# FRANCE-JAPAN COLLABORATION ON THE SFR

# SEVERE ACCIDENT STUDIES

Outcomes and Future Work Program

S. KUBO1, F. PAYOT2, H. YAMANO1, F. BERTRAND2, A. BACHRATA2, L. SAAS2, C. JOURNEAU2,

S. GOSSE2, A. QUAINI2, A. SHIBATA3, T. IITSUKA4, O. CZARNY5

1 Japan Atomic Energy Agency, Tokyo, Japan

2 Commissariat à l’Energie Atomique et aux Energies Alternatives, France

3 Mitsubishi FBR Systems, Inc, Tokyo, Japan

4 Mitsubishi Heavy Industries, Ltd, Kobe, Japan

5 Framatome, Lyon, France

Email contact of corresponding author: kubo.shigenobu@jaea.go.jp

**Abstract**

The paper presents major outcomes of the France-Japan ASTRID collaboration, as well as work programs from 2020 to 2024 in the field of severe accident study. In the ASTRID collaboration, severe accident sequences were summarized based on the various safety analyses for the important accident phases, which contributed to strengthen the confidence in the ASTRID severe accident progression for a robust safety demonstration and identification of R&D programs of common interest. Collaborative analyses have been conducted to evaluate ASTRID mitigation device efficiency for mitigation of power excursions, material relocation, and debris bed and molten pool behavior on the core catcher. The methodology for mechanical consequence assessment was also developed. In order to support the reactor studies, experiment studies have been planned and conducted regarding the reaction of core material mixtures, in-pile experiments for the fuel pin failure and material relocation through a steel duct structure, and out-of-pile experiments for the fuel coolant interaction (FCI) in the sodium pool. Severe accident analysis tool SIMMER-V with new simulation capabilities and SEASON platform have also been developed.

Based on these successful achievements, several tasks to study the large fields of severe accident domains are continuously common interest among the involved parties, which include development of severe accident analysis methodologies and synthesis of SA sequences and consequences, thermodynamics, kinetic and thermo-physical studies of core material mixture, development and validation of SIMMER-V, experiment programs on the molten core material relocation and FCI. After defining the technical contents and implementation plans, the five-year study programs have started.

## INTRODUCTION

Japan and France have continued collaboration on the research and development (R&D) of sodium-cooled fast reactors (SFRs) for a long time, and severe accident studies play an important role in it. The collaboration focused on in-pile fuel pin failure experiments in CABRI, a research reactor at the CEA Cadarache Center, from the 1980s to the 2000s [1, 2]. The elucidation of phenomena of fuel failure in SFRs advanced through tests in CABRI, and validation and development of analysis tools such as SAS4A [3] progressed. Through the collaboration, Japan and France have accumulated much knowledge on the evaluation of core damage behaviour of SFRs. In the 21st century, with progression of R&D for the commercialization of SFRs in Japan, a specific plan was presented for a large-scale SFR, called JSFR, of mitigation measures of core damage based on the results of the Experimental Acquisition of Generalized Logic to Eliminate re-criticalities (EAGLE) using the Impulse Graphite Reactor (IGR) in Kazakhstan [4, 5]. The ASTRID collaboration carried out from 2014 to 2019 was a full-scale France-Japan bilateral one for the ASTRID, a Generation IV demonstration reactor in France [6]. In the severe accident field, the collaboration was conducted on the core catcher design, the analysis and evaluation of event sequences and consequences of severe accidents. The experimental studies and analysis code development were also jointly studied. Japan and France have continued their SFR R&D after 2020, and severe accident studies have been continued, and occupy an important position in the SFR R&D collaboration from 2020 to 2024.

## APPROACH TO THE COLLABORATION OF SFR SEVERE ACCIDENT STUDIES

Based on the safety design philosophy of defence-in-depth, it is common in Japan and France that the basic safety design of SFRs has to take measures for prevention and mitigation of core damage in design extension conditions, in addition to the measures for the prevention of abnormal operation and failures, control of abnormal operation, detection of failure, and control of accidents within design basis. The mitigation of core damage is commonly done by introducing design measures to achieve in-vessel retention (IVR) and demonstrating its efficiency based on experiment data and analysis results, so that the collaborative R&D can proceed efficiently.

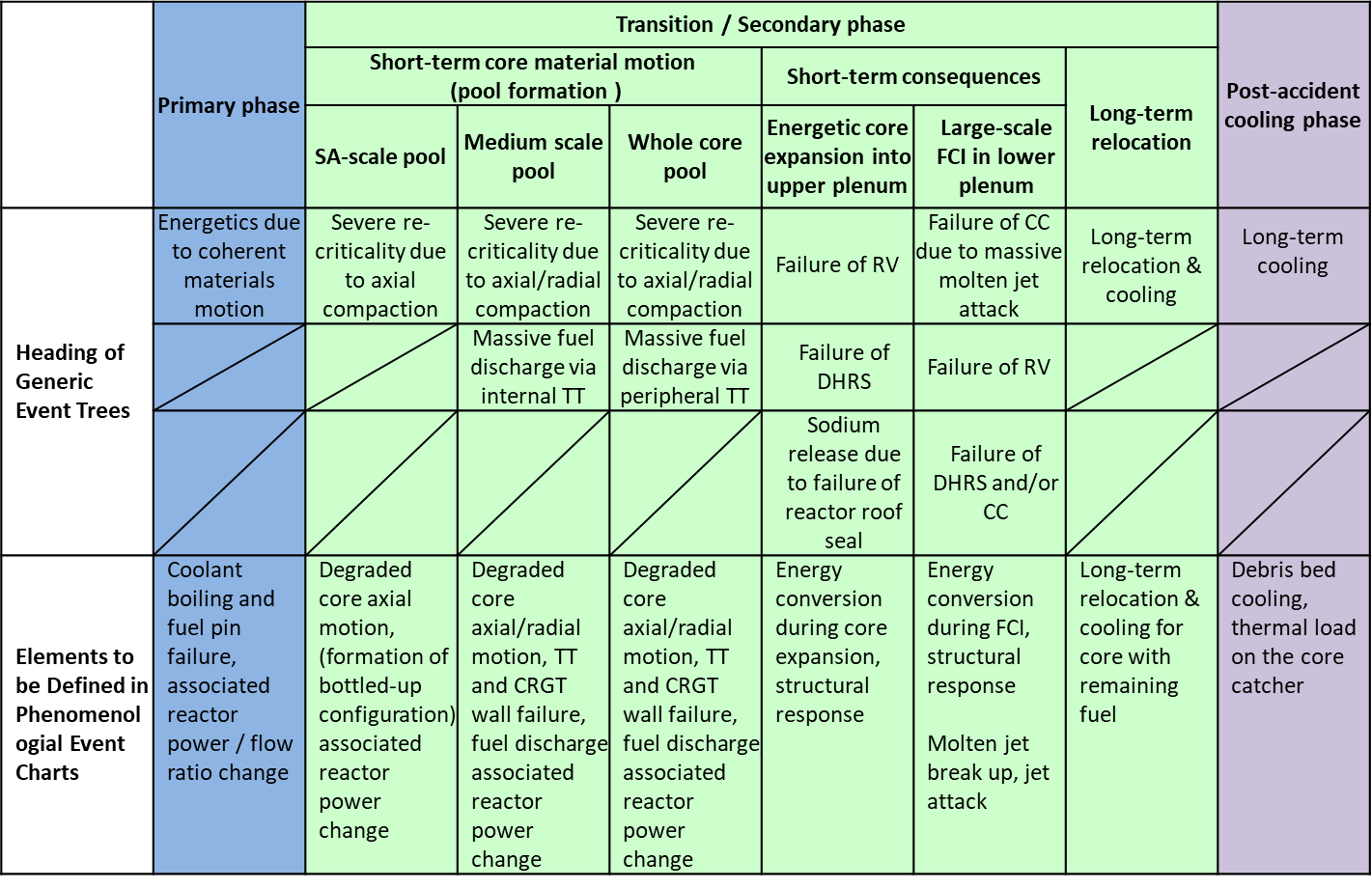
The JSFR, which has been developed in Japan, adopts a homogeneous core, and its sodium void reactivity is limited to avoid prompt criticality caused by coolant boiling. It has a dedicated steel duct inside the fuel assembly for early molten fuel discharge, control rod guide tube for facilitating additional discharge of degraded fuel from the core, and a core catcher installed inside the reactor vessel [4]. French ASTRID is equipped with a heterogeneous core for significantly reducing the sodium void reactivity to a value close to zero or even negative, the transfer tube for fuel discharge, and the in-vessel core catcher [6]. Although there are design variations, it is possible to apply common analysis methodologies to many parts of the system. Japan and France are implementing comprehensive R&D collaboration by combining the development of analysis methodologies, reactor analysis, experiment data acquisition, and analysis code development, focused on phenomena and design provisions, in which both countries are interested.

## REACTOR STUDY

### 3.1. Synthesis of severe accident sequences and consequences

In the safety design of the ASTRID, the mitigation measures were taken for Unprotected Loss of Flow (ULOF), Unprotected Transient Over Power (UTOP), and Unprotected Sab-Assembly Fault (USAF). These are severe accident initiating sequences, and the measures have been evaluated to show their effectiveness. Analyses and evaluation have been performed for accident progression and consequences for each accident phase of ULOF, which can cover the whole severe accident phases, i.e., primary phase, transition / secondary phase, and post-accident cooling phase. The effects of mitigation measures against core damage, such as sodium void reactivity reduction, the transfer tube, and the in-vessel core catcher, have been evaluated in this study. In addition, the confinement capability of severe accident effects has been evaluated on the assumption of energetic core expansion.

*TABLE 1. Summary of ASTRID ULOF severe accident events*



TT: Transfer Tube, CRGT: Control Rod Guide Tube, RV: Reactor Vessel, CC: Core Catcher, DHRS: Decay Heat Removal System, FCI: Fuel Coolant interaction

The results of these analyses and evaluations were integrated, and the spectrum of severe accident event progression has been organized into Generic Event Trees (GETs). Furthermore, the Phenomena Identification Ranking Tables (PIRTs) have been created to analyse dominant phenomena and important physical parameters in each event tree branching point (headings). The Phenomenological Event Charts (PECs) have been created to identify the most probable sequence. The headings of GETs and elements to be defined in PECs are shown in Table 1 [6].

After summarizing both Japan and France’s ideas in the ULOF PIRT, importance and uncertainty were ranked for each extracted elementary phenomenon. These results were reflected in the code development and test research plans shown in the next sections to clarify the issues to be addressed in each R&D. Each achievement of R&D will be fed back to the overall evaluation of severe accident studies using GETs, PECs, and PIRTs. Figure 1 shows the relation between the severe accident study tasks to be conducted in the France-Japan collaboration on the SFR development program from 2020 to 2024.



*FIG. 1. France-Japan SFR severe accident study task structure*

### 3.2. Development of analysis methodology and its application

#### 3.2.1. Event sequence analysis

SAS-4A, SAS-SFR, SIMMER [7], and other codes have been developed as severe accident analysis codes for SFRs and have been applied to reactor analysis and validation based on the CABRI, SCARABEE [8], EAGLE, and other test results.

In this France-Japan collaboration, application methodologies for the ASTRID analysis has been developed using SIMMER as the main analysis tool.

The following are characteristics of the event sequences, focused on core damage analysis of the ASTRID.

* In the CFV (Cœur à Faible coefficient de Vide sodium) core of ASTRID, the reactivity feedback due to coolant temperature increase and coolant boiling is negative in general, and the ULOF primary phase is mild event sequences with a power decrease.
* After the transition phase, the core fuel melting progresses by re-criticality due to fuel and steel relocation accompanying core damage progression. Simultaneously with this core melt progress, the transfer tube walls installed in and around the core are broken by heat load from the molten fuel and fuel discharge paths are open.
* The core goes under a neutronical shutdown state due to fuel discharge from the transfer tubes.

These series of event transitions were analysed using SIMMER in the following method.

* An analysis geometry, including the reactor vessel, was modelled in three dimensions, and access behaviour to individual transfer tube was analysed (CEA) [9].
* A parametric analysis was conducted using a local analysis model focusing on a single transfer tube, in which design conditions, physical parameters, and SIMMER modelling methods were varied as parameters. High sensitivity items on the fuel discharge were identified (CEA, JAEA, MFBR) [9].

In the collaboration from 2020, with a view to the commonization of the analysis methods of Japan and France, the analysis conditions will be standardised, and benchmark analysis will be performed to analyse the cause of the difference in analysis results between Japan and France. Sensitivity analyses will be carried out in order to systematically analyse the effect of the important phenomena and parameters extracted by the PIRTs.

The CEA is developing DETONa and other codes as a tool to enable statistic analysis [10].

#### 3.2.2 Evaluation of mechanical energy containment

Assuming a hypothetical core state such as massive compaction of fuel, the generation of a huge amount of nuclear thermal energy due to re-criticality melts the fuel over a wide area in the core, and fuel temperature may exceed its boiling point in a region where it becomes the maximum. An evaluation method was developed for evaluating the resistance of the reactor structure to the mechanical energy generated by the expansion of the core materials at high temperature and high pressure in such a state. For this purpose, SIMMER and tools for analysing structural responses (EUROPLEXUS for CEA and AUTODYN for JAEA) are used [11]. SIMMER is used to analyse core expansion behaviour, including heat and momentum exchange with sodium existing around the core. Based on the results, an analysis method was developed, in which the relation between the pressure and expansion volume evaluated by SIMMER is given as the input of the structural response analysis code, regarding the expanding core as a pressure source for the expansion work. This analysis method developed by JAEA for the prototype fast breeder reactor MONJU was applied to ASTRID, and in addition to the conventional analysis for the upper core plenum, its application to the lower plenum was examined in Japan and France.

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#### 3.2.3 Coolability evaluation

The design strategy of the ASTRID against a core damage accident is to achieve IVR. For IVR, degraded fuel discharged through transfer tubes from the core region is accumulated, retained, and cooled on the core catcher. JAEA and MFBR have developed a one-dimensional plant dynamics code, Super-COPD, which not only simulates the whole plant dynamics but also evaluates the coolability of the debris bed consisting of degraded core materials on the core catcher. Using Super-COPD, JAEA and MFBR performed calculations and clarified coolable ranges of key parameters such as a debris bed height, debris bed power, coolant temperature around the debris bed [12]. In the calculations, debris bed cooling, sodium flowing around the core catcher, and simplified heat transfer to a heat sink were modelled and applied to the ASTRID configuration. Because of these, the effects of operation conditions of decay heat removal systems on the coolability of the debris bed can be evaluated.

In the collaboration from 2020, these models will be coupled with a multi-dimensional thermal-hydraulic code, and the coolability of the debris bed will be evaluated in consideration of thermal-hydraulics in the reactor vessel.

Furthermore, the following items are planned to be studied:

* Coolability of core remaining fuel
* Fragmentation and quench of fuel discharged into the reactor lower plenum
* Erosion of structural materials due to jet impingement of molten materials

## CODE DEVELOPMENT AND EXPERIMENTAL STUDIES

### SIMMER-V development

JAEA and CEA collaboratively develop SIMMER-V based on SIMMER-III and SIMMER-IV owned by JAEA [7]. The development targets are a detailed fuel pin model called DPIN3 and an enhancement of calculation performance. One of the objectives of code development is to achieve the capability of consistent evaluation from initiation of core disruptive accident to fuel relocation outside the core region, as based on the above development. Another development objective is a linkage analysis system combined with several SIMMER-V calculations, according to the progress of core disruption.

The development of its prototype has been already completed. Assessments, verifications and validations of DPIN3 and physical models in SIMMER-V are being conducted by simple calculations and experiment analyses.

### Thermo-dynamic study

Unlike an oxidative environment in light water reactors (LWRs), chemical interactions were not very significant in past studies of SFR severe accidents under a reducing environment, which have mainly focused on fast transient phenomena relating to energetic re-criticality potential. Since safety assessment methodologies for long-term core material relocation behaviour are also important, JAEA and CEA have been studying, as a common technical issue, the chemical interaction among core material mixtures during core damage processes [13]. One of the specific issues is the chemical interaction between stainless steel (SS) used in reactor structures, and boron carbide (B4C) in neutron absorber rods or shielding structures, and further interaction with oxide fuel under severe accident conditions. Based on severe accident studies of LWRs, this collaborative study is intended to acquire basic experiment data on chemical interaction between B4C, SS and UO2 fuel—the effect of Pu in (U,Pu)O2 fuel is considered in thermodynamic calculations, to obtain high temperature thermodynamic data and thermo-physical properties, and to model both thermodynamic equilibrium and liquefaction kinetics of the mixtures.

In the framework of the previous Implementing Arrangement covering the period 2014 to 2019, the first task was the comparison of Japanese and French thermodynamic databases on solidus and liquidus temperatures of B4C-SS composition calculated using the Calphad method. The second task was thermodynamic studies and experimental programs for modelling core material mixtures. JAEA has developed a B4C-SS eutectic reaction model for severe accident simulation code SIMMER [14], for which thermophysical property data were obtained in a wide temperature range from solid to liquid states [15, 16]. These collaborative tasks were successfully accomplished, and continuous R&D items were also identified.

In the collaboration from 2020, JAEA continues further experiments and analytical studies in the B4C-SS system. CEA conducts experiments and analytical studies in the B4C-SS-UO2 system and develops Calphad modelling of the mixtures to be coupled with severe accident codes. The collaborative tasks are the extension of these experiment data and the development of the analytical model for enhancing the simulation capability.

### Experiment programs using IGR and related facilities

An experimental study using a test reactor is one of the effective and valuable measures to study phenomena that occur in severe-accident sequences, to validate evaluation methodologies such as computer simulation, and to demonstrate the effectiveness of designs for mitigating severe-accident consequences. IGR in the National Nuclear Centre of the Republic of Kazakhstan (NNC-RK) is a unique test reactor that can melt test fuel-pins with the bundle scale, and JAEA has conducted an experimental study program, EAGLE, as a joint-research program with NNC-RK. Through the program, knowledge of phenomena that affect the sequences of core degradation, such as molten-fuel discharge through the inner duct structure, have been obtained [17]. At present, the EAGLE program has entered phase 3 (EAGLE-3), and molten-core discharge through the control rod guide tube, accumulation and coolability of molten-core relocated into the inlet sodium plenum, and coolability of core-remnant fuel (in-place cooling) are studied [18].

CEA has initiated an experimental test program called SAIGA (Severe Accident In-pile experiments for Gen-IV reactors and the Astrid prototype), which aims to carry out one experimental test in IGR. The SAIGA experiment intends to simulate an accident sequence caused by the mismatch between power to coolant flow under ULOF conditions, that is, coolant boiling, fuel-pin disruption and propagation, a transfer tube wall failure resulting in fuel discharge [19].

In the France-Japan collaborative study, information about experimental programs EAGLE and SAIGA are exchanged to proceed with each program effectively. In addition, the collaborative validation study of the SIMMER-V code has been performed on the basis of experiment data obtained through the phases 1 and 2 of the EAGLE program (EAGLE-1 and EAGLE-2).

### Fuel-Coolant Interaction experiments

Molten-core materials discharged from the original core region flow into the sodium pool around there, and this leads to thermal interaction with sodium, called Fuel-Coolant Interaction (FCI). Sodium vaporisation and rapid expansion of sodium vapour caused by FCI adversely affect the integrity of in-vessel structures. FCIs also fragment molten-core materials and fragment disperse in the reactor vessel. Therefore, FCI is one of the important phenomena during the downward discharge of molten-core materials because it affects not only the integrity of the lower structures of the reactor vessel, including the core catcher, but also the state of the debris bed accumulated on there.

One of the representative studies of FCI using uranium-oxide and sodium is the FARO-TERMOS experiment, and knowledge on pressurisation, the length of molten fuel flow, and debris bed of fragmented fuel were obtained [20]. JAEA has performed a series of studies simulating FCI by using molten metal and sodium, in which a sequence of thermal interaction and fragmentation process is observed directly by using an X-ray imaging system [21]. In the France-Japan collaborative study, a new software program, SPECTRA (Software for Phase Extraction and Corium Tracking Analysis), which is being developed by the CEA, is applied to process the above X-ray images, so that discrete melt fragments and sodium vapour expansion can be tracked [22, 23]. FCI processes and their consequences are investigated in detail by integrating results of X-ray image analyses, temperature transient data, state of solidified melt, and other data.

## CONCLUSION

Severe accidents of oxide-fuelled SFRs have been long studied, and many findings have been accumulated though experiments and analyses. Based on these findings, the collaboration of severe accident studies between Japan and France is conducting R&D focusing on evaluating the effectiveness of mitigation measures against core damage during a severe accident that are applicable to Generation IV SFRs. In the collaborative SFR development program from 2020 to 2024 following the ASTRID collaboration from 2014 to 2016, programs for reactor analysis methodologies, numerical simulation tools, and research on key phenomena are under way. The study results will be summarized in the form of severe accident event sequences and consequences expressed by the event trees and event charts, in which dominant phenomena and their sensitive parameters will be incorporated with experiment data and calculation results to be obtained in severe accident studies. Analysis of PIRTs is being conducted to deepen mutual understanding on severe accident phenomena and to identify further study points. The design conditions and efficiency of the mitigation design measures for the future SFR will be refined.

ACKNOWLEDGEMENTS

The present study includes the result of “Technical development program on a fast reactor international cooperation, etc.” and “Technical development program on a common base for fast reactors” entrusted to the Japan Atomic Energy Agency (JAEA) by the Ministry of Economy, Trade and Industry (METI).

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