PROGRESS OF THRMAL HYDRAULIC EVALUATION METHODS AND EXPERIMENTAL STUDIES ON A SODIUM-COOLED FAST REACTOR AND ITS SAFETY

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ABSTRACT

In the framework of the Generation-IV International Forum, the safety design criteria (SDC) has been established incorporating safety-related R&D results on innovative technologies and lessons learned from Fukushima Dai-ichi nuclear power plants accident in order to provide the set of general criteria for the safety designs of structures, systems and components of Generation-IV Sodium-cooled Fast Reactors (Gen-IV SFRs). A number of thermal-hydraulic evaluations are necessary to meet the concept of the criteria in the design studies of Gen-IV SFRs. In this paper, the authors focus on four kinds of thermal-hydraulic issues associated with the SDC, i.e. fuel assembly thermal-hydraulics, natural circulation decay heat removal, thermal striping phenomena, and core disruptive accidents, and provide a description of their evaluation method developments including V&V and necessary experimental studies for the Japan Sodium-cooled Fast Reactor (JSFR). These evaluation methods are planned to be eventually integrated into a comprehensive numerical simulation system that can be applied to all phenomena envisioned in SFR systems and that can be expected to become an effective tool for the development of human resource and the handing down of knowledge/technologies.

KEYWORDS
Sodium-cooled fast reactor, safety design criteria, fuel assembly, natural circulation decay heat removal, thermal striping, CDA, experiment, numerical simulation, V&V

1. INTRODUCTION

In the framework of the Generation-IV International Forum (GIF), much effort has been devoted to the establishment of the "Safety Design Criteria (SDC)" for the Sodium-cooled Fast Reactor (SFR) system. The objective of the SDC is to provide the set of general criteria for the safety designs of structures, systems and components (SSCs) of the Gen-IV SFR system, where the criteria are clarified systematically and comprehensively consistent with the GIF's basic safety approach and with the aim of achieving the safety and reliability goals defined in the GIF Roadmap [1],[2]. The SDC has been established by maintaining the basic structures of texts in the IAEA SSR 2/1 [3], by considering the specific features of SFR system, and by incorporating lessons learned from Fukushima Dai-ichi nuclear power plants (NPPs) accident as shown in Fig.1, where eighty-three criteria for the overall plant design and specific SSC designs are described. It is expected that the SDC would be disseminated, updated and utilized for SFR design at international level and such activities have been started in interactions with research and design organizations of SFRs, regulatory bodies and their technical support organizations and international organizations such as IAEA and OECD/NEA[4].

The SDC defines the safety approach based on basic characteristics of the SFR as follows:

- Due considerations on the possibility of a positive void reactivity in the center area of the reactor core are necessary for the reactor core design to have an inherent reactivity feedback to control reactor power and passive reactor shutdown capability, and to have as well inherent features to prevent the re-criticality leading to significant mechanical energy release during a hypothetical core disruptive accident.
- Liquid sodium has high thermal conductivity and high boiling temperature at atmospheric pressure. Hence, decay heat removal is possible using natural circulation due to the favorable coolant characteristics. However, sodium is chemically active and it is therefore necessary to manage sodium leaks (sodium fire on contact with air and reaction with water or concrete).
- As an SFR operates at a relatively high temperature compared to a Light Water Reactor (LWR) and in high fast neutron fluence conditions, due consideration should be paid to thermal striping and thermal shock phenomena as well as creep and radiation effects on fuel and structural materials.
- As an SFR is operated under low pressure conditions, coolant leakage does not lead to the type of loss of coolant accident anticipated in an LWR with depressurization, coolant boiling and the loss of cooling capability. Therefore, an emergency core cooling systems for coolant injection under high and low pressure conditions, as used in the LWR, are not necessary in an SFR. The only requirement for SFR core cooling is the retention of the sodium coolant level above the reactor core in the reactor vessel along with sufficient heat removal capability.

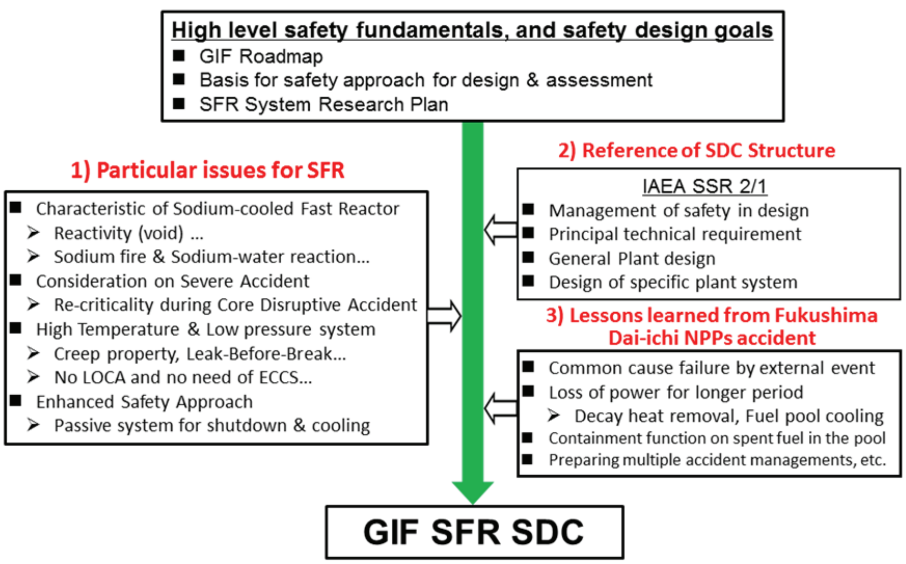


Figure 1: Basic scheme to outline the SDC [5]
(Figure 3: Basic Scheme to Outline the SDC, Page 9 of Reference 5)

In the design study of the Gen-IV SFR systems, a number of thermal hydraulic evaluations are necessary to meet the SDC and to confirm the feasibility of the designs. In this paper, the authors focus on 1) fuel assembly thermal hydraulics, 2) natural circulation decay heat removal, and 3) core disruptive accidents (CDAs) among thermal-hydraulic issues that are associated with the SDC and provide a description of their evaluation method developments including V&V and necessary experimental studies for Japan advanced Sodium-cooled Fast Reactor (JSFR) [6]. In addition, a newly developed V&V procedure intended for a numerical simulation method of thermal striping phenomena is introduced.

2. FUEL ASSEMBLY THERMAL HYDRAULICS

In the design of Gen-IV SFR systems, it is important to assure high levels of safety and reliability along with achieving economic competitiveness with future light water reactors. In the core design of the JSFR, installation of a rhombic prism-shaped inner duct inside a fuel assembly is being considered as one of the countermeasures against CDAs as well as high burn-up and high linear heat rate for enhancing economic competitiveness. In order to confirm the core design feasibility, the clarification of thermal-hydraulic phenomena in fuel assemblies is necessary under various operating conditions such as normal operation, transient/accident condition, or deformed pin-bundle geometry condition from the viewpoint of the assessment of fuel pin structure integrity. Japan Atomic Energy Agency (JAEA) has been developing a numerical simulation system that offers methodologies to clarify thermal-hydraulic phenomena in a fuel assembly and that can substitute for mock-up experiments as much as possible. In parallel, several fundamental experiments have been also conducted to create database for modeling and code validation.

2.1. Numerical Simulation System for Wire-wrapped Fuel Pin Bundle

Figure 2 shows the numerical simulation system for thermal hydraulic analyses in a wire-wrapped fuel pin bundle. This system consists of three kinds of thermal hydraulic analysis codes with fuel deformation analysis code. The first is a subchannel analysis code ASFRE [7], that uses more empirical correlations in the physical modeling and is applied to whole fuel assembly simulations. This code is mainly used for design parameter analyses because of its high computation efficiency. The second is a finite element analysis code SPIRAL [8]. This code contributes to detailed simulations of local flow and temperature fields in a fuel assembly. This code also has a role to offer the thermal-hydraulic correlations to ASFRE. For instance, when ASFRE simulates temperature fields in a deformed fuel pin bundle with the help of the fuel deformation simulation code BAMBOO [9], SPIRAL can offer the data for improving the empirical correlations of ASFRE that are applicable to the deformed subchannels. The third is a direct numerical simulation (DNS) code [10] that offers fundamental data to improve turbulence models incorporated in SPIRAL. As one can easily imagine, the simulation cost is extremely increased from the first to the third simulation and it is apparently impossible to apply DNS to the whole assembly simulation due to the limitation of the current computer capability. Therefore, such interactive use of the three simulation codes that we propose is a practical way. BAMBOO is a fuel deformation simulation code and it can calculate fuel pins and wrapper-tube deformation due to thermal expansion and irradiation effect (swelling). Combination use of ASFRE and BAMBOO enables to simulate thermal hydraulic phenomena coupled with deformation.

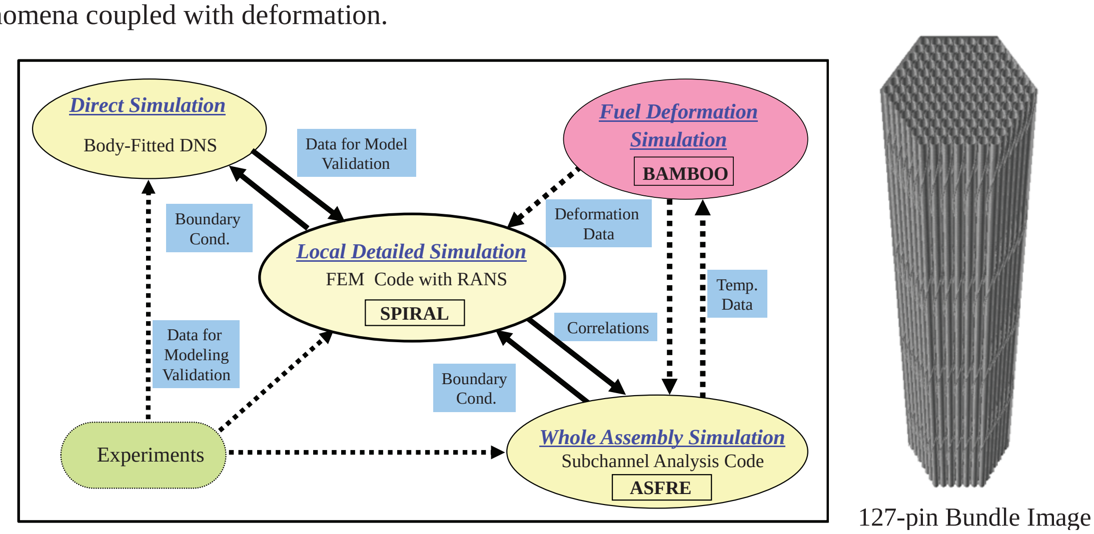


Figure 2: Numerical simulation system for thermal hydraulic analyses of a fuel assembly

2.2. Experimental Study and Code Validation

A number of water and sodium experiments from fundamental level to mock-up have been carried out in JAEA to obtain data for the clarification of in-core phenomena, derivation of thermal-hydraulic correlations, mechanistic modeling and component code validation of the above-mentioned numerical simulation system. As an example, a fundamental experiment and its numerical analysis is introduced. A wire-wrapped 3-pin bundle water model was applied to investigate the detailed velocity distribution in an inner-subchannel surrounded by three pins with the wrapping wire [11]. Figure 3 illustrates the image of the test section. The test section consists of an irregular hexagonal acrylic duct tube and fluorinated resin pins which have nearly the same refractive index with that of water and a high light transmission rate. This refractive index matching enables to visualize the inner subchannel through the outer pins. The velocity distribution in the inner subchannel with the wrapping wire was measured by PIV (Particle Image Velocimetry) through two sides of the duct tube. Typical flow velocity conditions in the pin bundle were 1.6m/s (Re = 13,500) and 0.36m/s (Re = 2,700). Feature of stream regime in the subchannel existing wrapping wire was visualized in vertical and horizontal plane. The time averaged velocity field in the horizontal plane was reconstructed from the two vertical plane data in different directions.

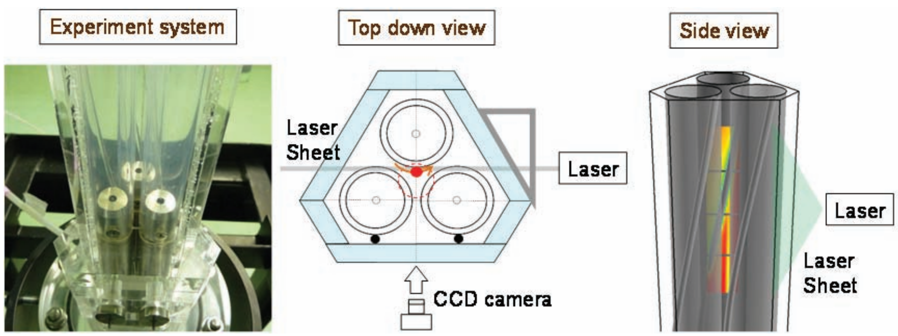
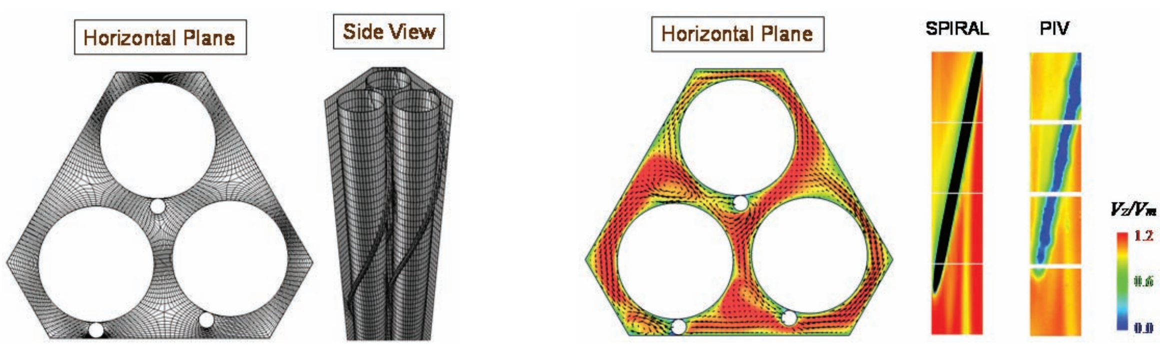


Figure 3: Test section of deformed 3-pin bundle water experiment

SPIRAL was applied to the experimental analysis as one of the validation studies. Figure 4 shows the computational mesh scheme and examples of the results with the measured data. It was confirmed that SPIRAL can reproduce the flow field in the complicated geometry.



(a) Computational mesh scheme (b) Example of numerical simulation results with measured data

Figure 4: Computational mesh scheme and simulation results with measured data

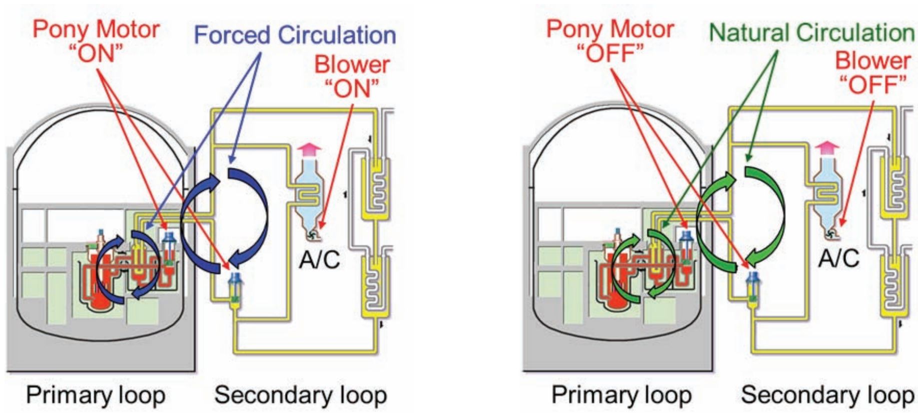
3. NATURAL CIRCULATION DECAY HEAT REMOVAL

3.1. Passive Decay Heat Removal of SFRs

In most of the previous and existing SFRs, such as the Japanese experimental SFR JOYO and the prototype SFR MONJU, the decay heat removal after a scram is generally relied on forced circulation (FC) by using active components such as pony motors and blowers as shown in Fig. 5 (a). However, the loss of all AC power could occur as the case of the Fukushima Dai-ichi NPPs accident. In such an event, SFRs can maintain the function of decay heat removal by natural circulation (NC) which is induced by buoyancy effect due to large temperature difference of the coolant in the heat transport system as shown in Fig.5 (b). Furthermore, NC cooling can be available even in CDAs for post-accident heat removal (PAHR).

3.2. Experimental Study on Natural Circulation Decay Heat Removal

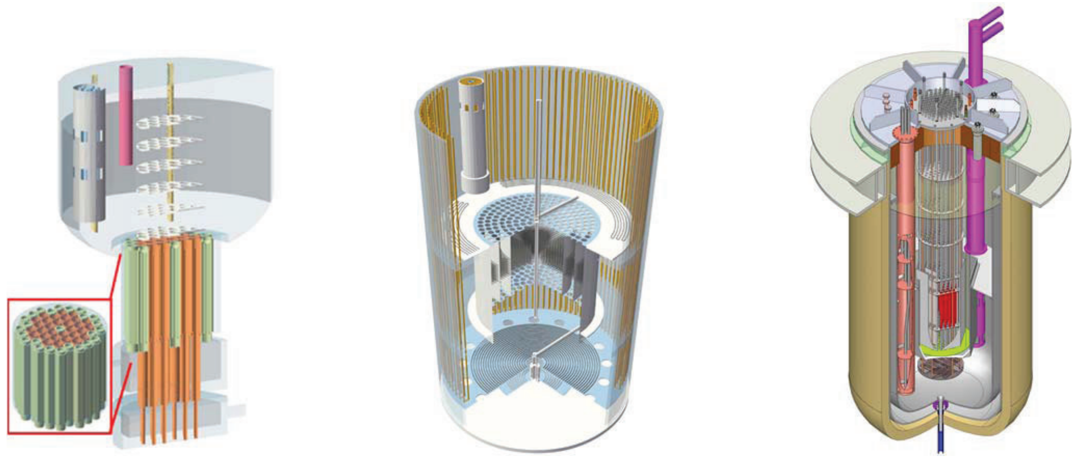
A number of studies related to NC cooling have been conducted for safety assessments of SFRs. NC transient tests at JOYO have clearly demonstrated that decay heat can be removed with NC in the primary and the secondary cooling systems [12]. In the design study of the JSFR, adoption of a fully passive NC decay heat removal system is being considered as one of the innovative technologies. A 1/10-scale water test of the JSFR was carried out and the results indicated that NC cooling works well in various events such as loss-of-offsite-power, sodium leakage in a secondary loop, and primary pump stick [13]. Then, a 1/5-scale sodium test with PLANDTL test facility of JAEA clarified that the primary reactor auxiliary cooling system (PRACS) using NC can enhance the robustness of the SFR safety against various external events [14].



(a) FC cooling (b) NC cooling

Figure 5: Decay heat removal operations in MONJU reactor

Currently, a series of experiments named "AtheNa-SA" are planned in JAEA for the clarification of decay heat removal and related thermal hydraulic phenomena in a reactor vessel under severe accident conditions. Figure 6 illustrates images of three kinds of test apparatuses. Elemental sodium tests for evaluating inter-wrapper flow effect caused by dipped heat exchanger (DHX) operation in the upper plenum on decay heat removal from the core region will be conducted with PLANDTL-II (modified PLANDTL), which is going to be modified in 2016. A 1/10-scale water test apparatus PHEASANT, which was built in 2015, contributes to understand complex thermal hydraulics in the reactor vessel especially when both DHX cooling and ex- vessel cooling (cooling through the reactor vessel) are available by detailed velocity and temperature field measurement. Furthermore, integrated tests will be conducted with a large scale sodium test apparatus AtheNa-RV, whose conceptual design is in progress.

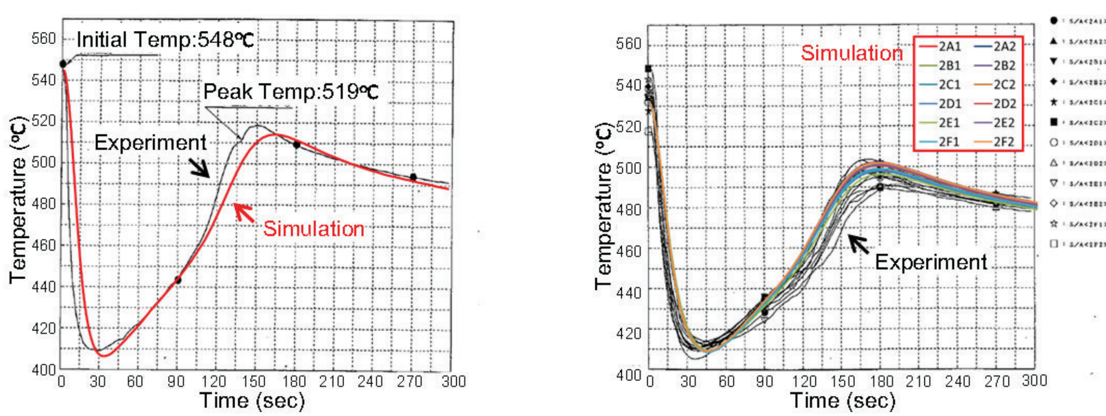


(a) PLANDTL-II (b) PHEASANT (c) AtheNa-RV

Figure 6: Experimental apparatuses of AtheNa-SA test series

3.3. Evaluation Tools for Natural Circulation Decay Heat Removal

A set of evaluation tools has been developed to ensure the safety of the JSFR which has a fully NC decay heat removal system [15]. The one-dimensional safety analysis method can evaluate the core hot spot temperature taking into account the temperature flattening effect due to flow redistribution caused by the buoyancy force and inter-assembly heat transfer in the core. The three-dimensional fluid flow analysis method can evaluate complicated phenomena like thermal stratifications and the scale effect due to the difference of dimensions between the JSFR and the test apparatuses, because it can exactly deal with the geometry and has turbulent models as well [16]. One-dimensional safety analysis method has been improved to reflect the validation analysis results using the water and sodium tests mentioned before in Section 3.2. In addition, real plant data of JOYO, EBR-II, and other SFRs are also being utilized to



(a) Center of core (zero layer) (b) Second layer from core center

Figure 7: Comparison of coolant temperature change at fuel assembly outlet in JOYO NC decay heat removal test between simulation results and measured data

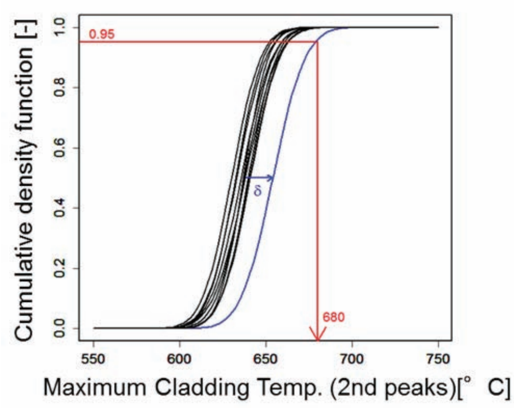


Figure 8: Statistical evaluation of core hot spot temperature during NC operation

investigate the scale effect. As an example, the comparison of the simulation results with the measured data related to the JOYO Mk-II core NC test is shown in Fig. 7. The simulated outlet temperature histories of fuel assemblies are in good agreement with the experimental data.

Furthermore, a statistical method has also been developed for evaluating hot spot temperature in the core [17]. Figure 8 shows the cumulative density functions of the maximum fuel cladding temperature at the secondary peak for the loss-of-offsite-power case of the JSFR. The core hot spot temperature with 95% probability can be evaluated by the blue colored curve which includes modeling errors.

4. CORE DISRUPTIVE ACCIDENTS

4.1. CDA evaluation code

CDAs have been a major concern in the safety of SFRs because of the energetics potential resulting from a recriticality event. After the Fukushima Dai-ichi NPPs accident, the CDA evaluation in beyond design accidents is recognized as a regulatory requirement for commercial LWRs in Japan as well as the existing prototype fast reactor. Mechanistic simulation of an accident sequence during a CDA is required to realistically assess the energetics potential and coolability after the accidents. SIMMER code is a comprehensive computational tool for the CDA evaluations that systematically models coupled multiphase thermal-hydraulic and space-dependent neutronic phenomena [18]. The basic framework of the SIMMER code, validation of the code and the application to CDA evaluation are summarized in this Chapter.

4.2. Overall framework of SIMMER code

A conceptual overall framework of SIMMER is shown in Figs.9 and 10. The SIMMER code models the five basic LMFR core materials: fuel, steel, sodium, control rod, and fission gas. A material can exist as different physical states; for example, fuel needs to be represented by fabricated pin fuel, liquid fuel, a crust refrozen on structure, solid particles, and fuel vapor, although fission gas exists in the gaseous state. Thus, the material mass distributions are modeled by 27 density components in the SIMMER. The energy distributions are modeled by only 16 energy components since some density components are assigned to the same energy component. For example, a mixture of different vapor components is defined by a single energy.

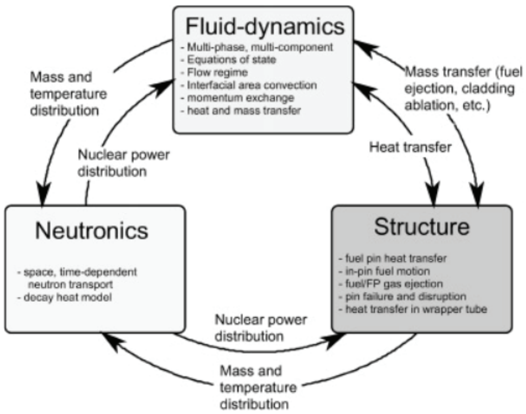


Figure 9: Overall framework of SIMMER code

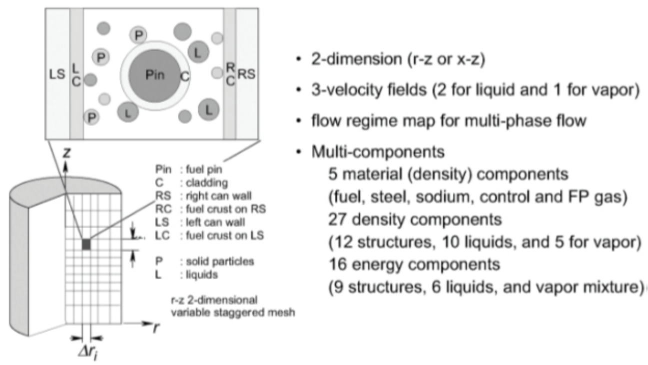


Figure 10: Fluid dynamics model in SIMMER

4.3. Code Validation

Key phenomena relevant to CDA evaluation in SFRs were systematically identified in the Phase 2 code assessment of SIMMER [19] as boiling pool dynamics, fuel freezing and relocation, fuel-coolant interactions (FCI), material expansion dynamics and core neutronics. The points of issues to be validate for code models were ranked comprehensively and assessed the applicability for CDA evaluation (Table 1).

Table 1 Matrix of code validation for SIMMER

Key phenomena	Issues and models to be validated
1. Boiling pool dynamics	<u>Boiling pool of molten fuel and molten steel</u> > Heat transfer between molten fuel and other components > Non-equilibrium vaporization/condensation for multi components > Source term of interfacial area <u>Small-scale boiling pool of simulant</u> > Turbulent effect > Momentum exchange between bubbles and heavy liquid
2. Fuel freezing and relocation	<u>Melting/freezing in pin-bundle channel</u> > Heat transfer between melt and structure, melting/freezing > Viscosity effect by solid particle <u>Melting/freezing in circular tube</u> > Heat transfer between melt and structure, melting/freezing
3. Fuel-coolant interactions (FCI)	<u>FCI (between fuel and sodium)</u> > Melt penetration into coolant > Non-equilibrium vaporization/condensation for multi component > Source term of interfacial area
4. Material expansion dynamics	<u>Energy reduction by structure, and large bubble behavior</u> > Effect of non-condensable gas on condensation > Entrainment due to instability of bubble surface
5. Disrupted core neutronics	<u>Reactivity change due to molten-fuel redistribution</u> > Kinetic space dependent neutron transport

Code validation example for a boiling-pool dynamics phenomenon is indicated in Fig. 11. The experiments used for the validation is an in-pile boiling-fuel-pool experiment in the SCARABEE BF2. In this experiment, UO_2 powder was nuclear-heated to melting and further above the boiling point, and behaviors of a molten and boiling fuel pool were measured through the cover gas pressure and reactivity change in the driver core. The SIMMER was able to express the axial heat-flux distribution along the crucible and other data such as the oscillation of the pool surface so on [20].

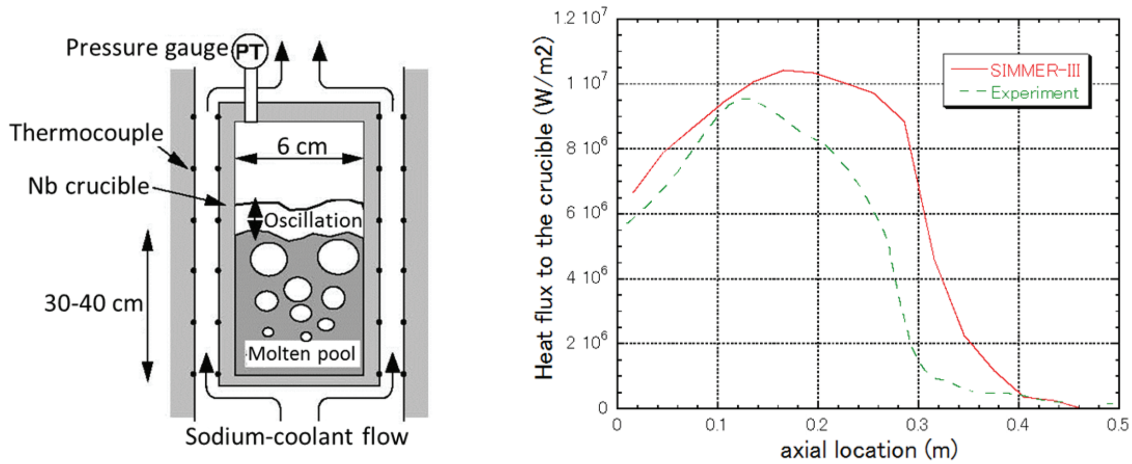


Figure 11: Schematic of SCARABEE BF2 and reproduction of heat-flux distribution on crucible wall [20]

4.4. Application for the CDA evaluation

Considering fast reactor core characteristics, anticipated transients without scrum (ATWS) is one of the significant accident sequences for CDA evaluation. Figure 12 shows the results of the recriticality evaluation for the prototype fast breeder reactor in the transition phase of the ULOF (unprotected loss-of-flow) sequence which is the representative one in the ATWS events for SFR. The analysis was performed with a realistic reference condition by using the SIMMER-IV code which has the most updated analytical models [21]. In this evaluation, the reactivity would not reach recriticality in the transition phase under the reference condition. In the early stage of the transition phase (up to about 28.0 sec), a power transient with a maximum reactivity of 0.94 $ would be caused due to the falling of upper dispersed fuel. The failure of CRGT structure and the formation of discharge path would take place by 30 sec for all CRGTs, and the molten fuels in the core region would be discharged through CRGTs. The fuel discharge through CRGTs would result in a remarkable subcritical state with about -50$ in reactivity. The fuel inventory remaining in the core region would ultimately (after 35 sec) reach about 50%, and that discharged below core bottom (including a region of low pressure plenum), above core top, and into core periphery would ultimately be about 20%, 15%, and 15%, respectively.

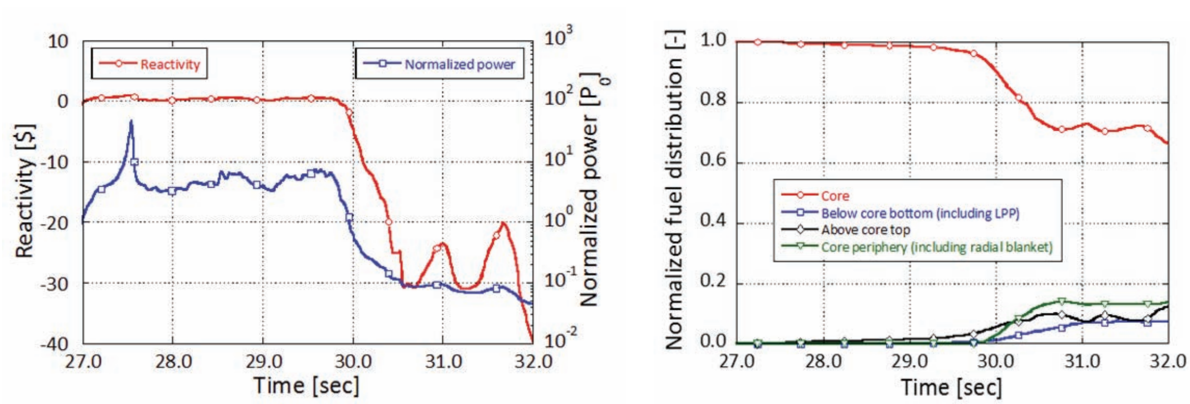


Figure 12: Reactivity/power transient and fuel distribution in transition phase by SIMMER [21].

5. THERMAL STRIPING PHENOMENA

5.1 Development of Procedure for V&V (V2UP) and Numerical Prediction

In the design of the JSFR, the Core Instruments support Plate (CIP) as the lowest and perforated plate of the Upper Internal Structure (UIS) is installed in the upper plenum of the reactor vessel. Below the CIP, hot sodium from fuel assemblies (FSs) mixes with cold sodium from control rod (CR) channels and also radial blanket fuel assemblies (RBFSs) and generates temperature fluctuation in the fluid that may cause the high cycle thermal fatigue on an adjacent structure. Therefore, this thermal striping phenomenon is one of the significant issues in the viewpoint of the design feasibility. In JAEA, a numerical estimation method for the thermal fatigue in the JSFR has been developed. In the development of numerical simulation codes and estimation methods, implementation of V&V is indispensable to make successful numerical predictions and evaluation of the physical problems. A procedure named V2UP (Verification and Validation plus Uncertainty quantification and Prediction) covering the V&V and the process of the numerical prediction including uncertainty quantification was proposed [22].

5.2 V2UP Procedure and Components

Figure 13 shows a flow chart of the V2UP procedure consisting of five components. In Component-I, the V2UP was initiated by the PIRT analysis. In Component-II, the development plan of the numerical simulation codes and methods and also the plan of the V&V are made and implemented. In Component-III, experiments are designed and arranged to meet the V&V objectives. And the uncertainties are integrated in Component-IV, and numerical prediction and estimation are finally performed in Component-V.

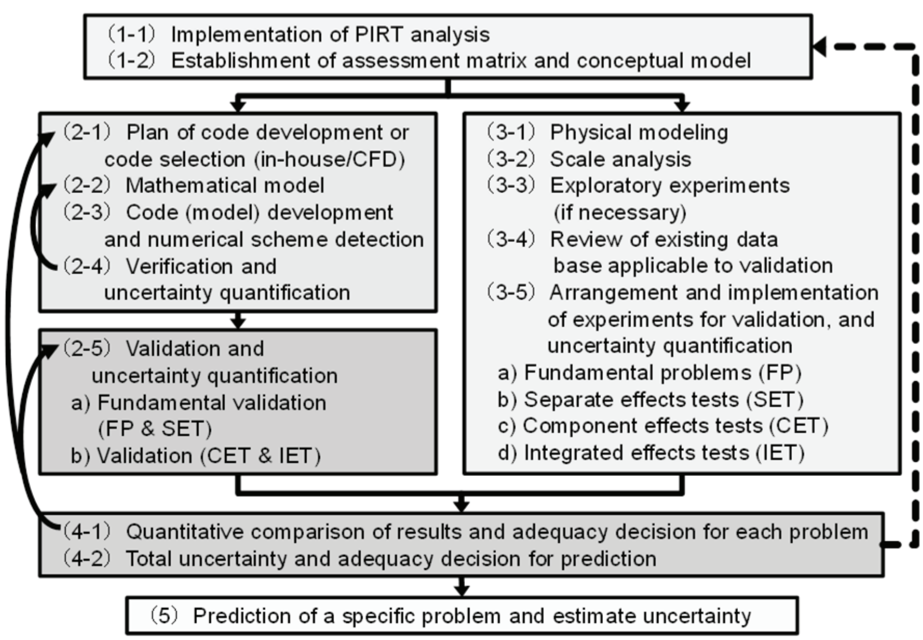


Figure 13: Flow chart of V2UP (Verification and Validation plus Uncertainty quantification and Prediction)[22]

5.2.1 Component I: Identification of Issue and Requirements for Estimation

The PIRT process in V2UP is based on the simplified nine steps PIRT by the US-NRC [23]. Through the PIRT process, conceptual model for the numerical estimation, as shown in Fig. 14 for example of thermal striping, can be constructed. To estimate thermal mixing around the RBFSs affected by the flow outside of the UIS in the upper plenum, the spatial connection analysis model consisting of the whole upper plenum analysis and the local area analysis is considered. In the spatial connection model, boundary conditions on the side surfaces (boundaries) of the local analysis model are provided by the numerical results of the upper plenum simulation. Transient data of the temperature in the structures obtained by the local area analysis is provided to the thermal stress analysis in the structures.

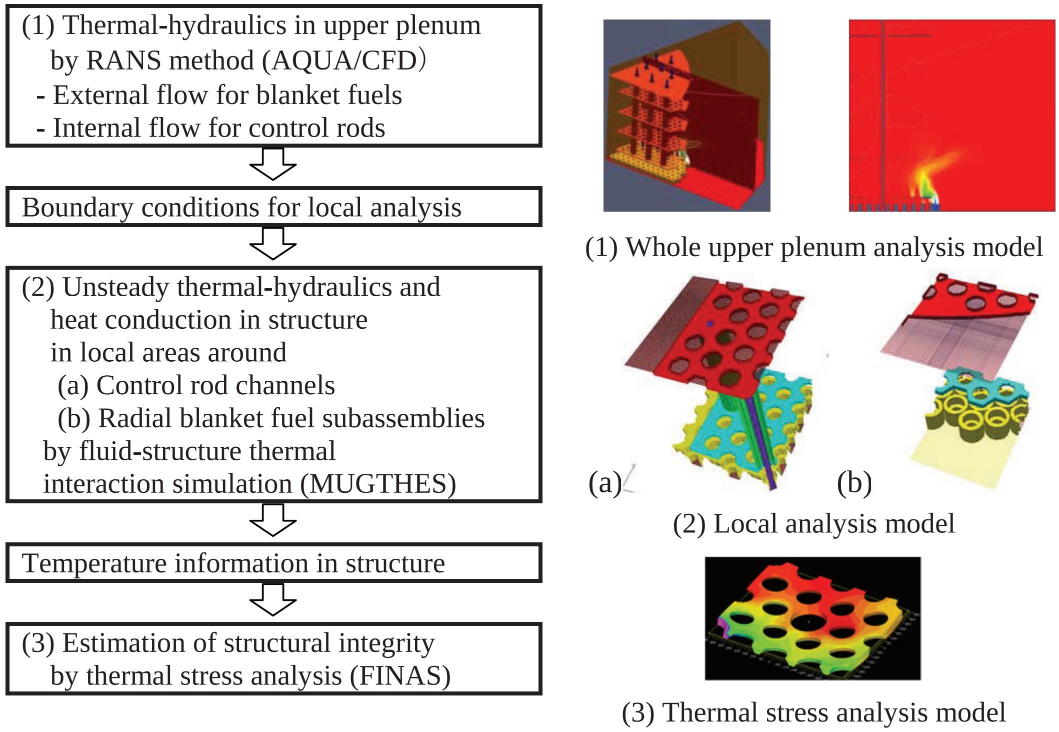


Figure 14: Concept model for numerical estimation of high cycle thermal fatigue in JSFR [22]

5.2.2 Component II: Verification and Validation plus Uncertainty Quantification

Numerical simulation codes to be utilized for the estimation of the high-cycle thermal fatigue issue of the JSFR are also shown in Fig. 14. AQUA [24] is a finite differential code for a multi-dimensional thermal-hydraulics analysis with Reynolds Averaged Navier-Stokes Simulation (RANS) approach. MUGTHES [25] employs the Large Eddy Simulation (LES) approach to predict unsteady thermal mixing phenomena and is designed to simulate the thermal interaction between unsteady thermal-hydraulics and unsteady heat conduction in structure simultaneously with the conjugate heat transfer model to provide the temperature histories in the structure. A numerical simulation code FINAS for structural mechanical stress analysis is used to estimate thermal stress in the structure by using the temperature histories in the structure [26]. In the standard of the ASME V&V-20 [27], the solution verification by using the Grid Convergence Index (GCI) method was introduced to quantify the uncertainty of the numerical results. Since limitation of the Roache's GCI method in the ASME guideline was detected, application of the least square version GCI originally established by Eça and Hoeksta was examined and a modified method named Simplified Least Square version GCI (SLS-GCI) was established [28]. As for the validation process, applicability of Area Validation Metrics (AVM) and the modified AVM (MAVM) were potentially indicated to the numerical results of thermal mixing problem in the T-junction piping system [29].

5.2.3 Component III: Experiments Development for the V&V

In the V2UP, the fundamental problems (FPs), the separated effect tests (SETs), the component effect tests (CETs), and the integrated effect tests (IETs) are defined. In the V2UP, the validation process is categorized into two steps; the fundamental validation consisting of the FPs and the SETs and validation process with the CETs and the IETs. After confirmation of comprehensive capability of the numerical simulation codes and methods for the target issue through the validation process, numerical prediction of the target phenomena is to be conducted. For the high cycle thermal fatigue issue in the JSFR, the experiments for the V&V are shown in Table 2 in a hierarchical formation. The SETs as element experiments in which include typical phenomena related to the target issue are chosen from the existing experiments. At present, the several tests of jets mixing phenomena (e.g. WAJECO and PLAJEST [30]) and also thermal mixing phenomena in a T-junction piping system (WATLON [31]) and TECREC [32] also been conducted. Moreover, full scale two jets configured water and sodium test apparatuses, TIWAT and TISOT, respectively, simulating a FS and a CR channel or a RBFS are to be planned to estimate the scale effect. As for the IET, a water test with 1/3 scaled 1/6-sector partial model (TAFUT [34]) for the upper plenum of the JSFR has been launched.

Table 2: Tentative hierarchical table of experiments for the V&V implementation [22]

System: Sodium-cooled Fast Reactor (JSFR)			*Target issue*: High cycle thermal fatigue in JSFR
Sub-system: Upper plenum	**IET**	- Existing (partial) model experiments for existing plants (1/1 or subscale) - 1/6 sector model test (1/3 scale) : **TAFUT** (water)	- Similarity of geometry - Applicability of numerical estimation methods including scale effect.
Component: FSs, RBFSs, CR channels, CR shaft and a part of CIP	**CET**	- Two jets test apparatus (1/1 scale) : ***TIWAT*** *(water) /* ***TISOT*** *(sodium)* [*in planning*] - Five jets test apparatus (1/3 scale) : **FIWAT**(water) / **FISOT** (sodium)	- Similarity of geometry - Applicability of numerical schemes, models and mesh arrangement including scale effect. - Proposals of mitigation measures for design
Elemental Phenomena : Thermal mixing phenomena and conjugate heat transfer	**SET**	- Triple jets (free jets / thermal interaction): **WAJECO** (water) / **PLAJEST** (sodium) - Dual jets of coaxial and parallel configurations (free jets / thermal interaction): (sodium / air / water) - Single jet (free jets / thermal interaction): - T-pipe (with / without thermal interaction) **WATLON** (water) / **TECREC** (water)	- Appropriate numerical schemes, turbulence model (LES) for unsteady motion of large scale eddies and conjugate heat transfer model - Confirmation of potential of the code for target issue
Fundamental problems	**FP**	Problems are to be identified through the PIRT (e.g. pipe flow, separation and wake flows, squire cavity flows, flows through orifice)	- Confirmation of correctness of numerical schemes, models in the code

5.2.4 Preparation and Implementation of Prediction

Through the V&V processes of Component-I, -II and -III, applicable numerical simulation code and numerical estimation method are prepared to predict the target issue in the full scale plant. And uncertainties are obtained from the CETs and IETs. Uncertainties in each problem should be integrated into one value concerning to the System Response Quantity (SRQ). Much more detailed investigations are needed to establish the method of integration of the uncertainties derived from the each examination in the V&V process.

6. SUMMARY AND CONCLUDING REMARKS

The SDC has been established in the framework of the GIF and is expected to be utilized for Gen-IV SFR design at international levels. The authors focused on four kinds of important phenomena among thermal hydraulic issues associated with the SDC and provided a description of their evaluation method developments for the JSFR design.

In order to evaluate fuel pin structural integrity under various operating conditions, a numerical simulation system for thermal hydraulics, deformation and their interaction in wire-wrapped fuel pin bundle has been being developed in JAEA. This system offers flexible and practical methodologies to clarify complicated phenomena in a fuel assembly. One can select single or coupling use of element programs of the system, depending on the particular issue to be evaluated. V&V of the element programs of the system is being conducted systematically. Validation of the interaction modeling between thermal hydraulics and deformation due to swelling and thermal expansion remains a big challenge. This numerical simulation system is expected to serve as a powerful tool for “design by analysis”.

In regard to NC decay heat removal that is one of significant advantages of SFRs, three kinds of evaluation methods have been developed for the JSFR; a one-dimensional safety analysis method which can evaluate the core hot spot temperature taking into account the temperature flattening and inter-assembly heat transfer effects in the core, a three-dimensional fluid flow analysis which can evaluate the thermal-hydraulics for local convections and thermal stratifications in the primary system and the decay heat removal system, and a statistical safety evaluation method for the hot spot temperature in the core. The safety analysis method and the three- dimensional analysis method have been being validated using water/sodium NC experimental data as well as real plant data of JOYO, EBR-II, and other SFRs. Currently, a series of experiments named “AtheNa-SA” are planned in JAEA for the clarification of decay heat removal and related thermal hydraulic phenomena in a reactor vessel under severe accident conditions.

A comprehensive computational tool that systematically models coupled multiphase thermal-hydraulic and space-dependent neutronic phenomena has been developed in JAEA for the evaluation of CDAs in SFRs. This tool can mainly treat boiling pool dynamics, fuel freezing and relocation, FCI, material expansion dynamics, and core neutronics as key phenomena of the CDAs. Systematic validation studies have been conducted using data of fundamental, out-of-pile and in-pile experiments. Because CDAs in themselves are complex phenomena, the confirmation process of the applicability of each model to actual phenomena in SFRs is important.

The framework of the V2UP procedure was shown for thermal striping phenomena evaluation method and processes at each step were briefly described. Through the PIRT analysis in Component-I, the target issue was clearly defined and the conceptual model could be established. Effective implementation of the V&V using the SLS-GCI and AVM/MAVM method is expected in Component-II. In Component-III, existing experiments were successfully classified in the hierarchical table with appropriate definition to the FP, SET, CET and IET. However, practical investigations and examinations along the V2UP procedure including uncertainty quantification and scaling analysis are required to make a robust framework.

The evaluation methods for sodium fire and sodium-water reaction phenomena, as the particular issues of SFRs, have been also developed in JAEA, although they were not picked up in this paper. The evaluation methods for all important issues related to design feasibility and safety are planned to be eventually integrated into a comprehensive numerical simulation system that can be applied to all phenomena envisioned in SFR systems. In parallel, experimental studies are continuously conducted for creating a database to offer quality data for modeling and its validation. These activities can be also expected to provide opportunities or tools effective for the development of human resource and the handing down of knowledge/technologies.

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