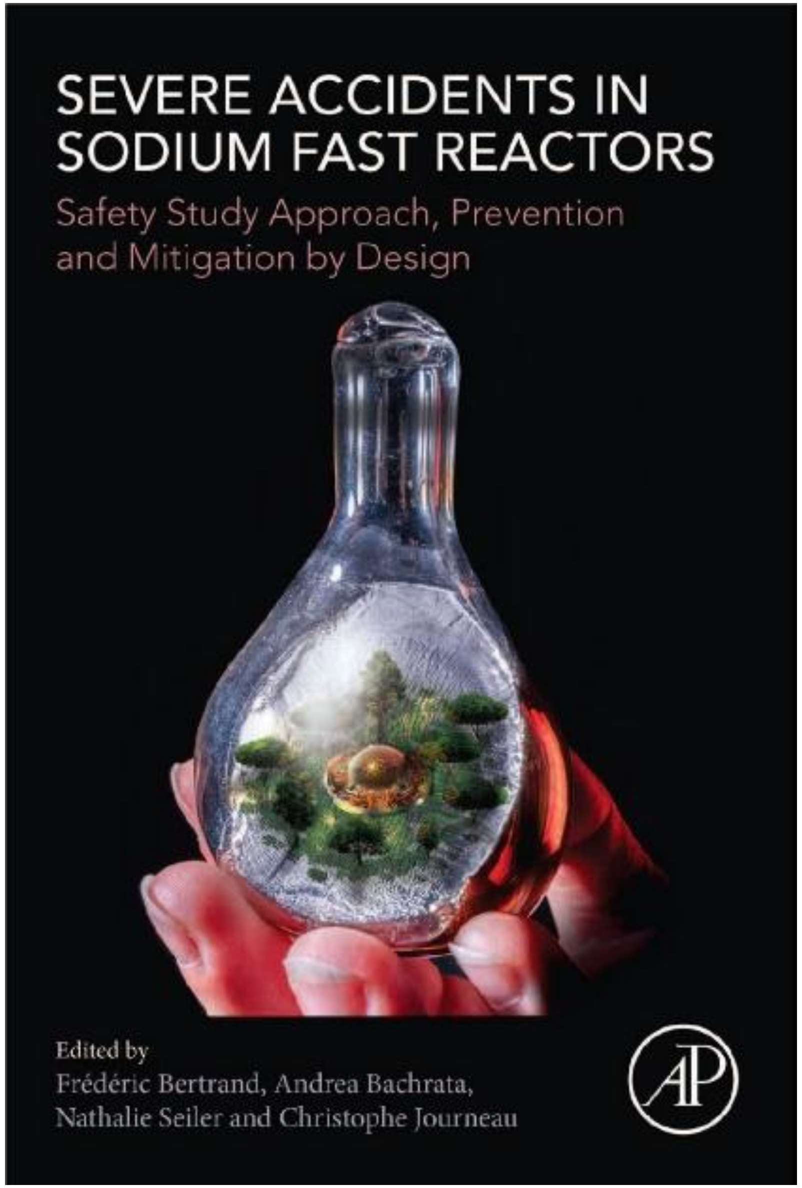
This book summarizes the work carried out as a team in a stimulating context (both national and international).

Following the postponement of ASTRID: progress has been made in mitigation and physics, but further R&D is still required.

R&D will continue within a **FRN/JPN** framework through the development of the Japanese prototype.
The authors wished to share their experience — both successes and difficulties.

If this helps future generations, even slightly, in their new projects, their objective will have been achieved.

THANK YOU FOR YOUR ATTENTION



Upcoming Webinars

Date	Title	Presenter
10 December 2025	Overview of Nuclear Energy Advanced Modeling and Simulation (NEAMS) tools to Accelerate the Development of Advanced Reactor Systems	Dr. Mauricio Tano-Retamales, Idaho National Laboratory, USA
28 January 2026	History of Nuclear Reactors Development	Prof. Christophe Demaziere, Chalmers University, Sweden
18 February 2026	Approach to Waste Characterisation as well as the Use of Synroc Technology for Advanced Fuel Cycle	Dr. Dan Gregg and Dr. Anton Peristyy, ANSTO, Australia

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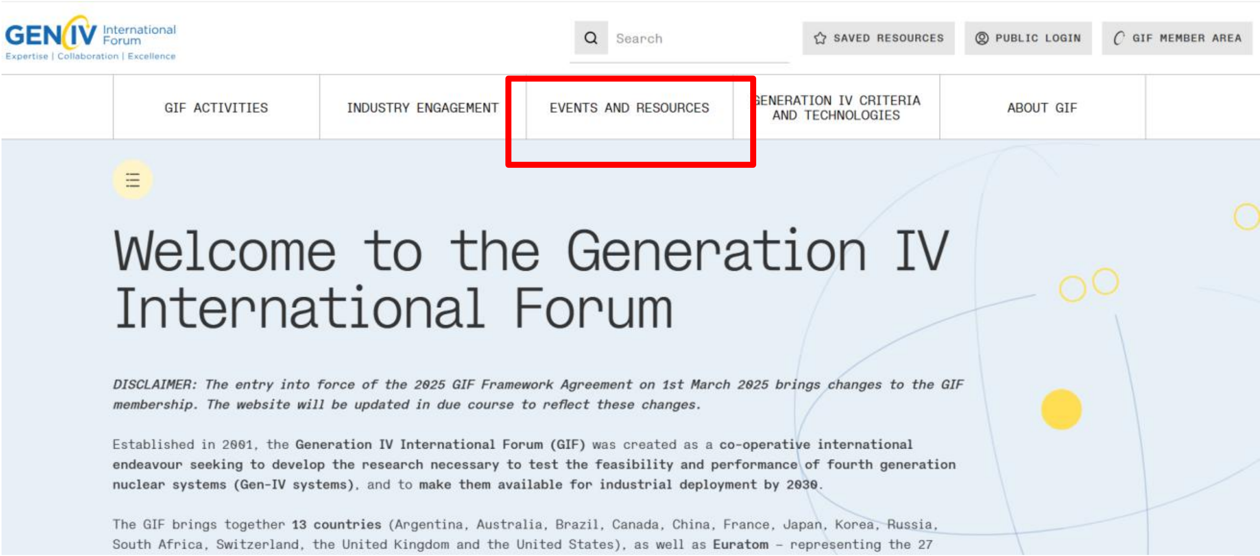

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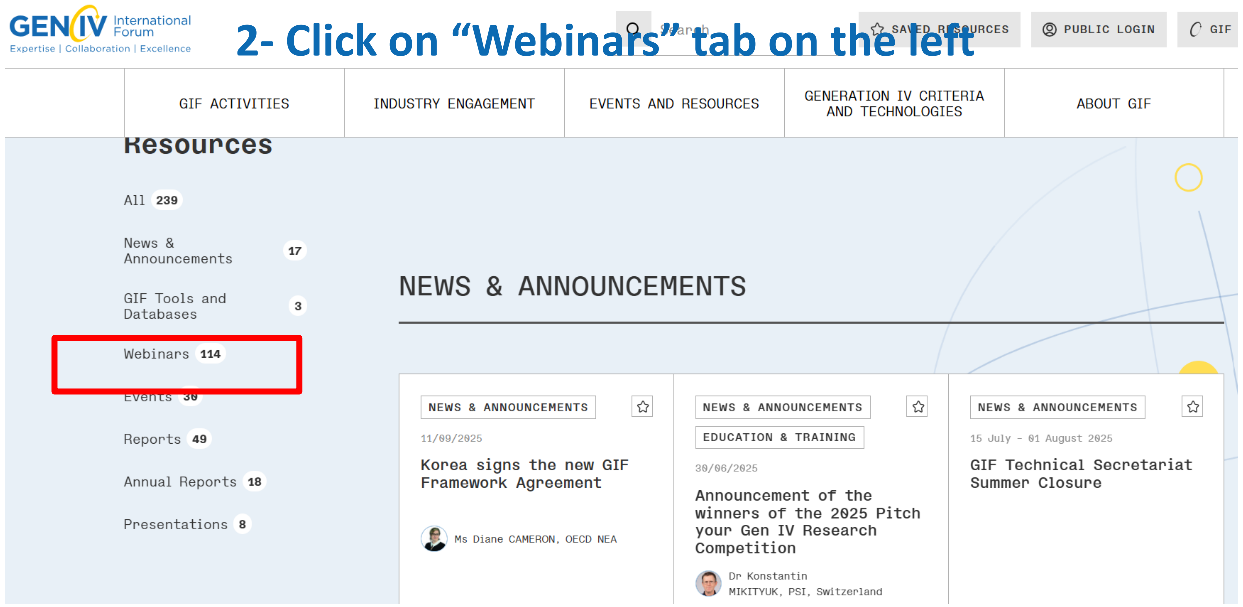

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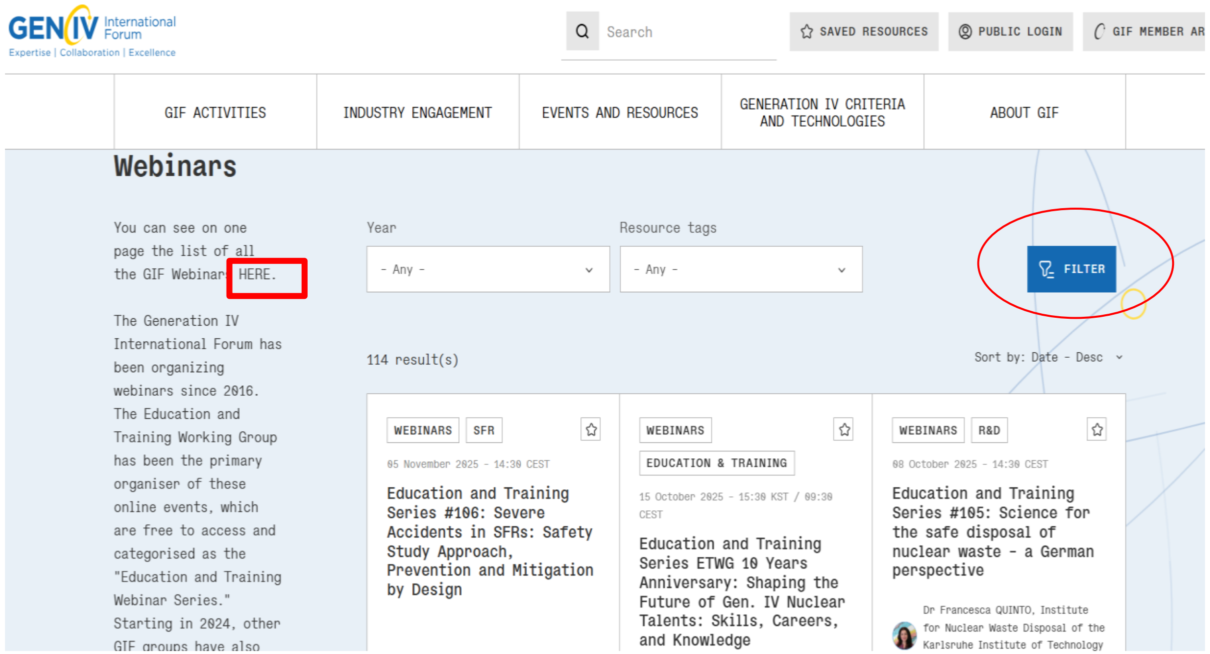

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QUESTIONS ?