Recent Activities of the Safety and Operation Project of the Sodium-Cooled Fast Reactor in the Generation IV International Forum

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Abstract: The Generation IV (GEN-IV) international forum is a framework for international cooperation in research and development (R&D) for the next generation of nuclear energy systems. Concerning the sodium-cooled fast reactor (SFR) system, there are five cooperation projects for R&D. The SFR Safety and Operation (SO) project addresses the area of the safety technology and the reactor operation technology developments. The aim of the SO project includes (1) analyses and experiments that support establishing safety approaches and validating performance of specific safety features, (2) development and verification of computational tools and validation of models employed in safety assessment and facility licensing, and (3) acquisition of reactor operation technology, as determined largely from experience and testing in operating SFR plants. The tasks in the SO topics are categorized into the following three work packages (WP): WP-SO-1 “Methods, Models and codes” is devoted to the development of tools for the evaluation of safety, WP-SO-2 “Experimental Programs and Operational Experiences” includes the operation, maintenance and testing experiences in experimental facilities and SFRs (e.g., Monju, Phenix, BN-600 and CEFR), and WP-SO-3 “Studies of Innovative Design and Safety Systems” relates to safety technologies for GEN-IV reactors such as active and passive safety systems and other specific design features.

In this paper, recent activities in the SO project are described.

Key Words: SFR, Safety, Gen IV.

1. INTRODUCTION

The R&D activities on SFRs in the framework of Generation IV International Forum are organized in five projects: System integration and assessment, Safety and operation, Advanced fuel, Component design and balance-of-plant, and Global actinide cycle international demonstration.

This paper summarizes recent advances from China, France, Japan, Republic of Korea, Russian Federation and Euratom for the three areas of the Safety and Operation Project:

1. Methods, Models and codes,
2. Experimental Programs and Operational Experiences
3. Studies of Innovative Design and Safety Systems.

2. METHODS, MODELS AND CODES,

CIAE (China) is developing the FASYS system analysis code to analyze the system response in a wide range of SFR transients.

The development plan includes two phases:

Phase I (2012~2016) devoted to main models development, main function definition and implementation, preliminary validation and preliminary applications.

Phase II (2017~2019) focuses on all models development, friendly interface development and detailed validation and verification.

The major code models include sodium loop thermal hydraulic model (pipes, pumps, IHX, pools), core analysis model (point kinetics and reactivity feedback, pin heat transfer and single-phase coolant thermal hydraulic model) and plant protection and control system models.

The models mentioned above are divided into three types: hydraulic model, thermal model and neutron kinetics model.

The major modules in the code include preprocessor module, system geometry building module, steady-state calculation module, transient calculation module, postprocessor module and output module.

The FASYS code is architected in the FORTRAN 95 language with good coding style, and is developed by mainly adopting the structured programming method.

The general structure of the code is presented in Figure 1.

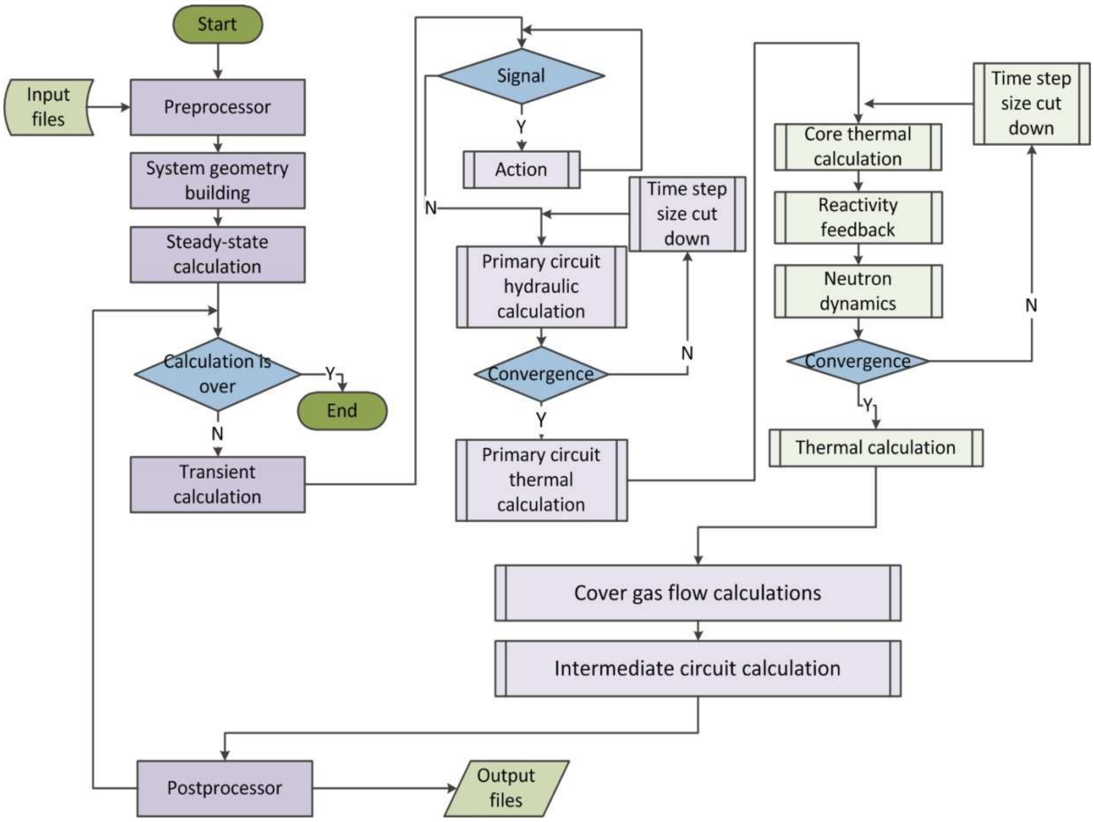


Fig. 1. Structure of the FASYS code

The validation of the code was initiated by comparing to the CEFR commissioning tests. In the case of the hydraulic model validation, Figure 2 shows the results on primary pumps flow during coast down.

In this experiment, both of the CEFR primary circuit pumps are coasting, and the pump flow meter value is recorded. As seen in the figure, the agreement is quite close, with FASYS predicting the pump flow with time, as compared to the experimental data.

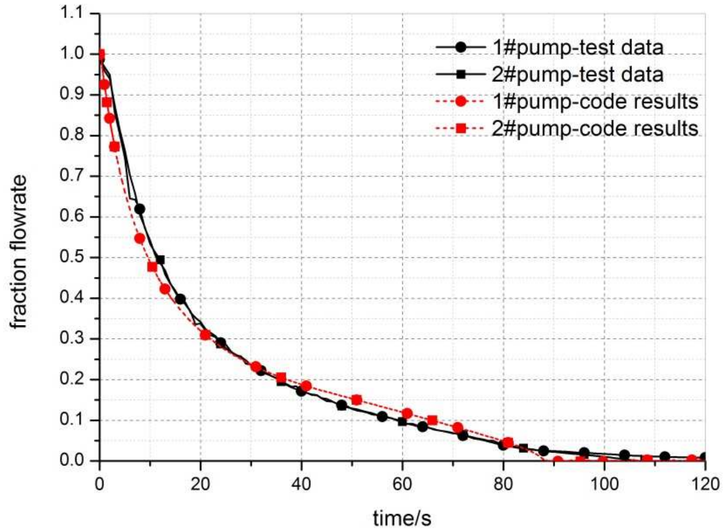


Fig. 2 – CEFR Primary pumps flows

Another exemple of the validation is presented in the figure 3 with the comparison between the LOSP (Loss Of Station Power) test data and calculation results in IHX first side inlet temperature and outlet temperature. It can be indicated that the calculation results were matched well with the test data.

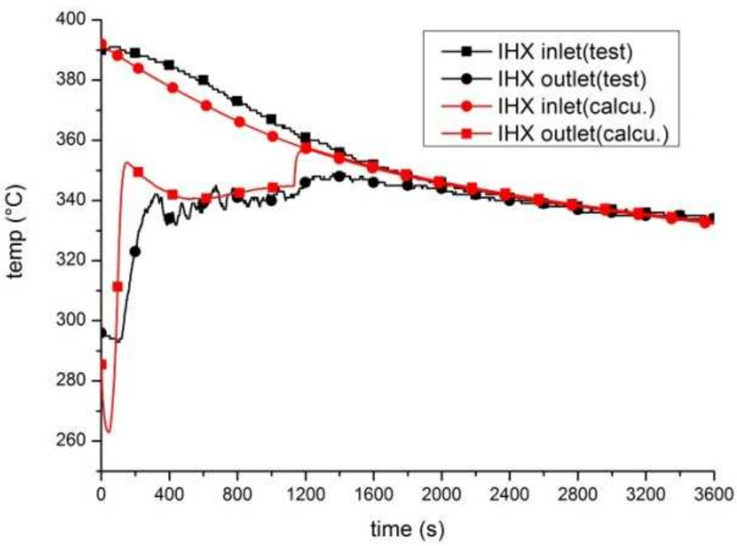


Fig. 3 – IHX temperatures during a LOSP

CIAE (China) is developing the HCDA hypothetical core disruptive accident analysis code. When the core melts, the reactivity increase, and delayed neutron fraction is small and prompt neutron lifetime is short, that may lead to prompt supercritical. Therefore scale change of core may lead to prompt supercritical, then core disruptive accident happen, and core release the energy which shock stick the plant system and main vessel. Core disruptive accident analysis is important for the fast reactor sever accident analysis.

The code adopts the improved Bethe-Tait model, and improves the equation of state and reactivity feedback, the assumption of the code model includes: reactor point kinetics equation, reactivity change adopt first order perturbation theory, six group delayed neutron,

The main code models include: neutron dynamics model, reactivity feedback model, core thermal model, core disruptive model. The general structure of the code is presented in Figure 4.

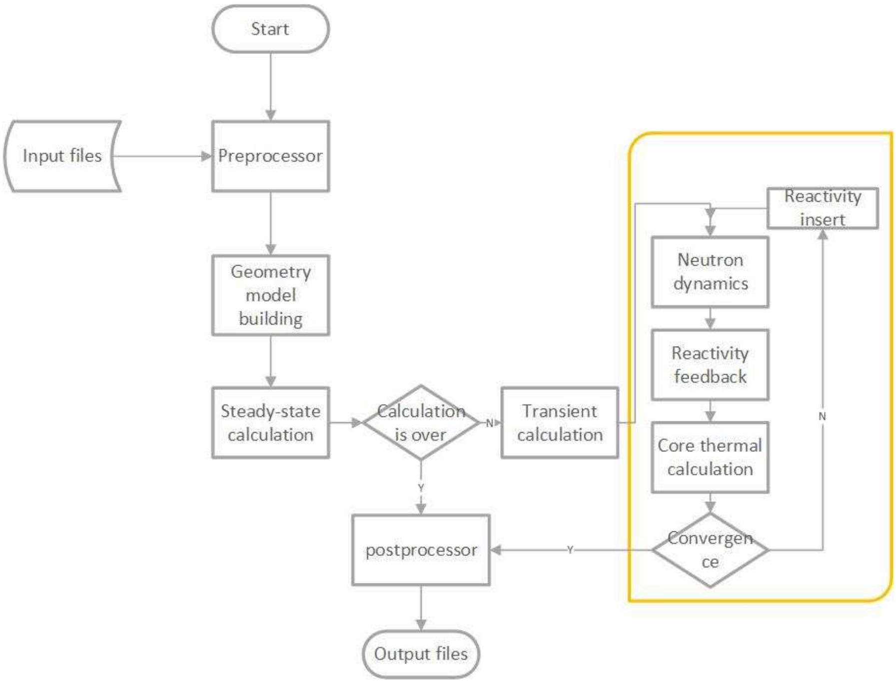


Figure 4 –Structure of the HCDA code

CEA is developing a physical-probabilistic tool dedicated to molten material core discharge during postulated severe accidents [1]. The study presented here deals with the assessment, against SIMMER results. This 0D tool handles heat transfers from molten, possibly boiling, pools to mitigation tube walls, fuel crust evolution, segregation/mixing of fuel/steel pools, radial thermal erosion of mitigation tube wall, and discharge of molten material with axial thermal erosion of the transverse tube, coupled with neutronic evolution of the fuel power. This very low time consuming tool enable large sensitivity studies on different physical and design parameters.

The physical models and the calculation scheme of the physico-statistical (also called analytical) tool are generic to the treatment of various material molten pool of constant radius. This tool is parametrized to facilitate sensitivity evaluations (such as initial reactivity, wrapper thickness of the mitigation tubes, initial material masses, fuel power…).

This tool couples the temporal evolutions of materials located inside the upper and lower fissile zones to the evolution of the global core neutronics. The considered molten pools are composed of steel and fuel which could be mixed or segregated (steel layer above a fuel lower pool). The spatial distribution of materials between these two pools evolves during the transient depending on material temperature.

Various configurations are treated: totally segregated materials, partially segregated configuration where the steel mass is distributed between a steel layer which is above and a lower mixed steel/fuel pool or totally mixed configuration where the pure steel layer has disappeared.

The lower mixed pool is considered homogeneous with physical properties dependent on the proportions of the various materials inside the pool.

For comparison with the SIMMER code purposes a degraded core state of a ASTRID like core is considered. The nominal geometry of such a core is presented in Figure 5.

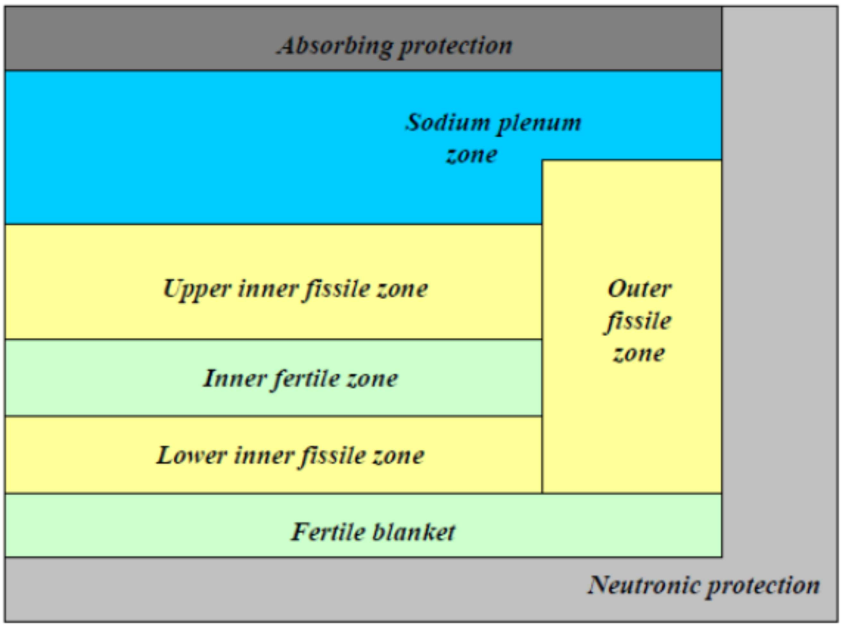


Figure 5 – ASTRID like heterogeneous core nominal geometry

For the reference case, the core is assumed at residual power and at initial time. The reactivity is null. It is assumed also no reactivity supply during the transient (caused for example by a sodium return inside the plenum).

The main compared results on a severe accident scenario are summarized on the following table.

	Analytical tool [s]	SIMMER [s]
Re criticality (ρ>0 pcm)	0.52	0.6
Prompt criticality (ρ>364.5 pcm)	0.89	1.17
Wrapper failure at upper fissile zone	1.22 (on the lower pool)	1.15
Start of materials mixing - Upper fissile zone	0.92	-
Fuel ejection from upper fissile zone	0.97	1.17
Wrapper failure at lower fissile zone	1.6 (on the upper pool)	1.13

The transient evolutions calculated with the simplified analytical tool and SIMMER are similar and the same reactivity contributions are observed. The material ejection in the analytical tool takes few tenth of seconds where as it is instantaneous in SIMMER. This behavior has been explained and seems realistic according to some past experimental results [2]. Finally, it has been demonstrated that this analytical code will be a valuable tool to perform sensitivity studies and highlights the most influent parameters. It will be used in support to the design of mitigation devices and will enable to perform large statistical treatment of uncertainties.

JAEA has been developing probabilistic risk assessment (PRA) methodologies against various external hazards, such as snow, tornado, strong wind, rainfall, volcanic eruption and forest fire [3]. The PRA methodology consists of external hazard curve evaluation and event sequence analysis methods to estimate a core damage frequency. To develop the forest fire PRA methodology, JAEA has evaluated an external hazard curve of the forest fire based on a logic tree [4]. The logic tree consists domains of “forest fire breakout and spread conditions”, “weather condition”, and “vegetation and topographical conditions”. A location nearby a typical nuclear power plant site in Japan was selected for our studies. The frequency of a large forest fire of the location is approximately 1/5 of the average in Japan. A number of forest fire simulations were performed to obtain a response surface for a frontal fireline intensity at different combinations of wind speed and humidity. The hazard curve has been successfully evaluated by Monte Carlo simulations where one sample gave a unique intensity from the response surface and its frequency was given by the combination of the branching probabilities in the logic tree.

JAEA has performed the fundamental experiments of sodium-concrete reaction (SCR) by thermal analytical techniques for developing the reaction model [5]. As a series of possible reactions on concrete ablation in SCR, kinetic behavior of Na_2O–SiO_2 and Na_2O–concrete aggregate reactions were investigated. Kinetic parameters were obtained by using kinetic laws such as Kissinger and Freedman methods.

Based on the work performed within the Euratom Collaborative Project on European Sodium Fast Reactor (CP-ESFR) [7], investigations of the impact of minor actinide (MA) bearing fuel on the transient behavior of the optimized oxide ESFR core were performed considering both Beginning of Cycle (BOC) and End of Cycle (EOC) conditions. The assessments were done using the thermal-hydraulic system code SPECTRA [8] by comparing several transient analyses for two reactor core configuration: a reference case for a core without MA-bearing fuel and a core with 4% homogeneous MA loading.

A detailed SPECTRA model of the ESFR consisting of the reactor primary system, the three cooling loops and two decay heat removal loops was developed. The nodalization of the SPECTRA model of ESFR is shown in Figure 6.

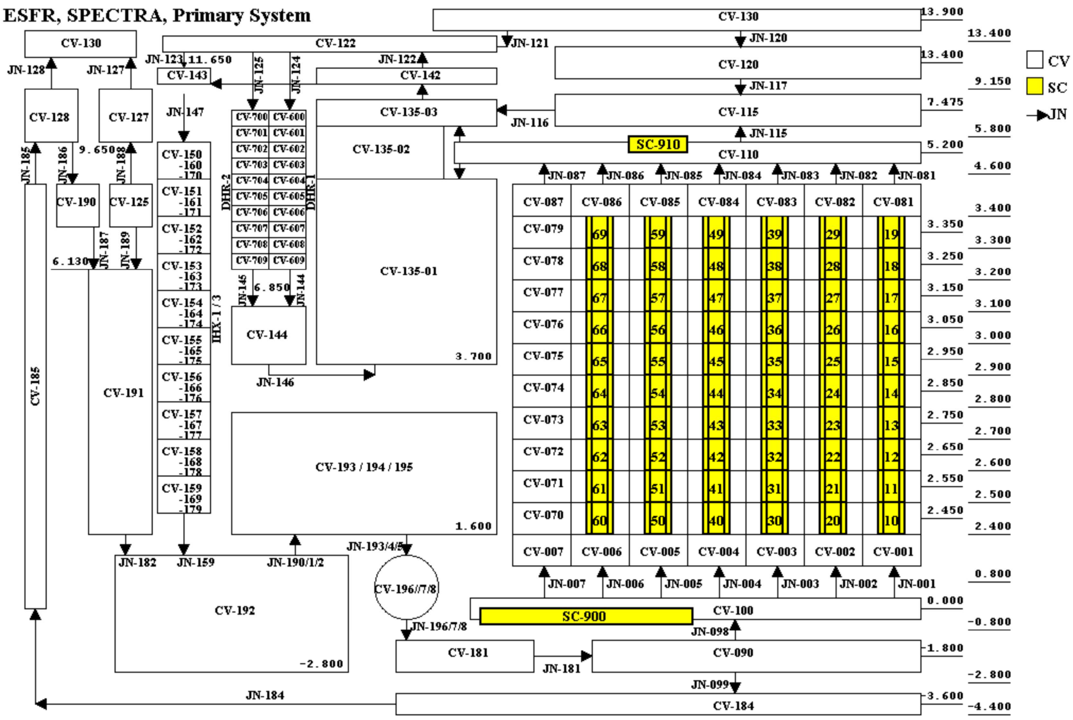


Figure 6 - Nodalization of the primary system - SPECTRA model of ESFR

The neutronics model is based on point kinetics considering relevant reactivity feedbacks including radial core expansion, control rod driveline expansion, fuel axial expansion, coolant temperature and fuel Doppler reactivity. The reactivity data for the model were obtained from full core three dimensional neutronics calculations.

The assessments of the impact of MA-loading on the ESFR core were performed covering a wide range of unprotected transients including: Speed reduction of all primary pumps, Coast-down of all secondary pumps, LIPOSO, Doubling of core bypass flow, Loss of FW on all SG-s, Runaway of grouped Control Rods, Coast-down of all primary pumps and Station blackout. In addition, protected transients were analyzed for those situations where unprotected transients resulted in exceeding the limit on maximum cladding temperature.

In general, the results of this study show that there is no drastic change of the transient behavior of the ESFR core due to MA loading for both BOC and EOC conditions. The qualitative behavior of the transients remains more or less unchanged. It is observed, however, that time-delays to reach sodium and clad temperature threshold are shortened when MA-fuel are loaded in the core. Furthermore, it is shown that MA-fuel loading leads to a significant reduction of the Doppler constant and an increase of the sodium void effect.

In the U.S., construction and operation of a nuclear power installation requires licensing by the U.S. Nuclear Regulatory Commission (NRC). A vital part of the licensing process is the analysis of the source terms that represents the potential release of radionuclides during normal operation and accident sequences. Historically, source term analyses have utilized deterministic, bounding assessments of radionuclide release to the environment. Significant advancements in technical capabilities and knowledge have enabled the development of more realistic analyses such that a mechanistic source term (MST) assessment is now expected to be a requirement for advanced reactor licensing.[9][10][11]

Argonne National Laboratory has assessed the state of development of an MST for sodium fast reactors (SFR) and qualitatively identified and characterized the major sources and transport processes.[12][13] Due to common design characteristics among current U.S. SFR vendor designs, a metal-fuel, pool-type SFR was selected as the reference design for this work, which allows gaps and uncertainties in the current knowledge base to be identified.

Radionuclides originate both in-vessel and ex-vessel, including in-core fuel, primary sodium and cover gas cleanup systems, and spent fuel movement and handling. Transport phenomena affecting various release groups include fuel pin and primary coolant retention and behavior in the cover gas and containment (see Figure 7). Radionuclides released from a primary sodium fire are also considered as potential sources. Available experimental data relevant to the aforementioned phenomena and operating incidents at domestically operated facilities have been reviewed.

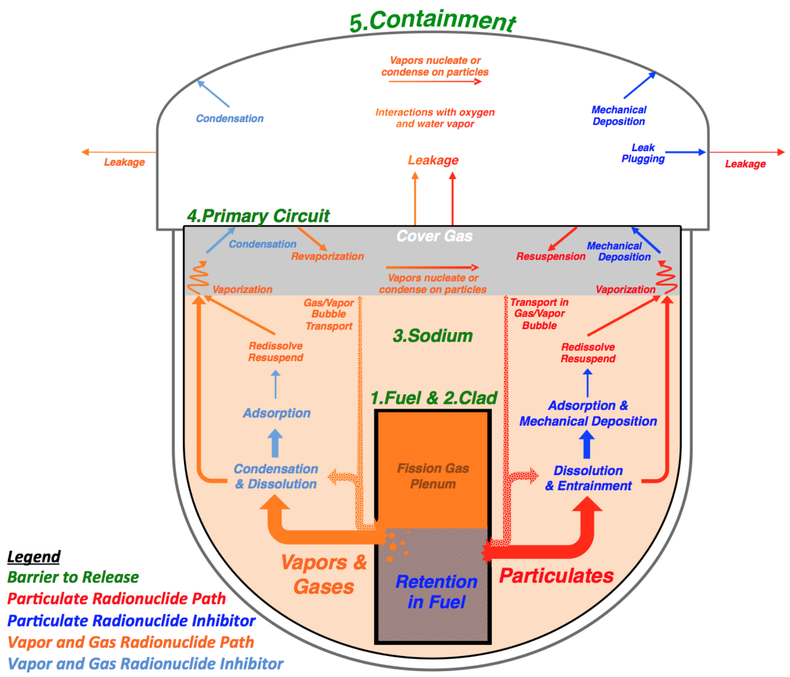


Figure 7: Radionuclide Transport Pathways in Pool-Type SFRs

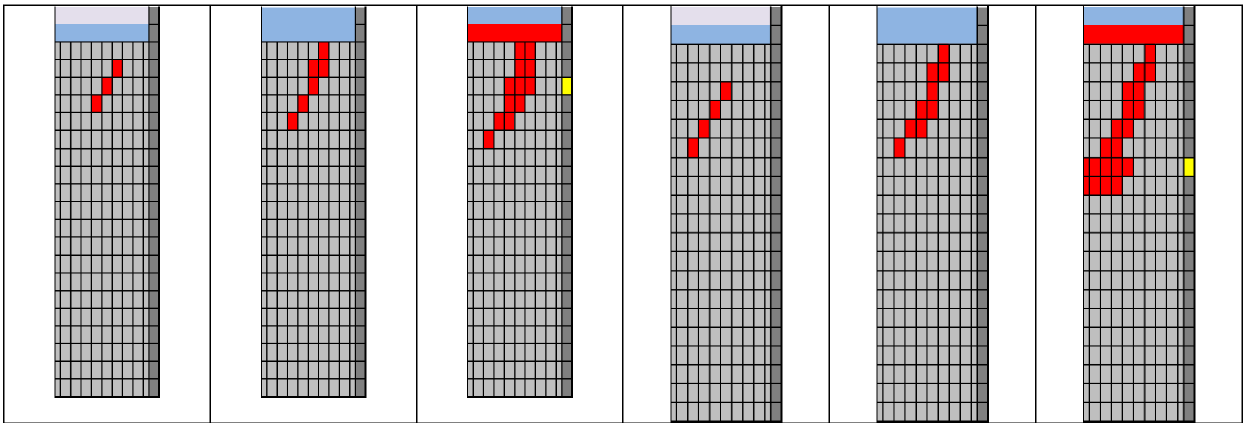
Following this initial assessment, Argonne developed estimates for the release fraction of radionuclides from metal fuel pins to the primary sodium coolant during fuel pin failures at a variety of temperature conditions. Release estimates were based on the findings of an extensive literature search that included past experiments and reactor fuel damage accidents. Data sources for each radionuclide of interest were reviewed to establish release fractions, along with possible release dependencies, and the corresponding uncertainty levels.

Considering the extensive range of phenomena affecting the release of radionuclides, the existing state of knowledge generally appears to be substantial, and may be sufficient in most areas. For core damage accidents, high retention rates can be expected within the fuel matrix and primary sodium coolant for all radionuclides other than the noble gases. These factors greatly reduce the magnitude of possible radionuclide release to the environment. Although the current knowledge base is substantial — and radionuclide release fractions were established for the elements deemed important for the determination of offsite consequences — the following gaps were identified.

- There is uncertainty regarding the transport behavior of iodine, barium, strontium, tellurium, and europium during metal fuel irradiation to high burnup levels. The migration of these radionuclides within the fuel matrix and bond sodium region can greatly affect their release during pin failure incidents. Post-irradiation examination of existing high burnup metal fuel can likely resolve this knowledge gap.

- Data is sparse regarding radionuclide release from molten high-burnup metal fuel in sodium, which makes the assessment of radionuclide release from fuel melting accidents at high fuel burnup levels difficult. This gap could be addressed with fuel melting experiments using samples from the existing high burnup metal fuel inventory.
- The available thermodynamic data regarding the behavior of lanthanides and actinides in liquid sodium is limited. However, a determination of the data requirements for MST development should be formally made prior to the expenditure of significant research efforts to expand that data.

An effort to expand the SAS4A models for the analysis of metal fuel cores has been performed in KAERI in the framework of collaboration with ANL. The SAS4A safety analysis code, originally developed for the analysis of postulated Severe Accidents in Oxide Fuel Sodium Fast Reactors (SFR), has been significantly extended to allow the mechanistic analysis of severe accidents in Metallic Fuel SFRs. The new SAS4A models track the evolution and relocation of multiple fuel and cladding components during the pre-transient irradiation and during the postulated accident, allowing a significantly more accurate description of the local fuel and cladding composition. The local fuel composition determines the fuel thermo-physical properties, such as freezing and melting temperatures, which in turn affect the fuel relocation behavior and ultimately the core reactivity and power history during the postulated accident. The models describing the fission gas behavior, fuel-cladding interaction, clad wastage formation and cladding failure models have been also significantly enhanced. The paper provides on overview of the SAS4A key metal fuel models emphasizing their new capabilities, and presents results of SAS4A whole core analyses for selected PGSFR postulated severe accidents. Figure 8 shows analysis results of molten fuel cavity evolution for the postulated severe accident.



PINACLE 28.91s	In-pin ejection 29.18s	LEVITATE 29.78	PINACLE 5.08s	In-pin ejection 5.41s	LEVITATE 5.56s
(a) LOF and TOP			(b) Peak Assembly Flow Blockage		

Figure 8 Molten fuel cavity evolution for the postulated severe accident.

3. EXPERIMENTAL PROGRAMS AND OPERATIONAL EXPERIENCES

JAEA has investigated the capability of natural circulation for core cooling in Monju during a station blackout (SBO) induced by an earthquake and a subsequent tsunami hit [6]. The plant dynamics analysis code Super-COPD was used for the investigation, which was validated by using the preliminary natural circulation test data in Monju. As a result, it was concluded that the decay heat can be safely removed by natural circulation under such an SBO condition.

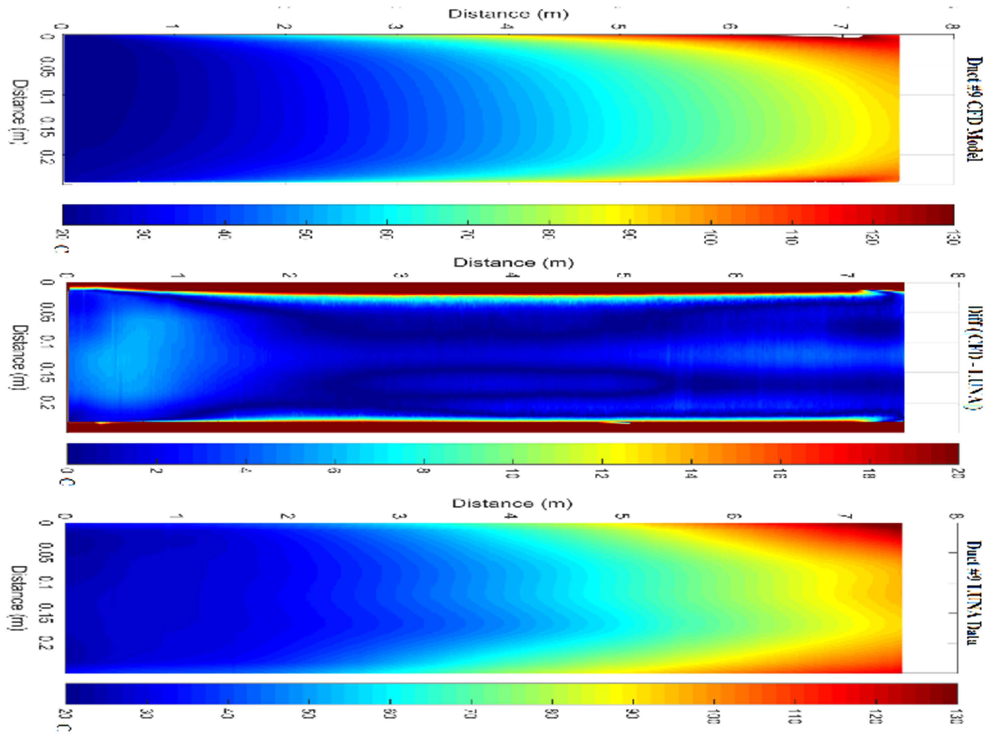


Figure 9: Riser gas temperatures. Left: CFD model, Right: Experimental data from LUNA fibers, Center: difference (error)

The U.S. Department of Energy completed testing at the air-based Natural convection Shutdown heat removal Test Facility (NSTF) at Argonne National Laboratory. After nearly four years of scaling studies, project preparations, and construction of the 1/2 scale test facility, experimental operations on the NSTF began in late 2013 with the initial fire-on and bake-out of the 220 flat plate resistance heaters. Shakedown and scoping activities were then completed in early 2014. The project team began data quality testing shortly after and has since completed over 1,300 hours of active test operations. Results comparing computational fluid dynamics simulations with high-fidelity thermal measurements are shown in Figure 9. Throughout the 20-month testing window, performance metrics were observed by varying parameters of integral power, power profile, single and dual chimneys, reduced discharge elevations, inclement weather, prototypic decay heat curves, blocked riser ducts, and adjacent inlet/outlet ports.

At steady-state conditions and prototypic power levels, the system performs its cooling related function well and is able to maintain safe limits on the reactor vessel walls regardless of design-basis faults. Disruptions due to minor blockage in riser channels and chimney ductwork (50% induced) introduce minimal rises in reactor vessel temperature (≤10°C observed) and do not pose a severe safety hazard. The system is robust to meteorological perturbations including wind excursions (24.5 m/s observed) and temperature fluctuations (−6°C – 32°C observed).

However, as with any natural circulation or chimney based system, the facility exhibits certain sensitivity to a subset of scenarios dependent on meteorological and operating conditions. Analogous to priming a chimney flue on a cold winter day, appropriate engineering controls must be made during sensitive operating times to prevent system wide instabilities that degrade heat removal performance. These sensitive operational windows have been found to be limited to: a) start-up periods when the system is still thermally and hydraulically developing, and b) periods of low power removal by the reactor vessel cooling system.

The controls used to mitigate meteorological perturbations during the test series centered on actuator valves along the chimney duct work. By introducing flow resistance on the outlet of either both or a single chimney loop, the NSTF is successfully able to overcome start-up and wind induced instabilities. Work is in progress towards examining passive alternatives to these active actuator valves and has encompassed separate effects studies on a reduced scale test facility. Several weather cap designs were tested for their effectiveness in preventing down-draft phenomena (e.g. flow reversals).

The planned objectives were achieved during the project period and resulted in a high-quality experiment test facility that is supported by a strong administrative program. The program has followed NQA-1 standards in all aspects of operation with compliance assessed during multiple audits. The test facility has successfully generated data which quantifies the heat removal performance during a wide range of operating conditions. The archived data suites are suitable to support efforts in ascertaining the viability of air-based reactor vessel cooling system concepts as a decay heat removal system for future reactor designs.

IPPE has performed experiment with modelling fuel pin failure under ULOF accident conditions in SFR. The experiment was carried out on 19-rod model assembly at the PLUTON test facility with sodium coolant. Main purposes of the experiment were as follows:

- Identification of principal mechanisms of degradation of fuel pin simulator claddings;
- Evaluation of axial material distribution in final state of the model assembly;
- Evaluation of blockage phenomena of the model assembly cross-sections;
- Estimation of material ejection outside of the model assembly.

Energy release was provided by thermite reaction:

$$Al + Fe_2O_3 = Fe + Al_2O_3$$

A specific thermal effect of thermite reaction was equal to 1.6 MJ/kg, temperature of the thermite reaction was about 3100 K. Sodium temperature in reaction zone of test section was preliminarily increased by heaters up to 550°C. Initiation of thermite reaction in fuel pin simulators of model assembly was provided by voltage supply to ignition system.

Figure 10 demonstrates a state of the model assembly after the experiment. Figure 10a shows a blockage of lower axial inlet holes of model assembly, and header zone of global degradation of rod bundle of 19-rod model assembly is presented in Figure 10b.

a) *b)*

FIG. 10. State of the model assembly after the experiment.

Three basic mechanisms of cladding degradation were identified:

- Temperature stresses in cladding material;
- Melting cladding material;
- Dynamic effects caused by fast conversion of thermal energy of fuel pin simulator corium into mechanical work during thermal interaction of corium with sodium.

Basing on the experimental results, it was evaluated that the coefficient of conversion of thermal energy of fuel pin simulator corium into mechanical work was equal to 0.115 %. Zone of global fuel pin simulator claddings degradation made approx. 65 % of the model assembly height and it was localized mainly in area of rod bundle with increased density of thermite load. Total amount of products of the thermite reaction ejected outside of the model assembly borders made 75-80 % of initial mass of thermite mixture. Almost total blockage of cross-section of the model assembly in its lower part was revealed.

5. CONCLUSIONS

In the framework of the GIF SFR Safety and Operation project, a large range of activities are being developed both on experiments and modeling by China, France, Japan, Republic of Korea, Russian Federation and Euratom.

They include the FASYS system code by China, a simplified approach and modelling of core degradation during severe accidents by CEA, PRA methodologies for external hazards and investigations on Monju Station Black Out by JAEA, analysis of Minor Actinides recycling by EURATOM, Mechanistic Source Term Assessment, developments of SAS4A for metallic fuels and tests on NSTF facility by U.S.

IPPE is performing experiments on the PLUTON facility for fuel pin failure under ULOF conditions.

The R&D programs will be extended to include common projects in the next years.

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