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Citation	Journal of Nuclear Science and Technology, 53(5), p.698-706
Text Version	Author's Post-print
URL	https://jopss.jaea.go.jp/search/servlet/search?5054631
DOI	https://doi.org/10.1080/00223131.2016.1143409
Right	This is an Accepted Manuscript of an article published by Taylor & Francis in Journal of Nuclear Science and Technology on May 2016, available online: http://www.tandfonline.com/10.1080/00223131.2016.1143409.

ARTICLE

Development of the Evaluation Methodology for the Material Relocation Behavior in the Core Disruptive Accident of Sodium Cooled Fast Reactors

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Abstract

The in-vessel retention (IVR) of core disruptive accident (CDA) is of prime importance in enhancing safety characteristics of sodium-cooled fast reactors (SFRs). In the CDA of SFRs, molten core material relocates to the lower plenum of reactor vessel and may impose significant thermal load on the structures, resulting in the melt through of the reactor vessel. In order to enable the assessment of this relocation process and prove that IVR of core material is the most probable consequence of the CDA in SFRs, a research program to develop the evaluation methodology for the material relocation behavior in the CDA of SFRs has been conducted. This program consists of three developmental studies, namely the development of the analysis method of molten material discharge from the core region, the development of evaluation methodology of molten material penetration into sodium pool, and the development of the simulation tool of debris bed behavior.

The analysis method of molten material discharge was developed based on the computer code SIMMER-III since this code is designed to simulate the multi-phase,

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multi-component fluid dynamics with phase changes involved in the discharge process. Several experiments simulating the molten material discharge through duct using simulant materials were utilized as the basis of validation study of the physical models in this code. It was shown that SIMMER-III with improved physical models could simulate the molten material discharge behavior including the momentum exchange with duct wall and thermal interaction with coolant.

In order to develop evaluation methodology of molten material penetration into sodium pool, a series of experiments simulating jet penetration behavior into sodium pool in SFR thermal condition were performed. These experiments revealed that the molten jet was fragmented in significantly shorter penetration length than the prediction by existing correlation for light water reactor conditions, due to the direct contact and thermal interaction of molten materials with coolant.

The fragmented core materials form a sediment debris bed in the lower plenum. It is necessary to remove decay heat safely from this debris bed to achieve IVR. A simulation code to analyze the behavior of debris bed with decay heat was developed based on SIMMER-III code by implementing physical models, which simulate the interaction among solid particles in the bed. The code was validated by several experiments on the fluidization of particle bed by two-phase flow.

These evaluation methodologies will serve as a basis for advanced safety assessment technology of SFRs in the future.

Keywords; Sodium-Cooled Fast Reactor, Core Disruptive Accident, Molten Core Material, Safety Assessment, In-Vessel Retention

1. Introduction

Sodium as coolant provides an excellent safety characteristic in removing decay heat from degraded core materials in the postulated unprotected core-disruptive accidents (CDAs) in sodium-cooled fast reactors (SFRs). Due to its high boiling point, the decay heat removal by natural circulation flow is possible without electric power. In order to fully make use of this advantage, the achievement of the in-vessel retention (IVR) of core materials is of prime importance in SFRs, since IVR keeps the configuration that the degraded core materials are covered and cooled by sodium, and the coolant flow paths in primary loop are available for the decay heat removal by natural circulation.

In the unprotected CDA of SFRs, molten core material relocates to the lower plenum of reactor vessel and may impose significant thermal load on the structures, which may result in the melt through of reactor vessel. Figure 1 shows the conceptual figure of the relocation process of molten core material from core to the lower plenum of reactor vessel. The most probable and effective relocation path for remaining core-materials is considered to be Control Rod Guide Tubes (CRGTs), since CRGT locates inside the core region and provides straight flow path to the lower plenum and these paths. The molten core material, which relocates from the core region through CRGT, flows into the single-phase pool of sodium in the lower plenum as a jet of molten material. The previous studies [1, 2] showed that the molten core materials are fragmented to fuel debris at the contact with single-phase sodium. The fragmented fuel debris sediments on the structure surface and forms debris bed.

< Figure 1 >

The achievement of the IVR requires two conditions. The first one is that the fragmentation of molten material jet in the lower plenum occurs before the jet reaches the structure surface in the bottom region of the reactor vessel and starts to erode the structure. The lower plenum should be designed so as to avoid this situation. Another condition is the decay heat removal from debris bed to keep the debris bed in solid phase and retained on the structure surface. The previous studies [3] showed that the sodium inside the debris begins to boil and finally the debris bed melts if the thickness of the debris bed exceeds a certain threshold value. On the other hand, it was also confirmed by in-pile experiment that the debris bed is fluidized by sodium vapor and the debris bed is flattened to thinner thickness than the threshold value [4]. This mechanism is called “self-leveling” of debris bed.

In order to enable the assessment of these relocation processes and prove that IVR of core material is the most probable consequence of the CDA in SFRs, a research program to develop the evaluation methodology for the material relocation behavior in the CDA of SFRs has been conducted. This program consists of three developmental studies, namely (1) the development of the analysis method of molten material discharge from core region, (2) the development of the evaluation methodology of molten material fragmentation in sodium pool, and (3) the development of the simulation tool of debris bed self-leveling behavior. This paper summarizes the results of each developmental study.

2. Molten Material Discharge

The first step in the development of the analysis method of molten material discharge from core region was the identification of phenomena, which dominates the material discharge behavior, by the analysis of the discharge behavior of the molten core materials from core through the typical CRGT of SFR. The analysis showed that the fuel-coolant interaction (FCI) at the failure of CRGT had a large effect on the discharge behavior since the heat loss from fuel to coolant enhances the blockage formation by the core material in CRGT. This analysis also showed the importance of FCI in lower plenum of reactor vessel since the pressure increase by FCIs in the lower plenum affects the discharge behavior. The SIMMER-III [5] code was selected as the basis of this study, since the code was developed to deal with the multi-phase, multi-component flow in the reactor core region during the CDAs of SFRs and systematic validation efforts have been performed on this code [6].

According to the identification of dominant phenomena, the physical models in SIMMER-III code relating to the FCI at the failure of CRGT were validated through the analysis of existing experiments [7]. A schematic figure of the experimental device is shown in Figure 2 [8]. The molten alumina was introduced into the upper trap of test device, and the melt through of inner duct, which contains liquid sodium inside it, sub-sequent thermal interaction of molten alumina and sodium (simulated FCI), and molten alumina discharge through the inner duct to lower trap were observed. The inner diameter of the inner duct was about 30mm, and its length was about 900mm. No significant pressure build up was observed at the failure of the inner duct in the upper trap and during the discharge process of molten alumina through the inner duct. The void sensor located at about 500mm below the bottom of upper trap detected the arrival of sodium vapor region within 0.1s, showing that the expansion of voided region occurred rapidly just after the failure of the inner duct.

< Figure 2 >

The first analysis of this experiment using standard input data of SIMMER-III showed significant pressure generation at the failure of the inner duct as shown in the default case in Figure 3 which was different from the experimental result. This means that the standard model of SIMMER-III overestimates the heat transfer rate from molten alumina to sodium and pressure generation by vaporization of sodium. Due to this overestimation, the molten alumina formed blockage at the outlet of inner duct in the analysis, which is also different from the experimental result.

The calculated result also predicted the formation of the blockage at the outlet of the inner duct due to the formation of alumina particles. However, the volume fraction of the alumina particle was less than 30% and it is too low to justify the blockage formation. Therefore, the blockage formation model was improved to predict the blockage when the volume fraction of alumina could be regarded as continuous component in the flow area.

The current physical model of SIMMER-III calculates the diameter of molten alumina to be the equilibrium diameter determined by the droplet breakup due to relative velocity of alumina and sodium. However, the visual observation in other simulant experiment using a high-density melt (an alloy with low-melting temperature) and water showed that the molten material kept large continuous phase in the coolant at the penetration into inner duct and the breakup of molten material by relative motion with coolant did not occur [8, 9]. The overestimation of pressure generation and the observation of breakup behavior of molten materials in the simulant experiment concludes that the current standard model of SIMMER-III overestimates the breakup of molten alumina and hence the heat transfer rate from molten alumina to sodium, and finally the pressure generation. Therefore, parametric study to investigate the effect of the diameter of molten alumina and sodium were performed. Figure 3 shows the pressure transients varying the diameter of molten alumina and sodium. It was shown that the use of larger diameter gave lower pressure transient, which is closer to the experimental result that almost no pressure generation was observed. The case with molten alumina diameter of 10mm and sodium diameter of 1mm reproduces the discharge behavior that the expansion of voided region occurred rapidly just after the failure of the inner duct and the molten alumina discharged continuously, was properly reproduced. The results of this simulation suggest that the FCI at the failure of a duct structure containing coolant by the surrounding high temperature melt is benign and, in its appropriate simulation by the SIMMER code, it is necessary to restrict the interfacial area between coolant and high temperature melt by setting the diameter of both components to the order of several 10^{-3} m.

< Figure 3 >

Concerning the FCI behavior in the lower plenum, the out-of-pile experiments performed in the project [10] were analyzed. As shown in Figure 4, the experimental apparatus mainly consists of a crucible, a nozzle (inner diameter: from 30mm to 150mm) and a water pool (depth: 1.4m, width: 1.0m, length: 1.0m). The melt (maximum mass: 400kg) of a high-density material (an alloy with low-melting temperature) was generated in the crucible by using sheathed heaters wrapped around the crucible and then discharged into the water pool through the nozzle by withdrawing the plug. Flow rate of the discharged melt was measured by a flow meter installed on the nozzle. Thermocouples were installed on the central axis of the nozzle to measure temperature distribution along a column of melt in the water pool. Pressure sensors were installed in the water pool to measure pressure changes associated with fragmentation of the melt. A high-speed camera (frame rate: 10,000 frames per second) was used to observe fragmentation behavior of the melt.

< Figure 4 >

Figure 5 shows the comparison of the transients of discharged mass of melt [11]. Detailed investigation of the analytical result showed that the underestimation of discharge mass flow in the standard model of the SIMMER code was caused by the suppression of melt discharge by the increase of void fraction due to FCI at the exit of injection nozzle. In order to improve this discrepancy, it is required to model the interface area between melt and water based on the visual observation by high-speed camera. The visual observation of melt penetration into water revealed that the molten melt did not disperse at the exit of the injection nozzle, but kept columnar shape [11]. This observation was taken into the modeling of jet breakup that the interfacial area of melt per unit volume was controlled so as not to exceed the value corresponding to the surface area of columnar jet. The melt discharge behavior showed good agreement with the experiments by this model improvement as shown in Figure 5.

< Figure 5 >

The improved and validated models will be implemented to three dimensional analysis code SIMMER-IV in order to confirm that the discharge of molten core materials through CRGT can be analyzed in three dimensional calculation system coupled with neutronics.

3. Molten Material Fragmentation behavior in Lower Plenum

The objective of the present study is to develop an evaluation method for the distance to fragmentation of molten core material discharged into the sodium plenum. In general, the distance for fragmentation is referred to as "breakup length". It is considered as a length in which an initially coherent stream of molten material disappears due to the breakup of the stream. In the past, some semi-empirical correlations [12,13] have been developed to evaluate the breakup length of molten materials in liquids such as water, based on experimental data and theoretical considerations on the breakup of a molten material blanketed by the thick vapor film of an ambient liquid. Besides, it has been noted that deep penetration of molten core material will be limited in a sodium environment due to thermal interactions resulting from the liquid-liquid direct contact [14]. Actually, several experiments and theoretical models on the fragmentation behavior of molten material in sodium have been reported [15, 16, 17]. Nevertheless, it is believed that there are limited quantitative and well-organized experimental knowledge on the distance for fragmentation under the conditions where a liquid-liquid direct contact is maintained between molten material and coolant. In particular, effects of thermal interactions resulting from the liquid-liquid direct contact have been less studied in terms of evaluation of the distance for fragmentation.

In order to develop an evaluation method of the distance for fragmentation of molten core material discharged into the sodium plenum, fundamental experiments were carried out using a high-density melt (an alloy with low-melting temperature) and water as simulants for the molten fuel and coolant, respectively [10]. In the experiments, the melt was discharged into a water pool through a nozzle (inner diameter: from 30 mm to 150 mm, which is though to be the melt jet diameter) under a simulated CDA condition in which formation of a stable vapor film is inhibited around the melt surface and thus a liquid-liquid direct contact is maintained between the melt and water. The visual observation by high-speed camera showed that the vapor expansion with pressure generation in the vicinity of the melt could facilitate the reduction of the distance for fragmentation. In this experiment, it was possible to clearly determine a distance for fragmentation from a cooling curve shown in Figure 6, assuming that a rapid temperature decrease is initiated by the fragmentation of melt.

< Figure 6 >

The experimental results showed that measured distances for fragmentation of the injected melt are almost smaller than approximately 10 percent of the prediction by the existing correlation developed for light water reactor condition [13] as shown in Figure 7. The correlation was derived based on the consideration that the balance between the inertial force and the resistance forces such as the buoyancy force dominates the breakup length of the molten jet. On the other hand, the pressure generation by FCI in the vicinity of the melt might facilitate the fragmentation, while the existing correlation assumed that the melt penetrates coolant with blanketed by stable vapor film that inhibits the liquid-liquid direct contact. Therefore, it is believed that the large difference between the measured values and predictions is attributed to the fact that the existing correlation does not take into account thermal fragmentation resulting from the liquid-liquid direct contact.

< Figure 7 >

Through this fundamental experiment, basic knowledge was obtained for the future development of an evaluation method of the distance for fragmentation of molten core material. The evaluation method will be reflected to the safety design of lower plenum of future SFRs so as to set rational height of lower plenum to avoid the impingement of melt jet to the lower structure in reactor vessel. This evaluation method will be also utilized in the safety assessment of existing and future SFRs

4. The development of Simulation Tool of Debris Bed Behavior

The fragmented core material sediments on the surface of structure in the lower plenum of reactor vessel to form so called “debris bed”. In order to achieve IVR, the decay heat of fuel debris needs to be cooled stably from the debris bed. If the thickness of debris bed exceeds a certain coolable limit, the sodium in the debris bed begins to boil and the fuel debris may melt again after the dry out of debris bed. However, when the bed thickness exceeds the limit in a certain local area of debris bed, sodium boiling occurs in the debris bed at the location and the formation of sodium vapor bubble will stir up the debris bed, moving the fuel debris from the area to another area with lower thickness, and thus flattens the debris bed. The debris bed thickness in the area becomes lower than the coolable limit thickness, thus enabling the stable cooling of the debris bed. This process is called "self-leveling" phenomena. These phenomena will also lower the neutronic criticality of the debris bed by decreasing the thickness of debris bed. Although the existence of this phenomena was observed in the past in-pile experiment [18], mechanistic simulation tool has not been developed. In this study, two kinds of simulation tool were developed, namely macroscopic and microscopic models. Both model were developed based on SIMMER-III and SIMMER-IV code. Macroscopic model treats the debris bed as one of fluid components in the SIMMER code, while microscopic model traces the motions and interactions of particles itself directly. Although microscopic model requires huge computational resources, it will enable the detailed analysis of the self-leveling behavior in a local-scale of structures in the reactor vessel. On the other hand, macroscopic model will be applied to the self-leveling behavior in reactor scale with practical calculation time.

4.1. Development of Macroscopic Model

The macroscopic model treats the debris bed as one of fluid components in the SIMMER code and introduces physical models, which express the contact and collision of debris. The contact between debris particles is modeled by "particle viscosity" and the collision of debris is modeled by "particle pressure" models [19, 20]. Since the interaction of solid particles in the debris bed behaves like Bingham fluid, a physical model to simulate the behavior of soil in field of civil engineering was introduced to both SIMMER-III and SIMMER-IV. In addition to the physical models for debris bed, the extension of the number of velocity fields in the SIMMER codes for solid particles were performed in order to simulate the sedimentation process of core debris properly.

The experiments simulating the self-leveling of debris bed [21] are analyzed to validate the physical models for solid particle behavior in debris bed. Figure 8 shows the schematic diagram of this experiment. In this experiment, cylindrical cone shaped debris bed of alumina particles in water pool with its diameter of 31cm and the pool depth of 18cm was the initial condition. A gas flow was injected uniformly from the bottom plate of the water pool to simulate self-leveling behavior. The transient of the inclination angle of debris bed were compared with experiments for the case with particle diameter of 1mm and 3mm. The SIMMER code reproduces the experimental results reasonably as shown in Figure 9.

< Figure 8 >

< Figure 9 >

The validation of the multi-velocity field for the particle components was also performed by analyzing the experiments by Snabre [22]. This experiment started with the uniform suspension in a container with inner rectangular section of 4cm by 2cm and height of 12cm, which was composed of the mixture of larger particles with diameters between 180μm and 200μm and small particles about 25μm in diameter and observed the segregation behavior of each particle. In order to measure the spatial distribution of the two particle groups, the particle groups were made fluorescent by impregnation of two kinds of organic dye, which were exited by laser with different wavelength. Figure 10 shows the comparison of experiments and the simulated results. The blank and solid circle in the left figure shows the calculated axial distribution of volume fraction of large and small particles by SIMMER, respectively. The photographs b and b’ in this figure are the snapshots for small and large particles, which were exited by different wage length lasers. SIMMER predicts well the location of each particle group and their boundary.

< Figure 10 >

As the final step in the development of the analysis method of self-leveling, the SIMMER code with macroscopic model was applied to the self-leveling of debris bed in the reactor scale [23]. The debris bed was located in the coolant plenum with its diameter of 3.3 m and height of 0.45m. The height of the mound-shaped debris bed was 31cm at the center and 6.5cm at the periphery. The debris bed consisted of fuel and steel particles and the volume ratio of fuel to steel was 6 to 4 and the total mass of debris bed was 5000kg. The initial temperature was set to 1200K to accelerate the boiling. The simulated decay heat was 2.2×10^7 [W]. Time variation of the spatial distribution of the volume fractions of the debris bed particles and sodium after 8.1 second is shown in Figure 11. As shown in the rightmost legend, the density of red and blue colors corresponds to the value of spatial distribution of the debris bed particles and sodium, respectively, and the spatial distributions of these materials are overlapped in each figure. The vector in this figure shows the superficial velocity of sodium vapor. The sodium in the debris bed begins to boil after 8.1 second. Almost all sodium in the debris bed boils at 9.6 second, and the bed is unpiled by two phase mixture flow of sodium and sodium vapor. Local sodium boiling after 10.0 second further planarizes the bed. This analysis also showed that the SIMMER code with macroscopic model could simulate self-leveling behavior in a reactor scale stably with practical calculation time, i.e. several days.

< Figure 11 >

4.2. Development of Microscopic Model

The microscopic model was developed in order to enable the detailed analysis of the self-leveling behavior in a local-scale of structures in the reactor vessel. The two dimensional and three dimensional discrete element methods (DEM) [24] were implemented to

SIMMER-III and SIMMER-IV, respectively. These codes were named SIMMER-DEM. Since the microscopic model simulates the motion of each particle in the debris bed directly, the microscopic model performs detailed analysis of self-leveling behavior with high accuracy but requires large computational resources. Therefore, the microscopic model will be applied to the local-scale detailed self-leveling behavior with realistic structure configuration, while the macroscopic model will analyze the self-leveling behavior in a reactor scale. The combination of the simulations by two models will provide indispensable information to the safety design of in-vessel debris catcher and the safety assessment of the accident progression towards the achievement of IVR.

The validation of the SIMMER-DEM code was performed [25] by the analysises of available experiments. The first application was the analysis of water sloshing experiments with particles [26]. Figure 12 is the schematic view of the test section and Figure 13 gives the comparison of visual observation of experiments and calculation. The SIMMER-DEM code reproduced the sloshing motion of particles with water appropriately.

< Figure 12 >

< Figure 13 >

The SIMMER-DEM code was also applied to the analysis of the experiment simulating the self-leveling of debris bed [21]. Figure 14 shows the transient of inclination angle showing good agreement of calculation and experimental result. In parallel to the implementation of DEM to SIMMER-DEM, the thread-parallelization using OpenMP and process-parallelization using MPI and their mixture were performed to SIMMER-DEM and considerable acceleration (the speed up ratio of 45.5 using 16 nodes by 16 threads) was achieved for large-scale computation.

< Figure 14 >

5. Conclusion

A research program to develop the evaluation methodology for the material relocation behavior in the CDA of SFRs has been conducted. This program consists of three developmental studies, namely the development of analysis method of molten material discharge from core region, the development of evaluation methodology of molten material fragmentation in sodium pool, and the development of simulation tool of debris bed behavior.

In the development of analysis method of molten material discharge from the core region, the SIMMER code was utilized as a technical basis of the development and the applicability of SIMMER code to the discharge behavior of molten core materials through CRGT were validated through the analysis of experiments. The development of evaluation methodology of molten material fragmentation in sodium pool was achieved by experimental study to discharge hot molten simulant materials into water with realizing the liquid-liquid contact specific in the SFR condition. The visual observation using high-speed camera gave understanding of fragmentation mechanism that the vapor expansion with pressure generation in the vicinity of the melt facilitates the reduction of the distance for fragmentation. Through this fundamental experiment, basic knowledge was obtained for the future development of an evaluation method of the distance for fragmentation of molten core material. The development of macroscopic and microscopic calculation codes based on SIMMER enabled complementary and practical mechanistic simulation of debris bed behavior, so that the parameters of macroscopic model could be estimated by the analyses using microscopic model, and the boundary condition of the local analytical system of microscopic model could be estimated by the whole-reactor scale analysis by macroscopic model.

The developed evaluation methodology described above can be applied to the safety assessment of existing SFRs and also to the safety design of in-vessel structure of future SFRs to achieve IVR, thus contributes to the enhancement of safety and economic competitiveness of future SFRs.

Acknowledgements

The present study is the result of "Development of Evaluation Methodology for Core-Material Relocation in Core Disruptive Accidents" entrusted to Japan Atomic Energy Agency by the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT). The authors would like to acknowledge invaluable contributions that the NNC RK team made to this study. The authors would also like to express their gratitude to Messrs. M. Sugaya and S. Hosono of NESI Inc., for their important technical contribution

References

[1] Theofanous TG, and Bell CR. An assessment of Clinch River Breeder Reactor core disruptive accident energetics. Nucl. Sci. Eng. 1986: 93:215–228.

[2] Tobita Y, Morita K, Kawada K, Niwa H, Nonaka N. Evaluation of CDA Energetics in the Prototype LMFBR with Latest Knowledge and Tools. Proc. ICONE-7; 1999 April 19-23; Tokyo (Japan). ICONE-7145.

[3] Endo H, Kubo S, Kotake S, Sato I, Kondo Sa, Niwa H, Koyama K. Elimination of recriticality potential for the self-consistent nuclear energy system. Progress in Nucl. Energy. 2002: 40: 577–586.

[4] Kotake S, Sakamoto Y, Mihara T, Kubo S, Uto N, Kamishima Y, Aoto K, Toda M. Development of advanced loop-type fast reactor in Japan. Nucl Technol. 2010 April: 170: 133–147.

[5] Tobita Y, Kondo Sa, Morita K, Maschek W, Coste P, Cadiou T. The Development of SIMMER-III, An Advanced Computer Program for LMFR Safety Analysis and Its Application to Sodium Experiments. Nuclear Technology. 2006 March: 153: 245-255.

[6] Tobita Y, Kondo Sa, Yamano H, Fujita S. The Development of SIMMER-III, An Advanced Computer Program for LMFR Safety Analysis. Proc. IAEA/NEA Technical

Meeting on Use of Computational Fluid Dynamics (CFD) Codes for Safety Analysis Reactors Systems Including Containment; 2002 Nov 11-14; Pisa (Italy).

[7] Kamiyama K, Tobita Y Suzuki T, Matsuba K. Development of an Evaluation Methodology for Fuel Discharge in Core Disruptive Accidents of Sodium-Cooled Fast Reactors. Journal of Energy and Power Engineering. 2014; 8; 5: 785-793.

[8] Kamiyama K, Saito M, Matsuba K, Isozaki M, Sato I, Konishi K, Zuyev AV, Kolodeshnikov AA, Vassiliev YS. Experimental study on fuel-discharge behaviour through in-core coolant channels. 2013 May; 50; 6: 629–644.

[9] Matsuba K, Imahori S, Isozaki M. Experimental Study on Void Development Behavior in a Simulated Coolant Channel. Proc. 6th Int. Topical Meeting on Nuclear Reactor Thermal Hydraulics, Operation and Safety (NUTHOS-6). 2004 Oct 4-8; Nara (Japan).

[10] Matsuba K, Isozaki M, Kamiyama K, Tobita Y, Suzuki T. Fundamental experiment on the distance for fragmentation of molten core material during core disruptive accidents in sodium-cooled fast reactors. International Journal of Nuclear Safety and Simulation. 2014 Nov; 4; 4,: 272-277.

[11] K. Kamiyama, Y. Tobita, T. Suzuki, K. Matsuba, "Development of an Evaluation Methodology for the Fuel-Relocation into the Coolant Plenum in the Core Disruptive Accident of Sodium-Cooled Fast Reactors," Proceedings of ICONE-22, July 7-11, Prague, Czech Republic (2014), ICONE22-30308.

[12] Spencer BW, Gabor JD, Cassulo JC. Effect of boiling regime on melt stream breakup in water. Proc. 4th Miami Int. Symp. on Multi-phase Transport and Particulate Phenomena; 1986 Dec 15; Miami Beach Florida (USA).

[13] Saito M. Experimental study on penetration behavior of water jet into Freon-11 and liquid nitrogen. ANS Proc. 1988 National Heat Transfer Conf; 1988; Huston, Texas (USA).

[14] Kondo Sa, Isozaki M, Imabori S, Furutani A, Brear DJ. Experimental study on simulated

molten jet-coolant interactions. Nucl. Eng. Des. 1995; 155: 73-84.

[15] Chu TY. Fragmentation of molten core material by sodium. Proc. Int. Topical Mtg. on LMFBR safety; 1982; Lyon (France).

[16] Magallon D, Hohmann H, Schins H. Pouring of 100-kg-scale molten UO2 into sodium. Nucl. Tech., 1992; 98: 79-90.

[17] Epstein M, Fauske HK. Applications of the turbulent entrainment assumption to immiscible gas–liquid and liquid–liquid systems. IChemE, 2001; 79 (Part A): 453-462.

[18] Ottinger CA. Coolability of stratified UO2 debris in sodium with downward heat removal: The D13 experiment. Sandia National Labs.; Albuquerque, NM 87185, 1987, NUREG/CR-4719 SAND86-1043.

[19] Anderson, K., et al. (1995). "Instabilities and the Formation of Bubbles in Fluidized-Beds," J Fluid Mech, 303, pp. 327-366.

[20] Schallenberg, J., et al. (2005). "The important role of local dispersed phase hold-ups for the calculation of three-phase bubble columns," Chem. Eng. Sci., 60, pp. 6027-6033.

[21] Cheng S, Yamano H, Suzuki T, Tobita Y, Gondai Y, Nakamura Y, Zhang B, Matsumoto T, Morita K. An experimental study on self-leveling behavior of debris beds with comparatively higher gas velocities. Proc. 8th Japan-Korea Symp. Nucl. Thermal Hudraulics and Safety (NTHAS-8); 2012 Dec; Beppu (Japan). N8P1003.

[22] Snabre P, Pouligny B, Metayer C, Nadal F. Size segregation and particle velocity fluctuations in settling concentrated suspensions. Rheol. Acta., 2008 Dec; 48: 855-870.

[23] Tagami H, Tobita Y. Numerical Simulation for Debris Bed Behavior in Sodium Cooled Fast Reactor. Proc. NUTHOS10, 2014 Dec 14-18; Okinawa (Japan). NUTHOS10-1141.

[24] Cundall PA, Strack ODL. A discrete numerical model for granular assemblies. Geotechnique. 1979 March; 29: 47-65.

[25] Guo L, Morita K, Tobita Y. Numerical Simulation of Gas-Solid Fluidized Beds by Coupling a Fluid-dynamics Model with the Discrete Element Method. Annals of Nuclear

Energy. 2014 Oct; 72: 31-38.

[26] Liu P. Dynamic Behavior of Multiphase Flow with Solid Particles. Kyusyu Univ. 2006 Doctoral thesis.

Figure captions

Figure 1. Supposed event sequence of the core disruptive accidents in SFRs.

Figure 2. Apparatus of experimental facility and test devices [8].

Figure 3. Pressure transient at the failure of inner duct [5].

Figure 4. Schematic of the Experimental Apparatus for Fundamental Experiments [10].

Figure 5. Comparison of Discharged Mass between the Experiment and Calculation [11].

Figure 6. Examples of axial temperature distributions on the central axis of the melt column [10].

Figure 7. Comparison between measured distances for fragmentation and the predictions by the existing representative correlation [10].

Figure 8. Schematic apparatus of self-leveling simulation test [16].

Figure 9. The transient of inclination angle in the experiment and SIMMER analysis [19].

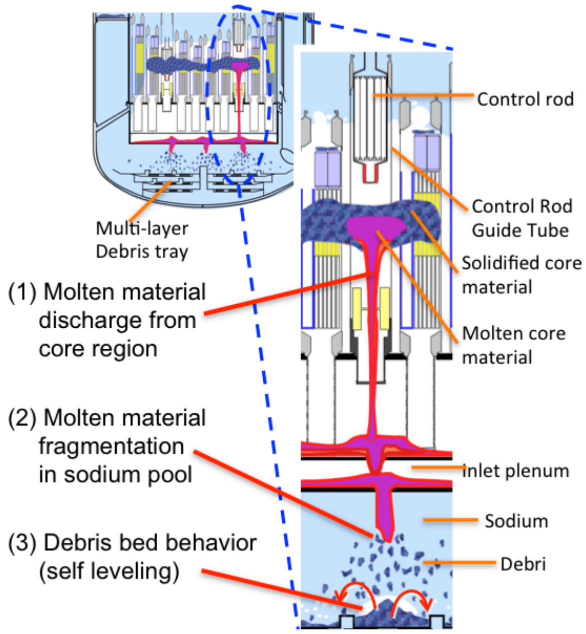
Figure 10. Comparison of segregationb behavior of SIMMER calculation and experiment [20].

Figure 11. Volume fractions of debris bed particles (red) and sodium (blue) in vertical-cross section for each time. Vector indicates vapor superficial velocity [21].

Figure 12. Schematic of the experimental apparatus for dam sloshing experiment [24].

Figure 13. Comparison of Dam Sloshing Experiment and SIMMER-DEM Simulation [24].

Figure 14. Comparison of self-leveling simulation experiment and SIMMER-DEM result.



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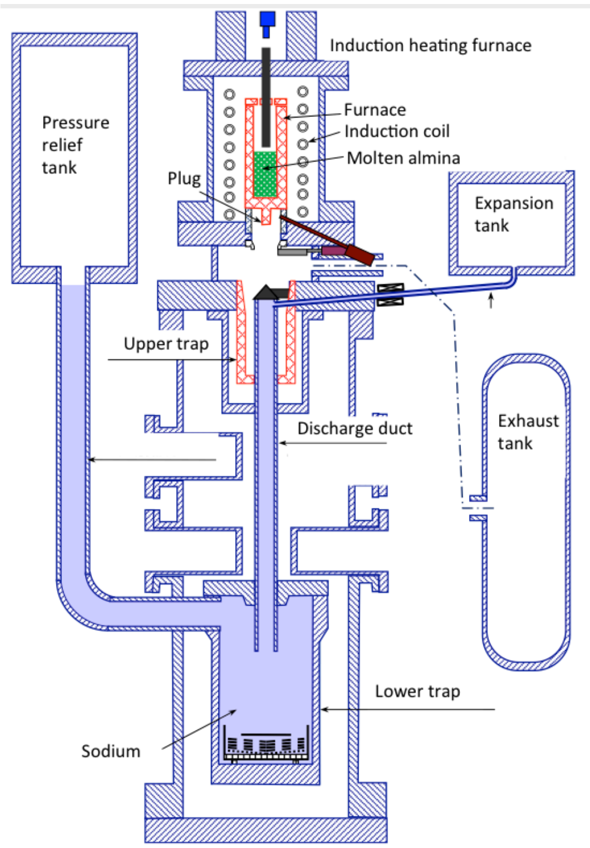

Figure 2. Apparatus of experimental facility and test devices [8].

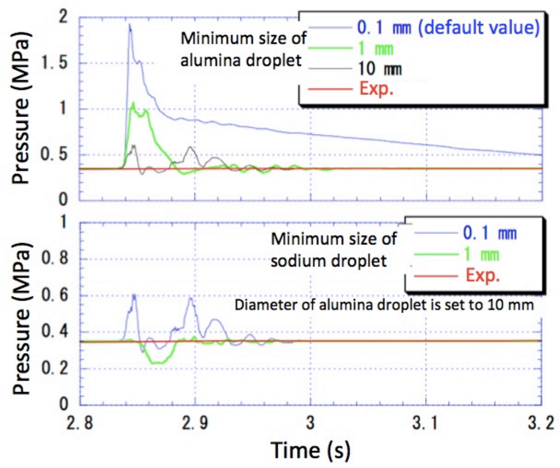

Figure 3. Pressure transient at the failure of inner duct [5].

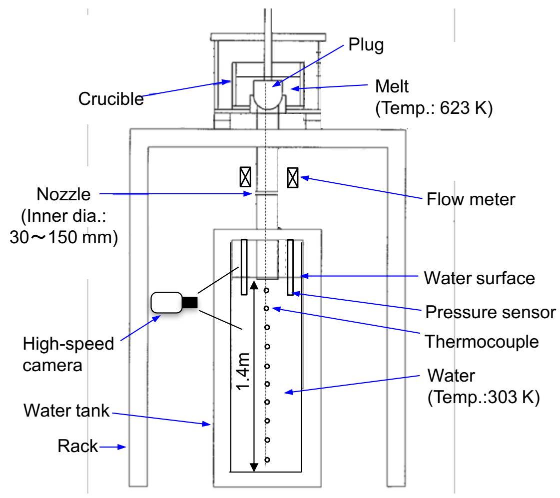

Figure 4. Schematic of the Experimental Apparatus for Fundamental Experiments [10].

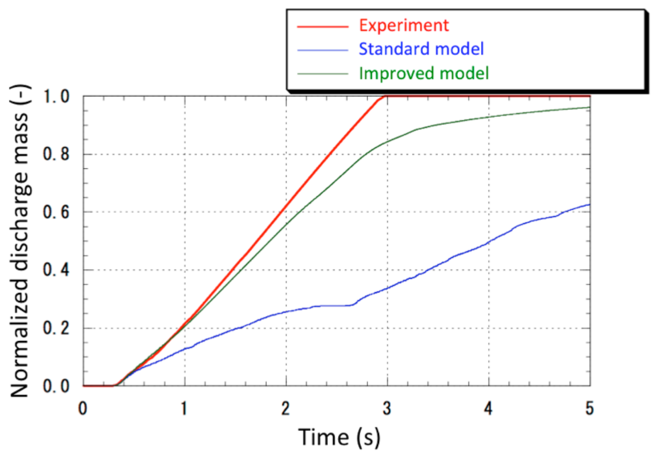

Figure 5. Comparison of Discharged Mass between the Experiment and Calculation [11].

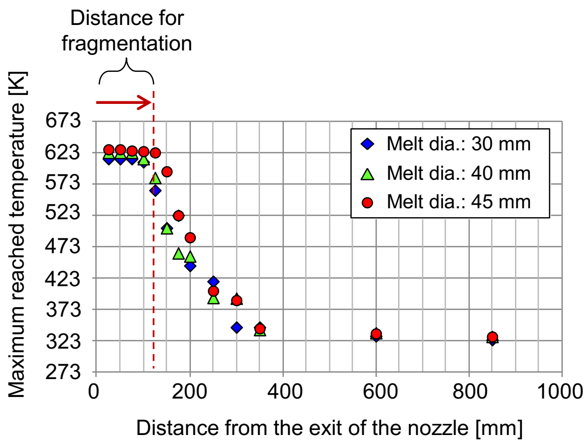

Figure 6. Examples of axial temperature distributions on the central axis of the melt column [10].

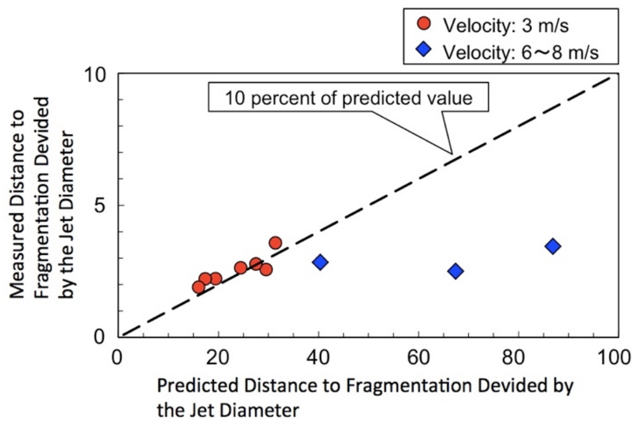

Figure 7. Comparison between measured distances for fragmentation and the predictions by the existing representative correlation [10].

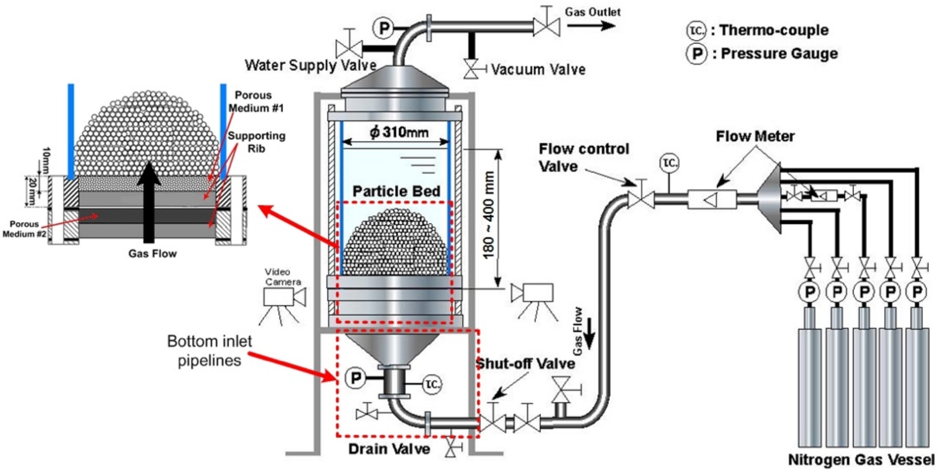

Figure 8. Schematic apparatus of self-leveling simulation test [16].

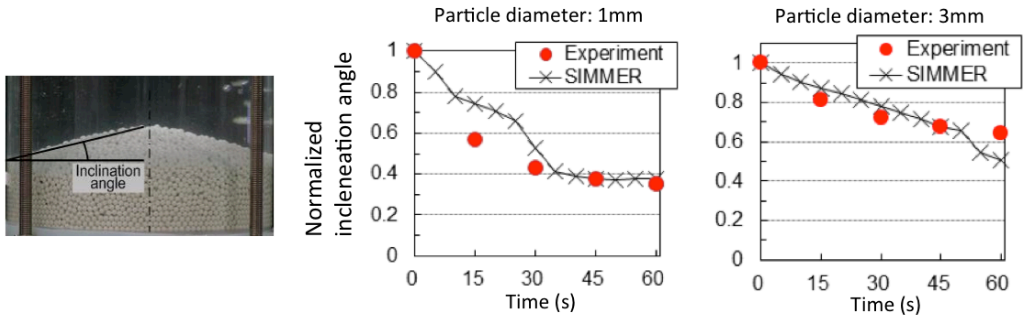

Figure 9. The transient of inclination angle in the experiment and SIMMER analysis [19].

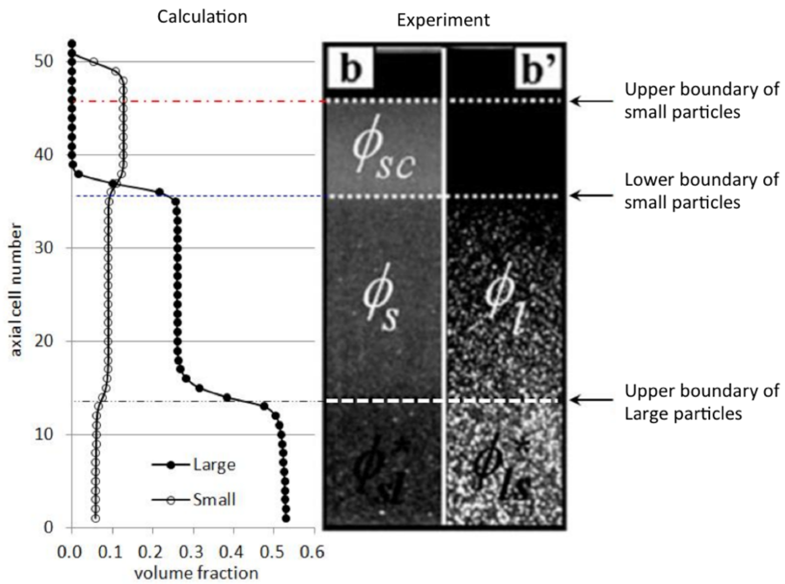

Figure 10. Comparison of segregationb behavior of SIMMER calculation and experiment

[20].

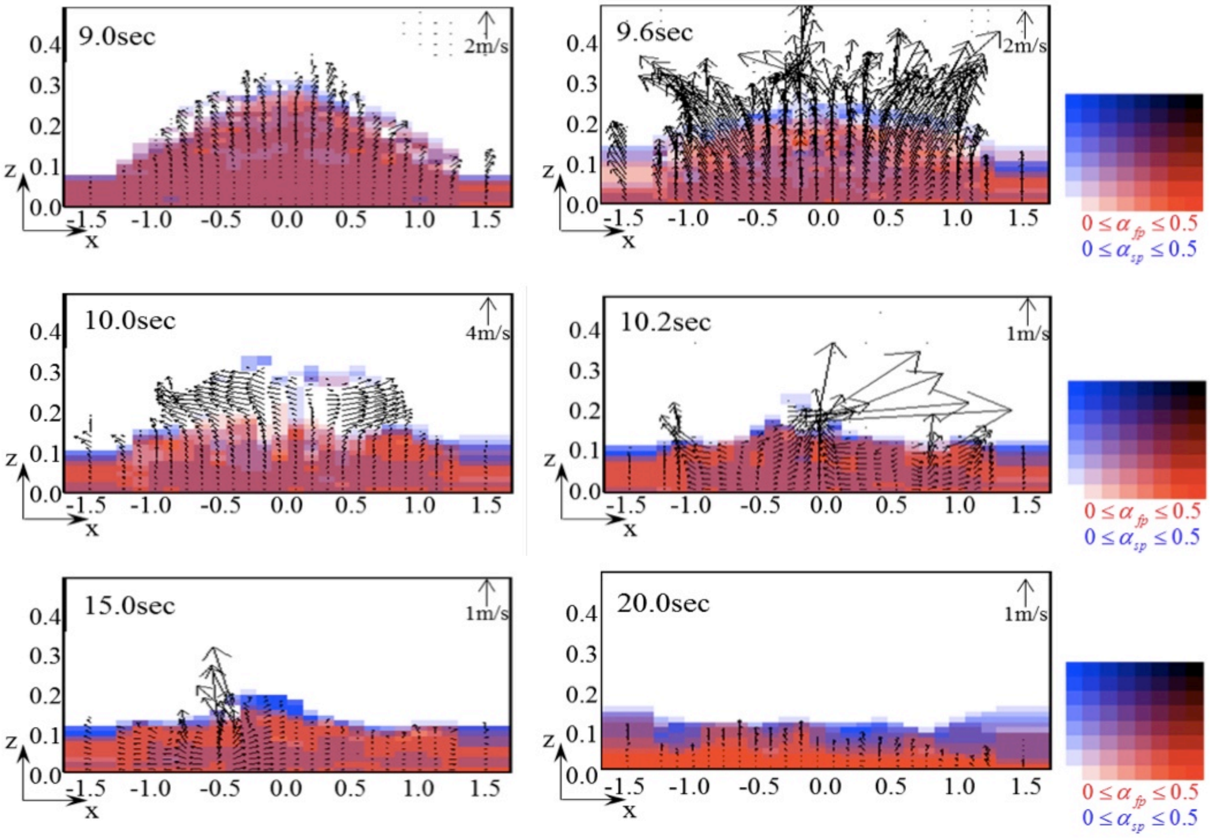

Figure 11. Volume fractions of fuel (red) and steel (blue) in vertical-cross section for each time. Vector indicates vapor velocity [21].

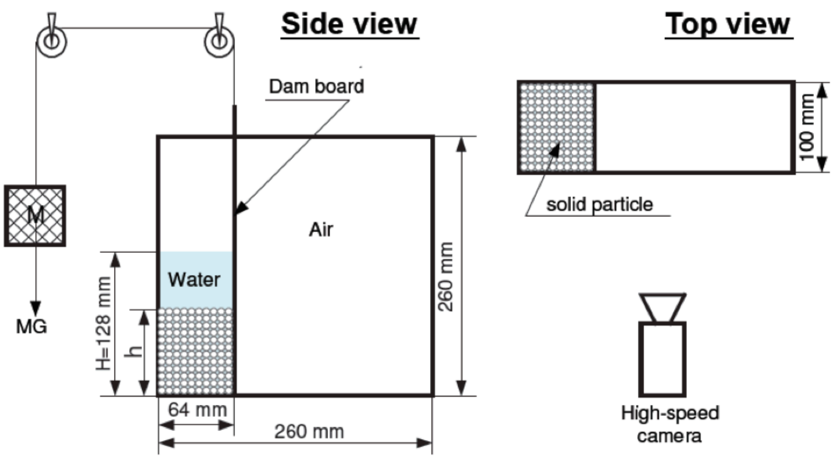

Figure 12. Schematic of the experimental apparatus for dam sloshing experiment [24].

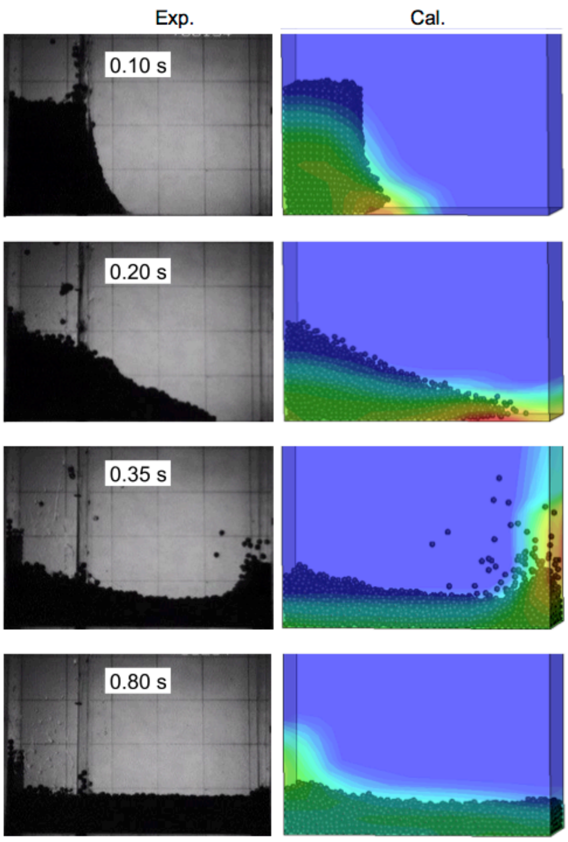

Figure 13. Comparison of Dam Sloshing Experiment and SIMMER-DEM Simulation [24].

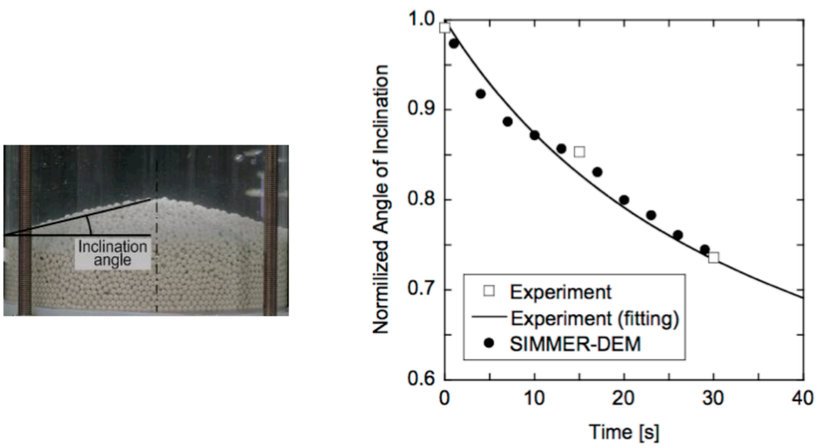

Figure 14. Comparison of self-leveling simulation experiment and SIMMER-DEM result.