Current Thermal Hydraulic Activities on Sodium-cooled Fast Reactors in Japan

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Abstract. Focusing on key issues on thermal hydraulics of sodium-cooled fast reactors (SFRs), namely, natural circulation decay heat removal, gas entrainment, sodium combustion, and sodium-water chemical reaction, this paper reviews the progress of evaluation methods on them. These issues are identified as safety concerns of SFRs in the safety design criteria (SDC) and safety design guidelines (SDG) for Generation-IV SFRs. To meet the SDC and SDG, design studies and related researches are being carried out in an SFR development project in Japan. The goal of the thermal hydraulic studies is to assure higher level of safety and reliability along with economic competitiveness with concurrent light water reactors.

Key Words: Safety design criteria, Safety design guideline, Thermal hydraulic evaluation method, Current activities in Japan

1. Introduction

A development project of the Japanese Sodium-cooled Fast Reactor (JSFR) [1] has been carried out for the commercialization of the next generation reactors. Especially after the Fukushima Daiichi nuclear power plant accident, we have recognized that the safety initiative is more significant than ever in the next generation reactors development. Therefore, general safety requirements are to be applied to the next generation reactors in common to the fleet of current generation reactor. It is emphasized that appropriate safety design requirements should be established so that the characteristics of each reactor design are adequately reflected and well-balanced performance is achieved. Accordingly, safety design criteria and guidelines are necessary for directing the SFR development project. In the framework of the Generation-IV International Forum (GIF), efforts have been made to establish the consistent safety standards, “Safety Design Criteria (SDC)” and “Safety Design Guidelines (SDG)” for the SFR systems.

Thermal hydraulics in an SFR is different from the one in a light water reactor (LWR). First, an SFR can maintain the decay heat removal function even under a loss of AC power event i.e. Station Blackout (SBO) by natural circulation (NC) of sodium induced by buoyancy force due to large temperature difference of the coolant in the heat transport system. Second, different types of fuel subassembly concepts are adopted. The reactor core power density of an SFR is 400-1,000 MW/m^3 that is 7-10 times larger and the coolant thermal conductivity is 200 times larger than LWRs. Therefore, the highest temperature in the core (the hot spot temperature) is a concern in safety analyses. Third, an SFR has a low pressure single-phase coolant system and free surface exists in the reactor vessel. The entrained gas bubbles may cause reactor power disturbance in the core and degradation of heat transfer performance in the intermediate heat exchanger. Design studies have been conducted to stabilize free surface hydraulic fluctuations. Lastly, sodium has chemical reactivity with oxygen and water. Hence sodium combustion and sodium-water reaction need to be prevented.

The thermal hydraulic characteristics of the SFR discussed here should be taken into account in the safety requirements and guidelines, namely, the SDC and SDG. A number of thermal hydraulic evaluations have been performed for the feasibility and design studies of the Generation IV (Gen-IV) SFR systems in conformity with the SDC and SDG. In this paper, the authors focus on 1) natural circulation decay heat removal, 2) gas entrainment, 3) sodium combustion, and 4) sodium-water reaction among thermal-hydraulic issues that are associated with the SDC and SDG. Descriptions of their evaluation method are discussed with experimental studies fulfilled in the JSFR project.

2. Safety Design Criteria and Safety Design Guidelines

The objective of the SDC is to provide a set of general criteria for the safety designs of Gen-IV SFR systems. The criteria are clarified systematically and comprehensively in consistent with the GIF's basic safety approach and with the safety and reliability goals defined in the GIF Roadmap [2, 3]. The SDC has been established on the basis of the fundamental structures of the IAEA SSR 2/1[4] which reflects characteristics of the SFR systems and lessons learned from the Fukushima Daiichi nuclear power plant accident [5]. The SDC consists of 83 criteria for the overall plant design and specific designs of structures, systems and components. It has been approved by the GIF in May 2013 and is currently under review by GIF organizations and regulatory bodies of the SFR system committee partners of the GIF.

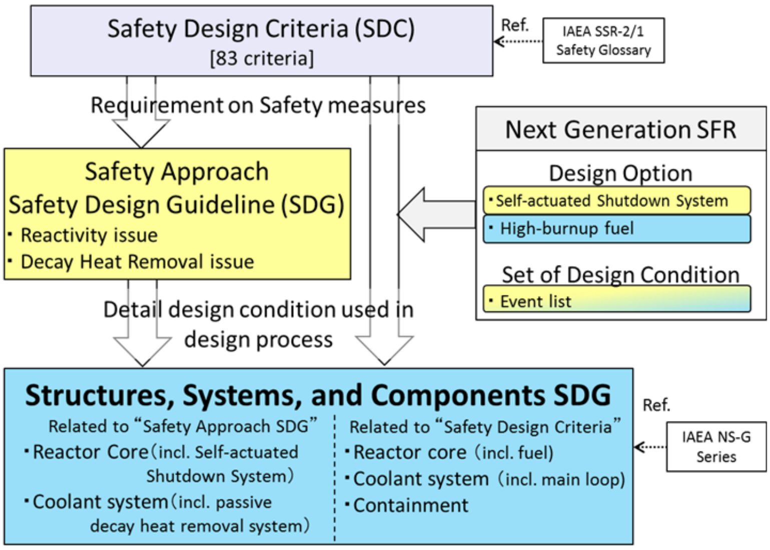


FIG. 1. Basic scheme to outline the SDG [8]

(Figure 2: Process of specification in SDG, Page 86 of Reference 8)

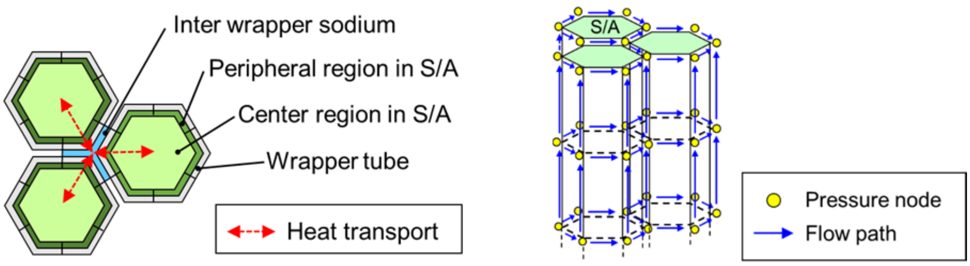
A harmonious collaboration with the member countries vigorously promoted the development of the SDG that shows SFR designers how to apply the SDC to actual designs of an SFR. The SDG provides recommendations on specific items including functions and mechanisms of the SFR systems to prevent and/or mitigate accidents and how and when they should operate. The "SDG on Safety Approach" summarizes design measures against issues of the core reactivity and the decay heat removal along with design options and conditions of Gen-IV SFRs based on 83 criteria of the SDC. The process for the development of the SDG is illustrated in FIG. 1. The "SDG on Safety Approach" was approved by GIF in March 2016 and its review has started with the International Atomic Energy Agency (IAEA). The "SDG on Structures, Systems and Components" describes the design measures in detail, and is now under development by the GIF. It is expected that the SDG and the SDC are disseminated, updated and utilized in SFR development worldwide. Such activities have started interactively with research and design organizations of SFRs, regulatory bodies and their technical support organizations, and international organizations such as the IAEA and the OECD/NEA [6, 7].

3. Evaluation Method for Natural Circulation Decay Heat Removal

A set of evaluation tools, i.e., a one-dimensional safety analysis method and a statistical evaluation method, has been developed to ensure the safety with regard to natural-circulation shutdown heat removal.

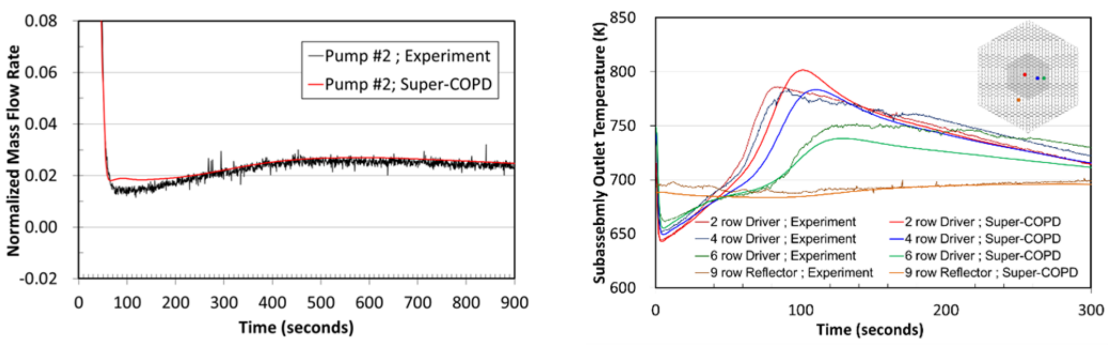
3.1. One-dimensional Safety Analysis Method

The one-dimensional safety analysis method can evaluate the hot spot temperature in the core taking into account the temperature flattening effect in the radial direction of the reactor core. Flow re-distribution caused by recirculation flow in the core and the inter-subassembly radial heat transfer across the core, and the sodium flow in the gap between wrapper tubes are major phenomena for the temperature flattening effect. A plant dynamics analysis code Super-COPD [9] has been developed for the safety analysis. Whole core thermal-hydraulics model in the Super-COPD code is shown in FIG. 2. The model includes all fuel subassemblies, radial blanket subassemblies, neutron-shield, and control rod independently [10]. The inter-wrapper gap flow is also modeled as a flow network as shown in FIG. 2 (b). The core model has been validated using the experimental data obtained at sodium test facility PLANDTL [11]. In addition, real plant data of JOYO, EBR-II, and other SFRs are also utilized to investigate the scale effect. As an example, the comparison of the simulation results with the measured data related to the EBR-II SHRT-17 test is shown in FIG. 3. Transient courses of the simulated mass flow rate and the outlet temperature of fuel assemblies are in good agreement with the experimental data [12].



(a) Radial heat transport (b) Inter-wrapper gap flow

FIG. 2. Whole core thermal-hydraulic model



(a) Mass flow rate in pump #2 (b) Outlet temperature of fuel subassemblies

FIG. 3. Comparison of the simulation results with the measured data of EBR-II SHRT-17 test [11]

3.2. Statistical Evaluation Method for Core Hot Spot Temperature

A statistical method has also been developed to evaluate the hot spot temperature in the core [13, 14]. The NC and the hot spot temperature are influenced by various factors such as geometry, fluid property, and thermal hydraulic characteristics of reactor components as well as simulation models and thermal hydraulic correlations for the predictions. Further, some of these factors influence each other via the buoyancy force. Thus, a statistical method considering such variations of influential factors is an effective tool to evaluate the hot spot temperature with a reliability coefficient. Figure 4 shows an example of a cumulative density function (CDF) of the maximum fuel cladding temperature at the secondary peak in case of the loss of offsite power in JSFR. In this example, uncertainties for 62 inputs (e.g., decay heat, gap conductance between fuel pellets and claddings, and pressure loss coefficient of fuel pin bundle) selected based on phenomena identification and ranking table (PIRT) evaluation were considered. These input uncertainties can be categorized into "known uncertainty group" which is described as probability density functions and "unknown uncertainty group" which is described only as ranges of variables. Monte Carlo samplings for the known inputs and for the unknown inputs were independently conducted to make 100 known input sets and 10 unknown input sets. A CDF was estimated with the results of plant dynamics analyses using the 100 known input sets and one of the 10 unknown input sets. In total 1,000 cases of plant dynamics analyses were performed and 10 CDFs were obtained. The core hot spot temperature can be evaluated as the temperature at 95% cumulative probability of the blue curve which was obtained by shifting the rightmost CDF curve to the right direction by modeling errors δ.

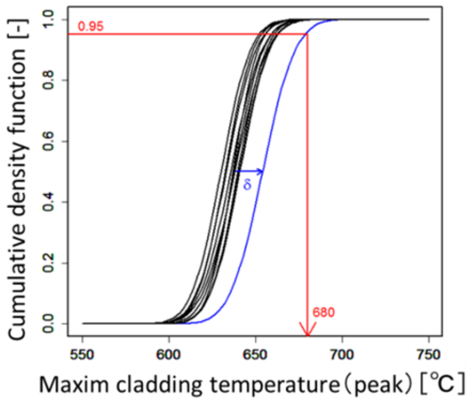
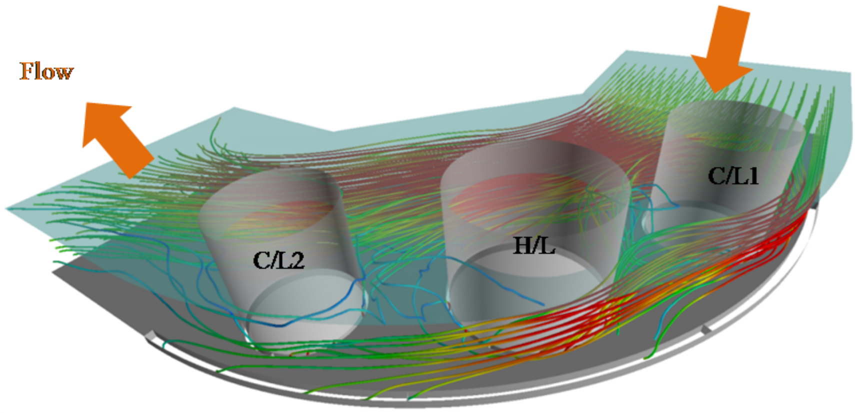


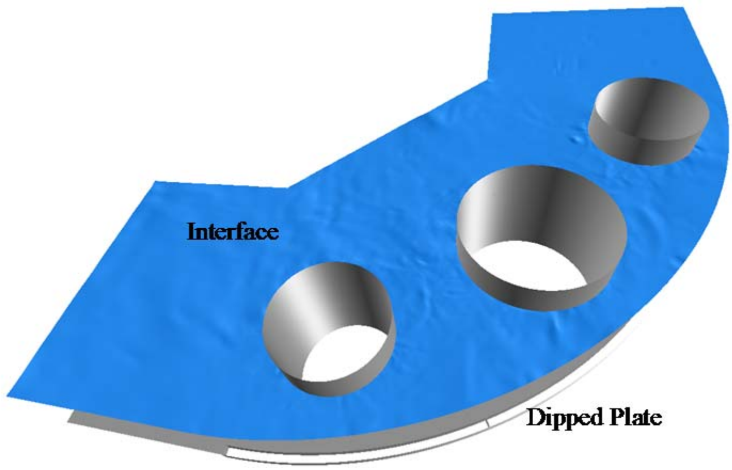
FIG.4. Statistical evaluation of core hot spot temperature during NC operation [13]

4. Study on Gas Entrainment

Thermal-hydraulics issues in an SFR reactor vessel should be addressed carefully. In particular, the gas entrainment (GE) on the free surface is one of the most important SFR-specific issues. The entrained bubble may cause reactor power disturbance in the core and degradation of heat transfer in the intermediate heat exchanger, and therefore, the design studies have been conducted to stabilize the free surface fluctuation which may induce the GE. In the latest design of JSFR, the only possibility of the GE remained is the one caused by a free surface vortex. JAEA and related organizations have studied this vortex-induced GE and proposed two kinds of prediction methods. One is the practical evaluation method composed of a vortex model (improved Burgers vortex model) with transient computational fluid dynamics (CFD) [15, 16]. The other is the direct evaluation of the GE with a high-precision numerical simulation method based on an interface-tracking approach [17].



(a)



(b)

FIG. 5. Simulation result of gas entrainment in large-scale test: (a) stream line around H/L and C/Ls, (b) interfacial shape.

4.1. Practical GE Evaluation Method

The past experiments show that the vortex-induced GE can be grouped into two types: the elongated gas core (interfacial dent) type and bubble pinch-off type. The elongated gas core type is caused by a highly-intensified vortex which makes the gas core length longer than a liquid depth. On the other hand, the experimental results show that the bubble pinch-off can be caused by strong downward velocity gradient near the tip of a gas core. The practical GE evaluation method is developed to evaluate these two types with relatively low computational cost.

Several simple experiments and a large-scale water test (1/1.8 scale partial model of the upper plenum in a reactor vessel of SFR) were conducted to investigate the onset mechanism of the GE and to obtain the validation data of the evaluation methods [18, 19]. The practical evaluation method was applied to the evaluation of the GE in the simple experiments and also in the 1/1.8 scale partial model test. As a result, the calculated values of the gas core length agreed well with the measured values, and the GE observed in the 1/1.8 scale partial model test was evaluated properly with the practical evaluation method.

4.2. High-Precision Numerical Simulation Method

In the development of the high-precision numerical simulation method, several numerical algorithms were newly developed to achieve the accurate simulation of interfacial flows. For example, an unstructured mesh scheme was chosen to accurately model the complicated structural components at the free surface region in the reactor vessel, and the high-precision calculation schemes, e.g. the accurate calculation method for physical variable gradient, were developed for the unstructured mesh. The interfacial deformation was simulated accurately with the high-precision volume-of-fluid algorithm which was also newly developed for the unstructured mesh. In addition, the formulations of momentum and pressure calculations were re-considered and improved to be physically appropriate for the gas-liquid interface.

The high-precision numerical simulation method was applied to the evaluation of the GE behaviors in the simple experiments. As a result, the GE observed in the experiment was reproduced successfully in the numerical simulation. In particular, the simulated shape of the gas core is very similar to the experimental result. The 1/1.8 scale partial model test was also simulated and the result provided the flow pattern and interfacial shape, including the bubble entrainment as shown in FIG. 5, which are consistent with the test data [20]. Therefore, the high-precision numerical simulation method is considered to be applicable to the accurate evaluation of the GE behaviour in an SFR.

5. Activities in Sodium Fire (Combustion) Evaluation

With regard to a numerical tool for sodium combustion phenomena, tools for better understanding of the complex phenomena related to sodium combustion have been developed by employing mechanistic approaches rather than parametric approaches as in FIG. 6 [21]. Currently, one of key issues in computational thermal hydraulics is code credibility. In order to ensure the code credibility especially from the viewpoint of code verification and validation (V&V), a PIRT process has been carried out for sodium fire phenomena [22, 23].

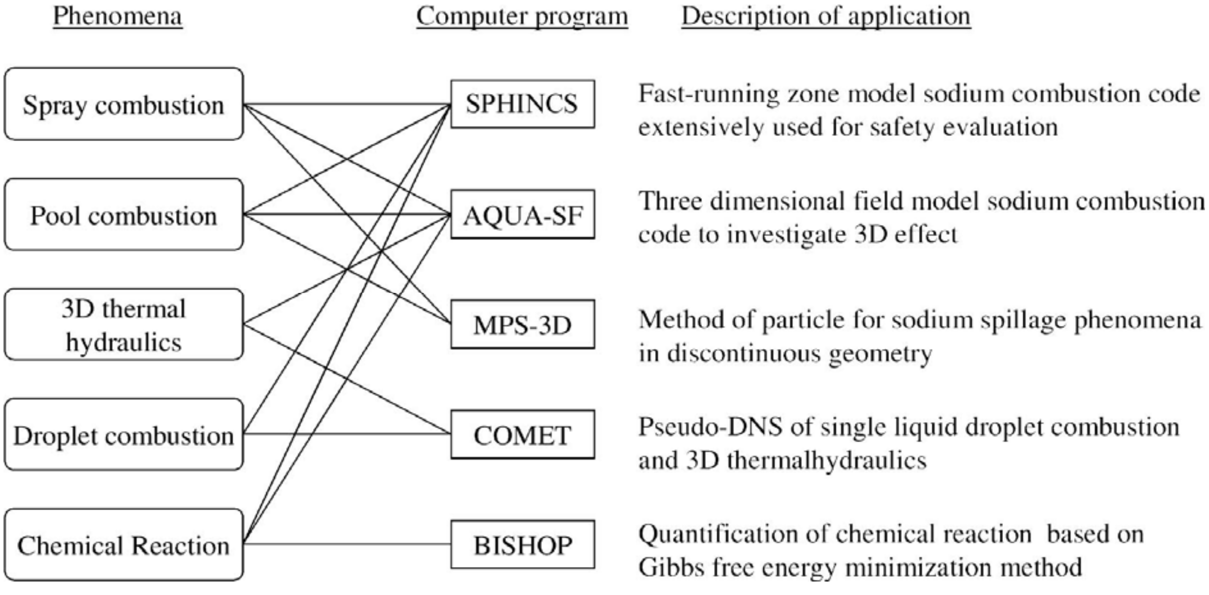


FIG. 6. Computer program lineup and its features for sodium fire [21].

Sodium fire influences the integrity of the building and internal components and the surrounding environment mainly due to chemical toxicity of reaction products. Hence multiple figures of merit (FOMs) are considered as follows: atmospheric pressure, atmospheric temperature, hydrogen concentration and steel liner for the integrity of the building structure, atmospheric pressure and temperature, component temperature, hydrogen concentration and aerosol concentration for the integrity of components and aerosol concentration for the environment. The ranking table of the important phenomena by experts' elicitation is summarized in TABLE I. The importance of short time period where a spray combustion is dominant is ranked higher in TABLE I, whereas the importance of long time period in which a pool combustion becomes dominant due to an enlargement of pool area by piling up of unburnt leaked sodium.

TABLE I: RANKING TABLE IN SODIUM FIRE PHENOMENA.

Category	Phenomenon \ Related Concern* / Figure of Merit	Atmospheric Pressure (1&2)	Concrete Temperature (1)	Component Temperature (2)	Atmospheric Temperature (2)	Steel Liner Temperature (1)	Hydrogen Concentration (1&2)	Aerosol Concentration (2&3)
Spray Combustion	1) Droplet Generation	H/L	M/L	M/L	H/L	M/L	L/L	H/M
	2) Spray Combustion	H/L	M/L	M/L	H/L	M/L	L/L	H/M
	3) Reaction Heat Transfer (spray)	H/L	M/L	M/L	H/L	M/L	L/L	L/M
Pool Combustion	4) Pool Enlargement	L/M	L/M	L/M	L/M	L/M	L/M	L/M
	5) Pool Combustion	L/M	L/H	L/H	L/M	L/H	L/M	L/M
	6) Reaction Heat Transfer (pool)	L/M	L/H	L/H	L/M	L/H	L/L	L/L
Heat Transfer	7) Heat Conduction	L/L	H/H	H/H	L/L	H/H	L/M	L/L
	8) Heat Convection	H/M	M/M	L/M	M/H	L/M	L/M	L/M
	9) Heat Radiation	M/M	M/M	L/M	M/M	L/M	L/M	L/L
Mass Transfer	10) Mass and Momentum Transfer	M/L	L/L	L/L	L/M	L/L	L/M	M/H
	11) Gas Species Transfer	L/L	L/L	L/L	L/L	L/L	H/H	M/M
	12) Aerosol Transfer	L/L	L/L	L/M	L/L	L/M	L/M	H/H
Chemical Reaction	13) Atmospheric Chemical Reaction	L/L	L/L	L/L	L/L	L/L	L/M	L/M
	14) Steel Liner Corrosion Wastage	L/L	L/L	L/L	L/L	H/H	L/L	L/L

*Concern about 1)Building Structure, 2)Components and 3)Circumference Enviroment

Since sodium fire may occur under a severe accident condition, the computational model of a sodium fire was also implemented into a severe accident code such as CONTAIN-LMR code [24]. With accordance to a recent progress of computer performance, the developed sodium fire models or tools were also applied to the severe accident analysis.

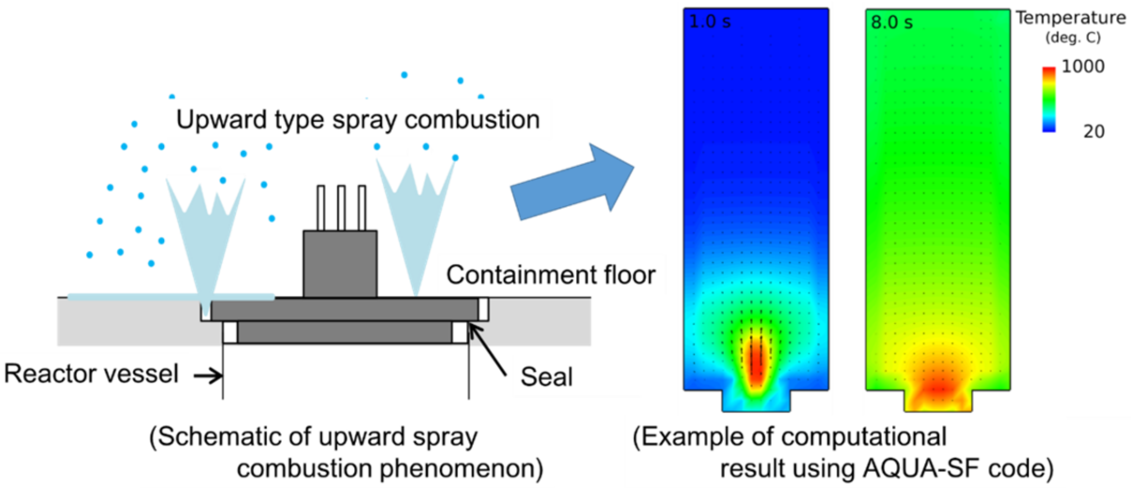


FIG. 7. Computational result of upward type spray combustion.

As an example, FIG. 7 shows a temperature distribution of upward type sodium spray combustion obtained by using multi-dimensional sodium fire analysis code AQUA-SF. It also shows the upward type sodium spray combustion which may occur when liquid sodium leaks from a reactor vessel to a floor of the containment by leaking from the seal.

Furthermore, sodium fire modeling and information exchange of related experimental results are now being conducted between Japan Atomic Energy Agency and Sandia National

Laboratory as an international collaboration in the field of advanced reactor modeling and simulation in Civil Nuclear Energy Research and Development Working Group (CNWG) to improve the code credibility.

6. Evaluation Method for Sodium-water Reaction Phenomenon

When pressurized water or vapor leaks from a failed heat transfer tube in a steam generator (SG) of an SFR, a high-velocity, high-temperature, and corrosive jet with sodium-water reaction may cause wastage or consecutive failure on the adjacent tubes. Prevention of the failure propagation is one of the major concerns in designing the SG. To evaluate the possibility of failure propagation, a multiphysics evaluation system comprising the multiple computational codes (see FIG. 8) was newly constructed [25-27]. The evaluation system includes modelling of multiphase flow, heat and mass transfer, chemical reaction, stress generation, variation of material properties, and failure of materials. This section presents the systematic experiments conducted for the development of the numerical methods and the examples of numerical analysis.

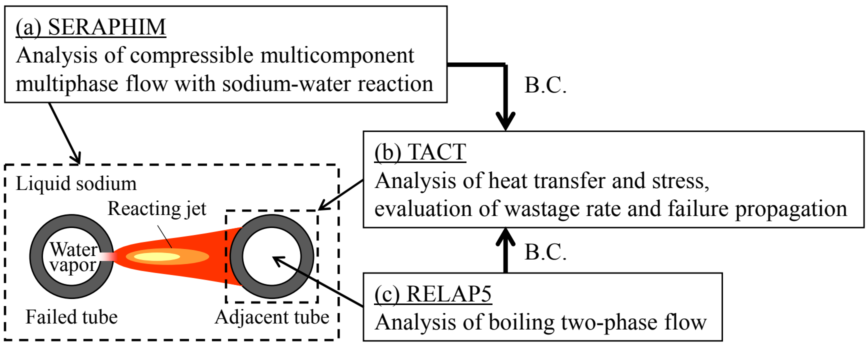


FIG. 8. Multiphysics evaluation system for sodium-water reaction phenomenon.

The systematic experiments shown in FIG. 9 were conducted for elucidation of the phenomena and validation of the numerical methods. In these experiments, we succeeded in identifying a dominant process in the sodium-water reaction, measuring the sodium-side and water-side heat transfer behaviors, measuring the rupture behaviors of the rapidly-heated tube, deriving a correlation between the wastage rate and some parameters relevant to liquid droplet impingement erosion and flow accelerated corrosion, and so forth.

As one component of the evaluation systems, a CFD code called SERAPHIM was developed to calculate compressible multicomponent multiphase flow with sodium-water reaction. The SERAPHIM code consists mainly of the multidimensional multifluid model considering compressibility, the numerical model for the reaction, and the liquid droplet entrainment and transport model. The applicability of the SERAPHIM code was investigated through the analysis of the experiment on water vapor discharging in liquid sodium under the actual condition of the SG. The computational domain consists of the cylindrical vessel, and the two simulated horizontal heat transfer tubes: one of which has a water vapor discharging nozzle at the center. The cylindrical vessel is initially filled with liquid sodium at the rated condition of the SG. Pressurized water vapor goes into the sodium pool vertically from the discharging nozzle. As shown in FIG. 10, the underexpanded jet appears and impinges on the target tube located above the discharging tube. The high temperature region is formed around the underexpanded jet by the reaction. The calculated temperature distribution agreed with the measurement result well. The location of higher impingement velocity of the liquid droplet is close to the wastage region confirmed in the experiment. It was demonstrated that the proposed numerical methods could evaluate the wastage environment.

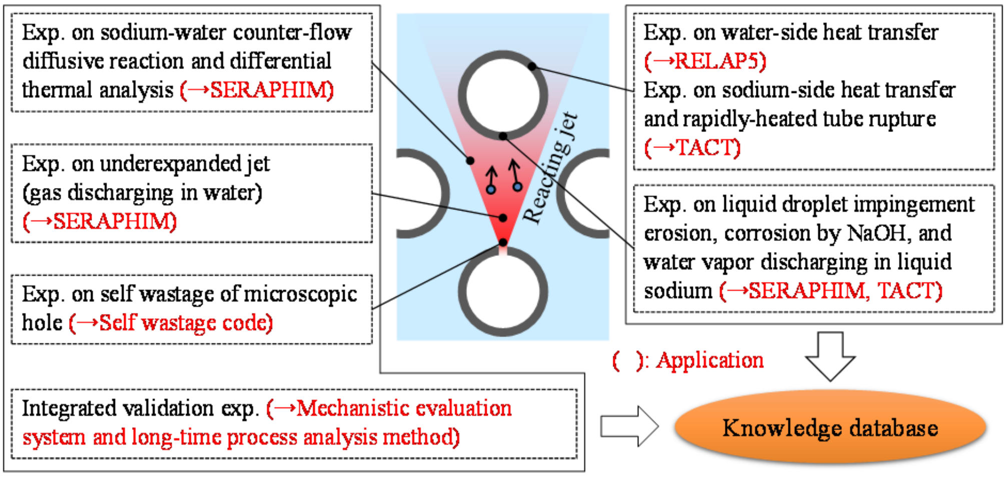


FIG. 9. Systematic experiments for development of numerical methods.

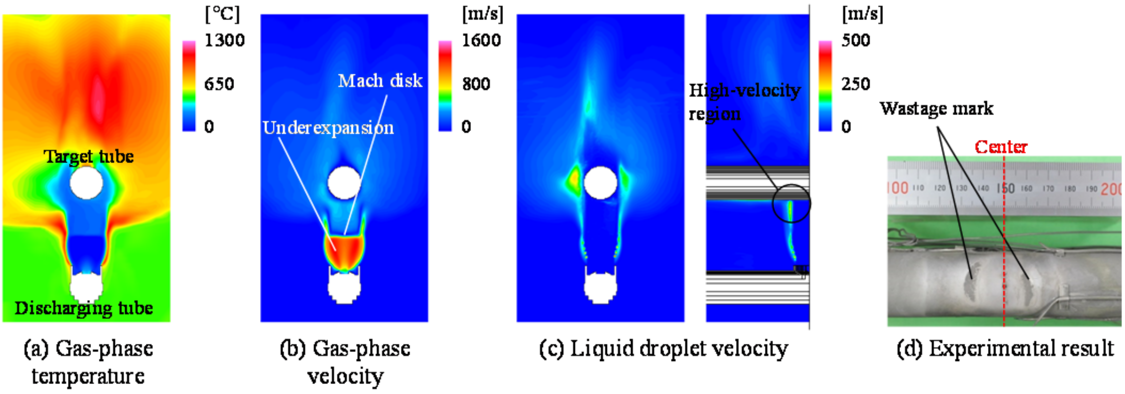


FIG. 10. Numerical results of water vapor discharging in liquid sodium.

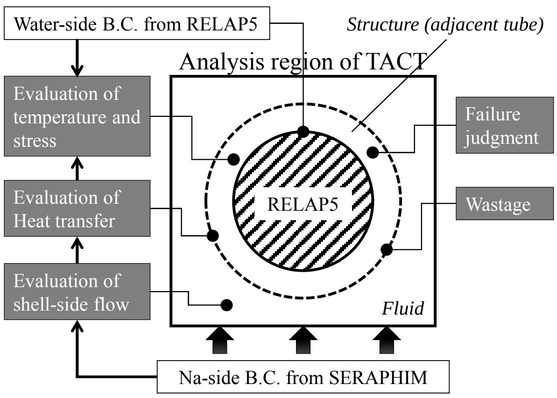


FIG. 11. Overview of TACT code.

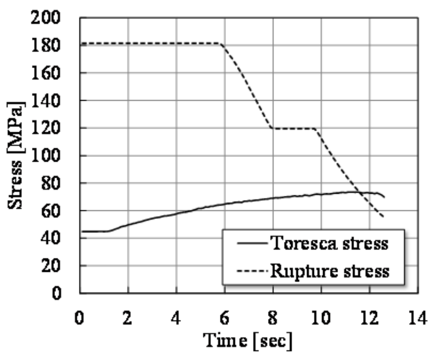


FIG. 12. Numerical results of over-heating tube rupture simulation test.

As shown in FIG. 11, the TACT code predicts shell-side flow around an adjacent tube, heat transfer from the reacting jet to the tube, wastage of a tube wall, and the occurrence of tube failure. The fluid-structure thermal coupling model, the stress evaluation model, the wastage model, and the failure judgment model are integrated into the TACT code. The TACT code has two failure judgment models: the ductile rupture model and the creep rupture model. The tube rupture experiment [28] was analyzed to confirm the capability of the failure judgment models. In this experiment, the internally-pressurized tube was heated rapidly by the high frequency induction heating until tube rupture occurs. As shown in FIG. 12, the calculated Tresca stress exceeds the rupture stress at 11.6 seconds, which indicates the occurrence of the ductile rupture. The prediction results are very close to the experimental results (12.3

seconds). The good agreement between the analysis and the experiment was also obtained in the case of the creep rupture.

Thermal hydraulic behavior of water inside the tube is evaluated with the RELAP5 code. During the tube failure accident, adjacent tubes are heated rapidly by the reacting jet. Kurihara et al. [29] measured the heat transfer characteristics inside the tube which was heated rapidly by high-frequency induction heating technique. Based on the experimental results, the original heat transfer correlations in the RELAP5 code were successfully modified for such a rapid heating situation.

7. Summary and Concluding Remarks

The authors focused on four important thermal hydraulic phenomena: NC decay heat removal, GE, sodium combustion, and sodium-water reaction. Progress of evaluation methods on these are shown as key issues for safety enhancement complying with SDC and SDG that have been established in the framework of the GIF.

A one-dimensional safety analysis code has been developed to evaluate the core hot spot temperature under the NC decay heat removal conditions taking into account the temperature flattening and inter-assembly heat transfer effects in the core. A statistical safety evaluation method has also been developed based on the one-dimensional code to evaluate the hot spot temperature with a coefficient of confidence. The safety analysis method has been validated using sodium NC experimental data as well as real plant data of JOYO, EBR-II, and other SFRs.

Two types of the evaluation methods for the GE phenomenon have been newly developed. One is the practical evaluation method composed of an improved Burgers vortex model and transient CFD. The other is the direct evaluation method by a high-precision numerical simulation method based on an interface-tracking approach. From the numerical analyses of the simple experiments and the 1/1.8 scale partial model experiment, it was demonstrated that these evaluation methods can evaluate the behavior of GE.

Numerical tools for better understanding of the complex phenomena related to sodium combustion have been developed by employing mechanistic approaches rather than parametric approaches. The developed sodium fire models or tools have also been extended to more complicated situations including upward type sodium spray combustion. In addition, a PIRT process has been carried out to ensure the code credibility especially from the viewpoint of code V&V.

To evaluate the possibility of tube failure propagation in the SG, a multi-physics evaluation system comprising the multiple computational codes was newly constructed. The applicability of the SERAPHIM code calculating compressible multiphase flow with sodium-water reaction was demonstrated through the analysis of the experiment under the actual condition of the SG. Development of the TACT code enabled prediction of heat transfer from the reacting jet to the tube and the possibility of tube failure. The original heat transfer correlations in the RELAP5 code were successfully modified for a rapidly heated tube.

The evaluation methods for all important issues related to the feasibility and the safety are planned to be eventually integrated into a comprehensive numerical simulation system that can be applied to all phenomena postulated in SFR systems. In parallel, experimental studies are conducted for creating a database to offer qualified data for modeling and its validation. These activities can also provide opportunities or tools effective for the development of human resources and the management of knowledge and/or technologies.

References

[1] ICHIMIYA, M., et al., "A Next Generation Sodium-Cooled Fast Reactor Concept and Its R&D Program", Nucl. Eng. Tech., 39, 171-186, (2007).

[2] USDOE & GIF "A Technology Roadmap for Generation-IV Nuclear Energy Systems", GIF-002-00 (2002).

[3] GIF Risk & Safety Working Group, "Basis for safety approach for design & assessment of Generation-IV Nuclear Systems", GIF/RSWG/2007/002 (2008).

[4] IAEA, "Safety of Nuclear Power Plants: Design", SSR-2/1 (2012).

[5] IAEA, The Fukushima Daiichi Accident, ISBN:978-92-0-107015-9, (2015)

[6] NAKAI, R., "Status of the international review on SFR SDC Phase I report and SDG development in Phase II", 5th Joint IAEA-GIF Workshop on Safety of SFR, Vienna, 23-24 June (2015).

[7] NAKAI, R., SOFU. T., "Safety Design Criteria for Generation-VI Sodium-Cooled Fast Reactor System", GIF Symposium, San Diego, USA, 14-15 Nov. (2012).

[8] JAEA R&D Review 2016-17 (to be published).

[9] YAMADA, F., et al., "Development of natural circulation analytical model in Super-COPD code and evaluation of core cooling capability in Monju during a station blackout", Nucl. Technol. **188** (2014) , pp.292-321.

[10] OYAMA, K., et al., "Development of natural circulation analysis methods for a sodium cooled fast reactor", Nucl. Sci. Technol. **53** (2015), pp.353–370.

[11] ONO, A., et al., "An experimental study on natural circulation decay heat removal system for a loop type fast reactor", Nucl. Sci. Technol. **53** (9) (2016) , pp.1385–1396.

[12] DODA, N., et al., "Benchmark analysis of EBR-II shutdown heat removal test-17 using of plant dynamics analysis code and subchannel analysis code", Proc. of International Conference of Advances in Nuclear Power Plants (ICAPP) (2016), 16295 (CD-ROM).

[13] DODA, N., et al., "Development of core hot spot evaluation method of a loop type fast reactor equipped with natural circulation decay heat removal system", Proc. of the 10th Korea-Japan Symposium on Nuclear Thermal Hydraulics and Safety (NTHAS10) (2016), N10P1050.

[14] WATANABE, O., et al., "Development of an evaluation methodology for the natural circulation decay heat removal system in a sodium cooled fast reactor", Nucl. Sci. Technol. **52** (2015) , pp.1102–1121.

[15] SAKAI, T., et al., "Proposal of Design Criteria for Gas Entrainment From Vortex Dimples Based on a Computational Fluid Dynamics Method", Heat Transfer Engineering, Vol. 29, No. 8 (2008) 731-739.

[16] ITO, K., et al., "Improvement of Gas Entrainment Evaluation Method –Introduction of Surface Tension Effect–", Journal of Nuclear Science and Technology, Vol. 47, No. 9 (2010) 771-778.

[17] ITO, K., et al., "A volume-conservative PLIC algorithm on three-dimensional fully unstructured meshes", Computers & Fluids, Vol. 88 (2013) 250-261.

[18] KIMURA, N., et al., “Experimental Study on Gas Entrainment at Free Surface in Reactor Vessel of a Compact Sodium-Cooled Fast Reactor”, Journal of Nuclear Science and Technology, Vol. 45, No. 10 (2008) 1053-1062.

[19] EZURE, T., et al., “Transient Behavior of Gas Entrainment Caused by Surface Vortex”, Heat Transfer Engineering, Vol. 29, No. 8 (2008) 659-666.

[20] ITO, K., et al., “Two-Phase Flow Simulation of Gas Entrainment Phenomena in Large-Scale Experimental Model of Sodium-Cooled Fast Reactor”, Progress in Nuclear Science and Technology, Vol. 2 (2011) 114-119.

[21] YAMAGUCHI, A., et al., “Numerical Methodology to Evaluate Fast Reactor Sodium Combusiton”, Nucl. Technol. **136** (2001), pp.315-330.

[22] OHNO, S., et al., “Development of PIRT and Assessment Matrix for verification and validation of sodium fire analysis codes”, J. Power and Energy Systems, **6**(2) (2012), pp. 241-254.

[23] AOYAGI, M., et al., “Identification of Important Phenomena under Sodium Fire Accidents based on PIRT Process with Factor Analysis in Sodium-Cooled Fast Reactor”, Proc of the tenth Korea-Japan Symposium on Nuclear Thermal Hydraulics and Safety (NTHAS10) (2016), N10P1039.

[24] MIYAHARA, S., et al., “Development of Fast Reactor Containment Safety Analysis Code CONTAIN-LMR (1) Outline of development project”, Proc. of 23th International Conference on Nuclear Engineering (ICONE23), (2015), ICONE23-1586.

[25] TAKATA, T., et al., “Computational methodology of sodium-water reaction phenomenon in steam generator of sodium-cooled fast reactor”, J. Nucl. Sci. Technol. **46** (2009) 613-623.

[26] UCHIBORI, A., et al., “Development of numerical evaluation methods for multi-physics phenomena under tube failure accident in steam generator of sodium-cooled fast reactor”, Transactions of the JSME, Series B, **79** (2013) 99-103 (in Japanese).

[27] UCHIBORI, A., OHSHIMA, H., “Applicability of a mechanistic numerical method for sodium-water reaction phenomena in steam generators of sodium-cooled fast reactors”, Mechanical Engineering Journal, **3** (2016) 1-9.

[28] HAYASHIDA, Y., HAMADA, H., “Evaluation of tube rupture simulation test (TRUST-1) for FBR steam generators”, PNC TN9410 97-002 (1996) (in Japanese).

[29] KURIHARA, A., et al., “Characteristics of heat transfer inside a tube during sodium-water reaction in a FBR steam generator”, Heat Transfer-Asian Research, **39** (2010) 628-633.