The Status of Safety Research in the Field of Sodium-cooled Fast Reactors in Japan

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Abstract. This paper describes the status of safety research activity in the field of sodium-cooled fast reactors (SFRs) in Japan, mainly on severe accident related issues. Core damage sequences are analyzed by applying probabilistic risk assessment methodology and categorized into typical accident phases, i.e., initiating phase, transitions phase, and material relocation and cooling phase. In order to utilize superior characteristics of sodium as coolant, achievement of in-vessel retention is one of important objective of safety design and evaluation for SFRs. Focus is on the later phases of accidents for which experimental data acquisition and code development are going on. A series of out-of-pile and in-pile experiments in EAGLE-3 and related tests in the MELT facility are being conducted for molten fuel discharge and cooling. Study on debris bed formation and self-leveling effect is also conducted. A fast-reactor safety analysis code, SIMMER, is developed to enhance its capability to be applicable such phenomena in the later phase of accidents.

Key Words: Generation IV SFR, severe accident, In Vessel Retention, Core Disruptive Accident

1. Introduction

Based on the lesson learned from the Fukushima Dai-ichi nuclear power plants accident, enhancement of measures against severe accident is focused in the safety design of nuclear power plants. Such measures shall be built-in rather added-on for the next generation reactors including Sodium-cooled Fast Reactors (SFRs). The reactor core of fast reactors including SFRs is not in the most critical configuration under normal operation condition. This means that the neutron spectrum hardening may occur and may cause positive reactivity effects when removing coolant and/or structure materials from the core or when compressing fissile materials in the core. Because of such core characteristics of fast reactors, reactivity insertion accidents assuming hypothetical core compaction and coolant removal have been focused on and evaluated as a major safety issue from the beginning of its development. Related safety research has been carried out among SFR developing countries. Hypothetical Core Disruptive Accidents (CDAs) were evaluated for the safety licensing of the experimental reactor JOYO and the prototype reactor MONJU in Japan. Up to now a plenty of experimental knowledge has been acquired about many important phenomena, which might appear in the core degradation process, and analysis codes have also been developed based on such experimental knowledge. In addition, Probabilistic Safety Assessment (PSA) of level 1 and level 2 for MONJU has been carried out by using these analysis codes and knowledge, the core damage progression and its consequences have been analyzing systematically [1][2]. Regarding demonstration reactors and commercial reactors in Japan, a design concept of an advanced loop-type reactor (JSFR) was created as the result of the Feasibility Study on Commercialized Fast Reactor Cycle Systems (FS) and related research and developments have been done in the Fast Reactor Cycle Technology Development (FaCT) project. Design measures for prevention and mitigation of severe accidents have been incorporated in the safety design of JSFR and further effort has been made for enhancing decay heat removal and containment capability to cope with severe plant conditions caused by severe external events [3][4]. In parallel, safety research in the field of severe accidents for SFRs has been carrying out to show effectiveness of the design measures.

This paper describes major outcomes obtained up to now and recent activity in the field of severe accident studies for SFRs in Japan. Here main objective of the studies is to provide basis for establishing design measures aiming at In-Vessel Retention (IVR) as stated in the next chapter.

2. Direction of safety design for future SFRs

Since a boiling point of sodium is about 900 degree-C at atmospheric pressure, SFR coolant systems should be designed with a sufficient margin up to coolant boiling point without pressurization. When a coolant leakage from the reactor coolant boundary is postulated, liquid sodium level in the reactor vessel, which is necessary to circulate sodium for decay heat removal, can be assured by retention of leaked coolant in back-up structure such as guard vessel. On the other hand, if degraded core materials are released from the reactor vessel into the containment structure, it is difficult to provide reasonable measures to retain them due to high temperature and chemical activity of sodium. Therefore, it is important to establish design measures for achieving IVR of the degraded core materials. For IVR achievement, not only retaining reactor coolant boundary function but also maintaining cooling paths to an ultimate heat sink is needed to be ensured. It is necessary that these functions are not lost against mechanical energy release due to re-criticality which could occur by temperature and material distribution changes of the core in the core degradation process. Furthermore, as the consequence of material relocation in the core degradation process, the material distribution in the reactor vessel has to be finally settled in stable sub-critical and coolable condition.

Study shall be made on various phenomena, which would appear during the core degradation from the beginning to the end, namely in initiating phase, transition phase, material relocation phase and final cooling phase, and also for transportation process of radioactive materials. The safety research for SFR with oxide fuel has been carrying out with such scope. In the conventional design including MONJU, CDA has been categorized as “beyond design basis”. Beyond design basis accident was categorized as evaluation for margin confirmation so far, however Design Extension Condition (DEC) has been considered for a design in recent years, then safety studies are required to confirm effectiveness of design measures in order to mitigate the CDA consequences, in particular, the measure to prevent severe re-criticality by molten fuel discharge from the core prior to large scale molten pool formation, the measure to promote fragmentation and solidification of the discharged molten fuel in a sodium pool, and the measure to ensure stable cooling of core debris. Figure 1 shows the event sequence and design measures to achieve IVR, and related phenomena. In the design and evaluation relating to the severe accident, it is important to take reasonable approach without excessive conservatism by evaluating physical event progression more precisely and quantitatively based on outcomes of the safety research. The safety research for SFR with oxide fuel has been progressed worldwide in the history of their developments. The focus of research has been moved into the latter phases of accident, i.e., material relocation and cooling phases in recent years.

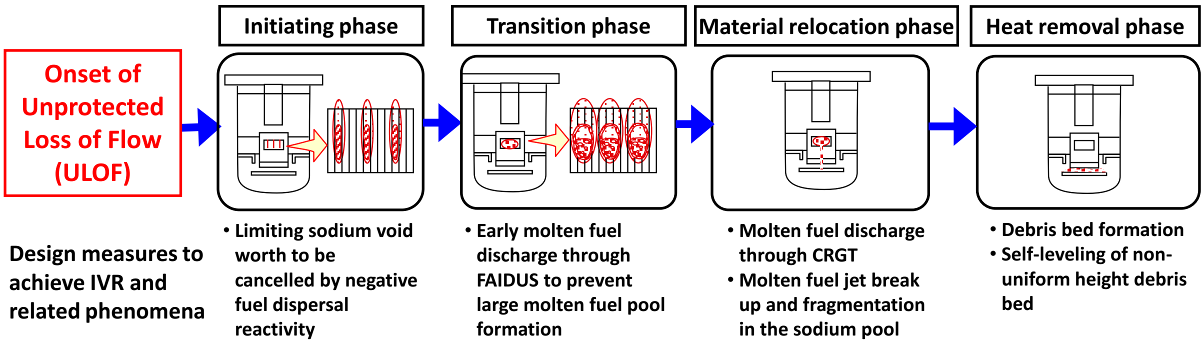

FIG. 1. ULOF sequence and design measures to achieve IVR

When one considers comprehensiveness and balance of design measures against severe accident, PSA is expected to provide good insights for identifying events or conditions to be considered for the design measures. Research and development have been carried out to develop evaluation methodologies and database of level 1 PSA and level 2 PSA for MONJU and JSFR in Japan [5][6]. Furthermore, enhancement of measures against external events is required based on the lesson learned from the Fukushima Dai-ichi nuclear power plants accident, and evaluation methodology for risks of external events on the reactor facilities which is under development.

3. Outcome of severe accident research

In this chapter, outcome of previous researches are summarized [7], and some recent researches are also introduced relating to evaluation of the core damage progression for IVR achievement.

Recently, research about the material relocation phase and the cooling phase is carried out, in which experimental study and development of analysis technology are carried out relating to molten fuel discharge behavior through a steel duct filled with sodium, break up and fragmentation of columnar molten fuel injected into sodium pool through a nozzle, and self-leveling behavior of a debris bed caused by internal sodium boiling in the debris bed [8].

3.1. Initiating phase

Fuel pin failure and subsequent material relocation during under cooling and/or overpower transients are points of interest. Reactor power during the transients is determined by the reactivity feedback obtained by the temperature change and material motion. In-pile tests, CABRI, TREAT, observing fuel pin failure and relocation behavior under accident conditions have been carried out [9][10][11]. Safety analysis codes such as SAS4A have been developed based on the obtained data. Analysis of typical core concepts is possible with improved SAS4A based on a database of CABRI tests which were carried out in Japan-European collaborations. A lot of data with various fuel pin specifications and transient conditions were acquired in CABRI tests. SAS4A was developed in US and has been improved enabling to treat an oxide fuel under Japan-European collaborations. Although fuel failure behavior depends on the transient condition including power and the fuel pin specifications, failed fuel tends to disperse in general. This brings strong negative reactivity feedback during the core degradation process. Although total sodium void worth over the whole core is generally positive for a middle scale and a large scale SFR, energetics in the initiating phase can be prevented by limiting the sodium void worth to be cancelled by the negative reactivity feedback due to mainly Doppler, fuel axial expansion and fuel dispersal effects.

3.2. Transition phase and material relocation phase

Although the core temporally reaches sub-critical condition after the initiating phase, core melting progresses because the core is not in a coolable condition, i.e., coolant dry-out and blockage formation in coolant channels by dispersed fuel in many fuel assemblies. Possibility of re-criticality occurrence by reactivity feedback associated with material relocation is focused. A lot of knowledge has been accumulated by experiments focused on fuel melt progression and relocation inside a fuel assembly and wrapper tube failure (SCARABEE, SIMBATH, MOL7C), ingress and solidification of molten fuel into gap space around the core peripheral region (TRAN-B, BLOCKER, THEFIS, GEYSER), molten fuel pool behavior (SCRABEE/BF2), fuel discharge through steel duct (CAMEL, EAGLE), pressure events due to molten fuel discharge into the sodium pool (FRAG, FARO/THEMOS, MFTF, THINA, MELT) or pressure events due to sodium re-entry into molten fuel pool (CORECT, MELT) have been carried out. SIMMER, a fast-reactor safety analysis code coupling multi-dimensions, multi-components and multi-phases thermal hydraulics model, fuel pin model and solid structure model with a space-and energy-dependent neutron kinetics model, has been developed to analyze transient reactivity and power change due to material and temperature distribution change in the core. In addition to physical model validation based on the above experimental data, three dimensional whole core calculations can be carried out, and more detailed analysis becomes possible.

An important phenomenon, which prevents re-criticality in the transition phase and leads to a stable cooling state in the material relocation phase, is molten fuel discharge behavior from the core. In particular, a fuel discharge phenomenon from the steel duct structure filled with sodium is focused on. In a conventional core design, control rod guide tubes (CRGTs) will be available for such fuel discharge routes. JSFR adopts FAIDUS; Fuel Assembly with Inner Duct Structure for enhancement of early fuel discharge as a built-in safety measure against severe accidents. A series of experimental studies, EAGLE-1 and EAGLE-2, have been carried out to understand molten fuel discharge behavior from the steel duct and to show effectiveness of FAIDUS [12][13]. Important experimental knowledges have been obtained about steel duct wall failure behavior due to contact of high temperature molten material, discharge behavior inside the duct with Fuel Coolant thermal Interactions (FCIs) and FCI behavior in the lower sodium pool.

(1) Wall failure behavior of steel duct

Analytical results on out-of-pile experiments and in-pile experiments of EAGLE-1 have shown that an outer surface of the steel duct wall is imposed high heat flux over $10MW/m^2$ at contact with high temperature molten core materials. The high heat flux in the in-pile experiments using mixture of UO_2 and molten steel was higher than that in the out-of-pile experiments using Al_2O_3 for simulant of molten core materials. The steel duct fails in quite a short time due to the high heat flux which exceeds a cooling effect by inner sodium. As a result, the molten core materials enter the steel duct without losing its mobility due to large temperature drop. This high heat flux could be brought by direct contact of molten steel with the duct wall. If molten fuel contacts the steel duct wall, fuel crust is formed on the surface of the duct, and heat transfer from the molten fuel is reduced significantly. If mixture of molten fuel and molten steel contacts the steel duct wall, the molten steel contacts the duct wall directly. Melting of the steel duct wall escalates at that point, and then the duct is failed. Test results of EAGLE-1 experiments can be reproduced well by adopting such a model for SIMMER-III [14].

(2) Discharge behavior inside the duct

Although sodium vaporizes at contacting with the molten materials entered into the steel duct just after steel duct wall failure, remarkable pressure events were not observed at this time in EAGLE-1. The reason is thought to be that effective contact area between molten materials and liquid sodium becomes smaller since liquid sodium is pushed away by expanding sodium vapor, which is generated just after molten materials ingress. This fact suggests that heat transfer from the molten materials to liquid sodium inside the duct would not increase in the steel duct, thus the molten materials discharged into the lower sodium pool without forming blockage due to freezing [15].

In SIMMER-III, one liquid component which has the largest volume fraction in a cell forms the continuous phase and the others forms dispersed droplets and its diameter is decided by relative velocity between fluids. SIMMER-III application to the EAGLE-1 out-of-pile test showed that this model gave too much heat transfer from molten material to coolant, thus, over-estimation of the pressure generation at the moment of molten materials ingress into the duct. In order to adequately simulate the results of EAGLE-1 out-of-pile tests, it was effective to set droplet diameter of the molten materials larger when they go into the duct. Another SIMMER-III analysis of FCI tests showed that it tended to give too much contact interface area between molten materials and coolant at the moment of molten material discharge from the duct outlet into the sodium plenum and coolant vaporization at the duct outlet was also over-estimated. The test results could be reasonably simulated by keeping a columnar shape of the molten materials before reaching its break-up length based on experimental results. As mentioned above, in order to correctly evaluate molten materials discharge behavior through the steel duct into the coolant plenum, refinement of physical model concerning the contact interface area between molten materials and coolant is essential. There is one other point on the blockage formation due to freezing of molten materials inside the duct. The momentum exchange model between liquid-solid mixture fluid and structure wall in SIMMER-III does not count dependence on the flow regime of solid-liquid two phase flow. This makes the blockage formation more likely in the calculation than in the experiments. By taking the flow regime dependence effect into account, the calculation results showed good agreement with the results of EAGLE-1 out-of-pile experiments [8] [16].

Depending on the core, fuel and reactor structure design, the molten fuel discharge paths can have various geometries, e.g., diameter, length, thickness of wall, direction of discharge and destination. Therefore, influence of duct geometries on the discharge behavior of molten fuel are necessary to be understood. In FAIDUS, molten fuel will be discharged upward into the upper plenum, and its initial driving force is accumulated pressure in the core region due to fission gas released from molten fuel at first, and then, sodium vapor pressure due to FCI will lead continuous upward discharge. These behaviors were observed in the EAGLE-2 tests [13]. CRGT is different from the inner duct of FAIDUS in diameter, length and thickness, and also has steel structure for receiving the control rod and/or for regulating coolant flow rate in its inside. The tests to understand effects by difference of structure specification of these steel ducts are carried out as the EAGLE-3 experiment program [17].

(3) FCI behavior in sodium pool

There are two focus points when a columnar-shaped molten fuel injected into a sodium pool. The fist point is maximum pressure produced by FCIs, or maximum work energy done by sodium vapor expansion, the other is evaluation of break-up length of molten fuel column and fragmentation behavior, which have close relation to design conditions of the core catcher. A lot of researches on mechanical energy generation potential due to FCI were carried out in the past. In the general CDA conditions of SFR, contact interface temperature between molten fuel and liquid sodium is lower than minimum film boiling temperature (liquid-liquid contact condition). Then large scale premixing of small molten fuel particles in liquid sodium pool does not occur, since stable sodium vapor film does not exist on the surface of molten fuel particles. Thus, energetic pressure events are not likely to occur. On the other hand, this liquid-liquid contact condition of molten fuel and liquid sodium enhances the immediate break-up of molten fuel column by small pressure events, resulting in shortening of the distance of molten fuel column penetration into the sodium pool [18][19][20].

Break-up length and size distribution of fragmented Al_2O_3 particles were measured using the apparatus of EAGLE out-of-pile experiments in which approximately 10 kg Al_2O_3 was discharged into the sodium pool of 1.3 m depth. The break-up length is considerably shorter than estimated value by using a model based on hydraulic instability on the column surface. It suggests influence of local pressure events due to liquid-liquid contact of molten Al_2O_3 and sodium [21].

Evaluation models for the break-up length under liquid-liquid contact conditions have been developing. Visual image observations of molten low melting point alloy injection into a water pool, and molten Aluminum injection into a sodium pool showed that local boiling of pool liquid on the surface of column may have an impact on break-up of the column. Taking these observation results into account, a model which predicts the measured results of the break-up length was developed. A new empirical correlation was made, in which the thermal interaction effect is considered by adding non-dimensional sub-cooling to an existing empirical correlation based on the balance between the column inertia and surrounding pool liquid resistance. This new correlation is applicable for different combination of materials for low melting point alloy-water, and Aluminum-sodium, however, it gave under-estimation for FARO/TERMOS tests with UO_2-sodium. Further studies are needed [22].

3.3. Post-accident heat removal phase

A number of experimental studies related to coolability of a debris bed, which is formed on the core catcher installed at the bottom of the reactor vessel, are carried out. Physical models have been developed for the debris bed cooling and validated by the experimental results. Coolable limit of a debris bed and its dependence on some physical parameters such as debris size, debris bed porosity and height can be clarified by applying such models for reactor cases at interest. Distribution of the particle diameter of component particles of the debris bed such as solid fuel debris and debris bed height shall be given to evaluate coolability of the debris bed. For this purpose, it is necessary to evaluate fuel fragmentation and solidification in the sodium pool just above the core catcher and also sedimentation on the core catcher. It is known that non-uniformly settling debris bed can relocate to flatter shape thanks to internal sodium vapor pressure (self-leveling). The experimental studies and development of analysis models related to these behaviors are carried out at Kyushu University.

It is thought that a columnar-shaped molten fuel injected into the lower plenum of the reactor vessel is dispersed and deposited in a broad area in a wide open space such as the lower plenum of the reactor vessel since it falls down on the core catcher after break-up and fragmentation process described in the previous section. Debris bed may have different heights in a local area on the core catcher. Sedimentation of debris, debris bed formation and self-leveling behavior due to internal coolant boiling were observed in experiments using simulant materials. Empirical correlations, which expresses time history of debris bed height change, were developed using dimension analysis of the experimental results obtained under various conditions using Al_2O_3, ZrO_2, Zn, stainless steel and Cu as simulant materials with various particle size and shape factors. These results are useful for development of physical models for reactor application and validation [23].

Two types of modeling approaches are introduced for development of analysis tools for the debris bed formation, namely continuum approach at a macroscopic level and discrete approach at a microscopic level. These models are incorporated into a series of the SIMMER codes. The former approach treats debris bed as one fluid component and models contact and collision between particles as particle viscosity and particle pressure. The latter one directly models interaction between individual debris particles by adopting discrete element method (DEM). Interactions between events in the thermal hydraulic field, which include coolant boiling, and movement of solid debris particles are calculated by a hybrid computational method, in which the Eulerian code, SIMMER, is combined with DEM based on a Lagrangian approach. An advantage of the former is permitting reactor application in the practical calculation time, and that of the latter is describing more detailed physical phenomena. Quantitative evaluation is permitted in combination with these advantages [24].

Break-up length of the molten fuel is significantly reduced due to thermal interaction between the molten fuel and liquid sodium. However, if sodium inventory is not enough comparing with injected molten fuel mass or a sodium pool depth is not enough, columnar-shaped molten fuel may directly reaches on structures such as the core catcher. Studies are carried out to investigate spreading behavior of molten materials on a flat plate in such a situation. Spreading behavior on a plate was recorded by a high speed camera after dropping the low melting point alloy on the plate located in a water pool. Since the molten materials spread on the plate under liquid-liquid contact condition, vapor bubbles are formed on the surface of the spreading alloy and the molten materials are solidified in a porous state. This study is in the preliminary stage at the moment, and step-up to an in-pile experiment is under planning [25].

4. Conclusions

Research and development in the field of severe accident have been carried out for establishment of design and evaluation technologies to achieve IVR toward commercialization of SFR in Japan. This would contribute for establishment of safety design criteria and safety design guideline for the next generation SFRs. Beyond design basis events were recognized as those for confirmation of safety margin in the past. Recently, DECs are taken into account in the safety design, and hence safety research is required to provide relevant information to evaluate effectiveness of design measures for mitigation of core damage consequences. The design and evaluation related to the severe accident should be in a rational manner so that excess conservatism can be excluded. This will be carried out by making evaluation of the core damage sequences more realistically based on the results of safety research. The rich database on severe accidents of oxide fueled SFRs has been accumulated worldwide in their long development history. These days, researches have been focused on tasks relating to later phase of core damage sequences, i.e., material relocation and cooling in the reactor vessel. Their results are expected to lead to a rational design of a safety design for the next generation SFRs. It is also important to continue further effort in this safety research field.

References

[1] AIZAWA, K., et al., “Application of Probabilistic Safety Analysis to an LMFBR Plant”, Forth International ENS/ANS Conf., Topic-III, Part 1, Geneva, Switzerland, June (1986)

[2] KONDO, S., et al., "Integrated Analysis of In-Vessel and Ex-Vessel Severe-Accident Sequences," 1990 Int. Fast Reactor Safety Mtg., Snowbird, Utah, U.S.A., August 12-16, (1990)

[3] KUBO, S., et al.,” Safety Design Approach for JSFR Toward the Realization of GEN IV Sodium Cooled Fast Reactor”, Proc. FR13, Paris, France (2013) IAEA-CN-199-150.

[4] CHIKAZAWA, Y., et al., “Design Study on Measures to Prevent Loss of Decay Heat Removal in a Next Generation Sodium-cooled Fast Reactor”, Proc. ICAPP 2017, Fukui & Kyoto, Japan (in press).

[5] KURISAKA, K., et al., “Development of Level-1 PSA Method for Sodium-Cooled Fast Reactor”, PSAM Topical Conference in Tokyo, 15-17 April, Tokyo, Japan, (2013).

[6] NAKAI. R., et al., “Development of Severe Accident Evaluation Technology (Level 2 PSA) for Sodium-cooled Fast Reactors (1) Overview of evaluation methodology development,” Proc. ICAPP’09, 9117, Tokyo, Japan, May 10-14, (2009)

[7] SATO, I., et al., “Safety Strategy of JSFR Eliminating Severe Recriticality Events and Establishing In-Vessel Retention in the Core Disruptive Accident”, J. Nucl. Sci. Technol., **48**(4) (2011) 556–566.

[8] TOBITA, Y., et al., “Development of the Evaluation Methodology for the Material Relocation Behavior in the Core Disruptive Accident of Sodium-Cooled Fast Reactors”, J. Nucl. Sci. Technol., **53**(5) (2016) 698–706.

[9] KUSSMAUL, G., et al., ‘‘The CABRI project—Overall status and achievements,’’ Proc. Science and Technology of Fast Reactor Safety, Guernsey, Vol. I, 103 (1986).

[10] HAESSLER, M., et al., ‘‘The CABRI-2 programme—Overview on results,’’ Proc. Int. Fast Reactor Safety Mtg., Snowbird, USA, Vol. II, 209 (1990).

[11] WRIGHT, A. E., et al., ‘‘Fast reactor safety testing in TREAT in the 1980s,’’ Proc. Int. Fast Reactor Safety Mtg., Snowbird, USA, Vol. II, 233 (1990).

[12] KONISHI, K., et al., “The EAGLE PROJECT to Eliminate the Recriticality Issue of Fast Reactors - Progress and Results of In-pile Tests -”, Proc. NTHAS5, Korea (2006) NTHAS5-F001.

[13] KAMIYAMA, K., et al., “Experimental Studies on the Upward Fuel Discharge for Elimination of Severe Recriticality during Core-Disruptive Accidents in Sodium-Cooled Fast Reactors”, J. Nucl. Sci. Technol., **51**(9) (2014) 1114–1124.

[14] TOYOOKA, J., et al., “Improvements to the SIMMER Code Model for Steel Wall Failure based on EAGLE-1 Test Results”, Proc. NTHAS10, Kyoto, Japan (2016) N10P1116.

[15] KAMIYAMA, K., et al., “Experimental Study on Fuel-Discharge Behaviour through In-Core Coolant Channels”, J. Nucl. Sci. Technol., **50**(6) (2013) 629–644.

[16] KAMIYAMA, K., et al., “Development of an Evaluation Methodology for Fuel Discharge in Core Disruptive Accidents of Sodium-Cooled Fast Reactors”, J. Energy Power Eng., **8** (2014) 785–793.

[17] MATSUBA, K. et al., “A Recent Experimental Program to Evidence In-Vessel Retention by Controlled Material Relocation during Core Disruptive Accidents of Sodium-cooled Fast Reactors”, Proc. NTHAS10, Kyoto, Japan (2016) N10P1100.

[18] KONDO, Sa. et al., “Experimental Study on Simulated Molten Jet-Coolant Interactions,” Nucl. Eng. Des., **115** (1995) 73–84.

[19] ABE, Y., et al., “Fragmentation Behavior During Molten Material and Coolant Interactions”, Nucl. Eng. Des., **236** (2006) 1668–1681.

[20] NISHIMURA, S. et al., “Fragmentation Mechanisms of a Single Molten Copper Jet Penetrating a Sodium Pool - Transition from Thermal to Hydrodynamic Fragmentation in Instantaneous Contact Interface Temperatures below its Freezing Point -,” J. Nucl. Sci. Technol., **47**(3) (2010) 219–228.

[21] MATSUBA, K., “Experimental Discussion on Fragmentation Mechanism of Molten Oxide Discharged into a Sodium Pool”, Mech. Eng. J., **3**(3) (2016) 15-00595.

[22] MATSUBA, K., et al., “An Empirical Correlation to Predict the Distance for Fragmentation of Simulated Molten-Core Materials Discharged into a Sodium Pool”, Proc. NUTHOS-11, Gyeongju, Korea (2016) N11A0089.

[23] MORITA, K., et al., “A New Empirical Model for Self-Leveling Behavior of Cylindrical Particle Beds”, J. Nucl. Sci. Technol., **53**(5) (2016) 713–725.

[24] GUO, L., et al., “Numerical Simulation of a Self-Leveling Experiment Using a Hybrid Method”, Mech. Eng. J., **1**(4) (2014) 14-00002.

[25] IMAIZUMI, Y., et al., “Fundamental Experiments of Jet Impingement and Fragmentation Simulating the Fuel Relocation in the Core Disruptive Accident of Sodium-cooled Fast Reactors”, Proc. ICAPP 2017, Fukui and Kyoto, Japan (in press).