NURETH14 Keynote

Thermal Hydraulics of Sodium-cooled Fast Reactors

- Key Issues and Highlights -

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Abstract

In this paper key issues and highlighted topics in thermal hydraulics are discussed in connection to the current Japan's sodium-cooled fast reactor development efforts. In particular, design study and related researches of the Japan Sodium-cooled Fast Reactor (JSFR) are focused. Several innovative technologies, e.g., compact reactor vessel, two-loop system, fully natural circulation decay heat removal, and recriticality free core, have been investigated in order to reduce construction cost and to achieve higher level of reactor safety. Preliminary evaluations of innovative technologies to be applied to JSFR are on-going. Here, progress of design study is introduced. Then, research and development activities on the thermal hydraulics related to the innovative technologies are briefly reviewed.

1. Introduction

Preamble: *Nuclear energy is faced with major challenges such as radioactive waste management, economics, non-proliferation, and enhanced safety with lessons learned from the recent Fukushima Daiichi accidents that took place March 11, 2011. We recognize the heightened importance of plant safety, where thermal hydraulics research and development plays a significant role in supporting safe operation of nuclear plants and fostering safety culture. In view of the global nature of these nuclear issues, it is imperative that forums such as NURETH provide a global communication channel to enhance exchange of ideas and critical information and encourage cross-fertilization of research and development efforts among all nuclear countries. The contexts that follow do not yet include lessons learned from the Fukushima Daiichi accidents but would not alter much in the near future since the technologies described are to meet further enhanced safety requirements to be expected after the 3/11 accident.*

1.1 Sodium-cooled fast reactors

Fast reactors are defined as those which use the fast neutrons for fission reactions taking advantage of the larger number of fission neutrons produced by fission. Therefore, the fast reactor core is designed such that newly born fast neutrons should not be slowed down and must assume a coolant with a lower slowing down capability or a lower coolant volume fraction (i.e., higher fuel fraction) that would result in the tight fuel lattice configuration. This means very narrow coolant channels and requires a fluid of very high heat transfer performances as a coolant.

Sodium has been consistently a candidate for the coolant in Japan in spite of the fact that several heavy liquid metal coolants like Pb or Pb-Bi eutectic alloy are also candidates mainly because of their better neutronics characteristics than sodium. Sodium is a good choice because of its excellent heat transport performances and excellent compatibility with the structure materials (mostly stainless steel) leading to no serious corrosion problems. The excellent heat transport performances are achieved by its excellent heat transfer characteristics as well as it excellent natural circulation decay heat removal capabilities.

One of the most favorable features of using the sodium as a coolant is the fact that the system does not have to be pressurized. (Note that He-gas has been also a good candidate but must be pressurized.) Therefore the LOCA scenario of various sizes being assumed for LWRs are not applied to the sodium cooled fast reactor system. In fact, should PSA level-1 be carried out, no design basis events would be found leading directly to serious core damages. Also it is pointed out that as long as the sodium level in the primary system is maintained, which can be done by providing guard vessels or guard pipes, the sodium will assure the natural convection decay heat removals that keep the core cooling. This is due to its excellent heat removal capabilities. These are big advantages over the use of pressurized water in LWRs.

1.2 On the use of sodium as a coolant

Since the operating pressure is low, depressurization is not necessary after nuclear shutdown of the core. Therefore no loss of coolant is assumed that took place unfortunately in the course of accidents at three units of Fukushima Daiichi Nuclear Power Station (1F NPS) from March 11, 2011. The core meltdown in these BWRs was due to loss of coolant in the core followed by loss of ultimate heat sink and no immediate coolant make-up after tsunami attacked the 1F NPS. In FBR systems, there is no coolant evaporation due to depressurization because the operating pressure is low. Also it is not likely for sodium coolant systems to suffer the loss of ultimate heat sink because the decay heat can be removed by natural circulation that requires no AC/DC power.

While safe decay heat removals are guaranteed by sodium, it does not mean we don’t have to conduct any further fundamental thermal hydraulics researches. Because the natural circulation does depend on the system design and components arrangement, we must optimize the design to maximize the heat transfer efficiency. In order for the optimization, more complete knowledge would be required for relatively low Re number turbulent flow and heat transfer phenomena in the core and heat exchangers and also for extremely low Re number laminar flow regime. Current state of the knowledge is low in this range of thermal hydraulics, in particular for buoyancy-driven flows in tight lattice subassemblies under low power conditions. These are relatively new subjects of recent investigation in contrast to the past investigations in the high heat flux forced convection phenomena including sodium boiling under the other extreme accident conditions.

Drawbacks of using sodium remain and are represented by the chemical reactions with water producing H_2 and the sodium fire in the air environment. The chemically active nature of sodium requires always inert gas environment. This makes component maintenance and repair work more difficult and higher cost. Also the sodium is opaque that makes the flow measurement more difficult than water.

These drawbacks are sometimes over-stated but could be overcome by suitable design consideration for safety and by technology innovation. By design, sodium-water reactions, which are made not likely to occur, should be accommodated, if any, and isolated from the primary system. Sodium fire is not energetic and can be put under the control without much technical difficulty. These events or incidents are not threatening to the core integrity. Even if not threatening, skirmishes over the incident were to be calmed down with additional cost.

We have learned lessons from the aftermath of the Monju sodium leak incident in 1995 that several measures were added for precautions against the sodium leak even in the secondary systems and also against the sodium-water reactions. These precautions should be rationalized in the future. At the moment, the benefits from using sodium must be emphasized and the drawbacks above should not be the basis on which designers refuse the sodium as a coolant.

Recommending sodium, however, we are not totally optimistic for the use of sodium yet for the following reason: It makes the maintenance and repair work in sodium-cooled FBR plants much more difficult than conventional nuclear power plants. The issue should be resolved. Otherwise this may become more critical when the FBR commercialization comes nearer.

Other features different from LWRs are:

- Volumetric power density is usually 3 to 4 times higher, or the core size is smaller, i.e., the core is made very compact. For this reason, fuel subassembly local faults should be evaluated.
- In practice, a hypothetical planer blockage was assumed of different sizes for local faults safety evaluations in the past. It is, however, difficult to identify the cause of local blockages. There was also a discussion that a total inlet blockage should be considered but this issue did not survive because the fuel subassembly inlet nozzle design eliminates any possibilities of blockage formation by foreign materials.
- More abundant neutron population in the core allows many fast reactor options including those for Pu burner, minor actinide incineration and fission products transmutations.

2. A Brief History to 4S and Sodium-cooled Fast Reactor (JSFR) development

From the beginning of nuclear energy program in 1960's, there has been a strong commitment to FBRs in Japan. The US CRBRP project was cancelled in 1983. Germany cancelled SNR300. Due to the elevated construction costs, many minor troubles reported in the French Super Phenix and with the fatal Chernobyl accident in 1986, overall enthusiasms in the US and European countries for FBR development faded away in 1980's to 1990's. During this world-wide stagnation and decline in FBR development, Japan has kept its FBR development program although the budget has been decreasing slowly. Nevertheless, in parallel to the governmental efforts, Japanese utility group continued its Demonstration Fast Breeder Reactor (DFBR) design study since 1978. In 1994, Japan Atomic Power Co. finalized a 660MWe plant design. This project was closed in 1999 and merged to the Feasibility Study initiated by Japan Nuclear Cycle Development Institute (JNC), currently Japan Atomic Energy Agency (JAEA).

Figure 1 displays the planned installation of nuclear power up to 2100, consisting of mainly LWRs until 2050, suggested by JAEC in its Framework. Note that in the figure the installed capacity is assumed to saturate at a level of 58GWe just for illustrative purpose.

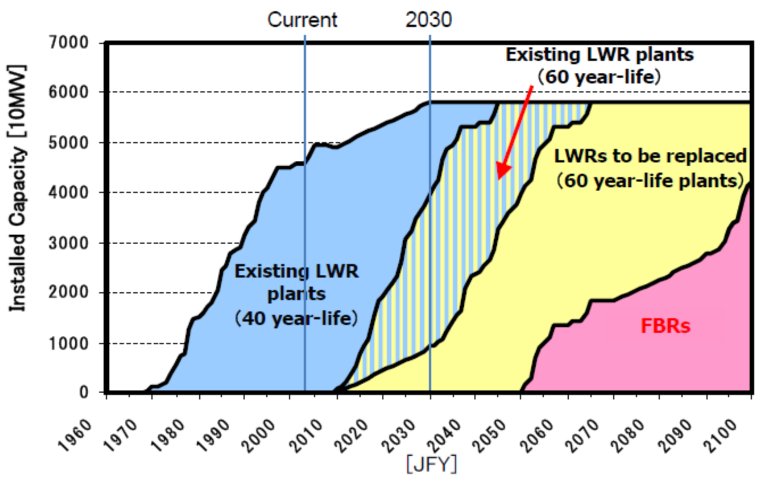


Fig. 1 Planned installation of nuclear power up to 2100

Thus, the FBR commercial timeline was kept being pushed out, and in the Framework for Nuclear Energy (2005), commercial FBRs are envisaged by 2050. Japan's nuclear energy policy now includes having a sodium cooled demonstration fast reactor JSFR (Japan Sodium-cooled Fast Reactor) in operation by 2025 [1].

A 4S concept has been proposed more than 20 years ago and designed by Toshiba and CRIEPI. Figure 2 shows a sketch of the primary system configuration. 4S stands for Super-Safe, Small & Simple. Both 10 MWe and 50 MWe versions are available. It uses sodium as a coolant with electromagnetic pumps to cool a low power density U-10%Zr metallic fueled core, with reactivity control by movable reflectors, and assures passive safety features, notably negative temperature coefficients, negative void reactivity and natural circulation decay heat removals capabilities. It assures 30 years of continuous operation without refueling. The design attracted considerable support in Alaska and the end of 2004 the town of Galena granted initial approval for Toshiba to build a 4S reactor in that remote location. A pre-application NRC review is under way with a view to application for design approval in 2012 (delayed from 2009 by the NRC workload). See Ref. [2].

Reflecting all the staggering experiences in Japan, in 1999 JNC initiated a feasibility study to review promising concepts, define a development plan by 2005 (Phase I), peer-reviewed in 2010 and establish a system of FBR technology by 2015. The parameters are: passive safety, economic competitiveness with LWR, efficient utilization of resources (burning TRUs and depleted U), reduced wastes, proliferation resistance and versatility. As a result of Phase II of the "Feasibility

Study on Commercialization of Fast Reactor Cycle Systems (FaCT)," the JSFR design (see Fig. 3) has been selected out of four basic fast reactor concepts with different coolants and fuel types for the future demonstration by 2025 and commercialization by 2050. Mitsubishi Heavy Industries (MHI) is a lead company in a consortium to build the JSFR [1]. It could be of any size from 500 to 1500 MWe. In this connection MHI has also set up the Mitsubishi FBR Systems. Requirements of higher safety and economical efficiency lead to some innovative concepts in JSFR. Examples of the innovative concepts are listed below:

- Enhanced passive features of reactor shutdown system and decay heat removal system,
- In-vessel retention in core disruptive accident (CDA) based on recriticality free concept,
- Compact reactor vessel based on hot-vessel concept and using simplified fuel handling system,
- Two-loop system for reduction of component number and scale merits of components,
- High reliability steam generator using double-walled tubes.

Several thermal hydraulic challenges have been recognized and studied for these concepts. Here, progress of design study is introduced and some of the activities on thermal hydraulic studies are shown as the current status of JSFR development.

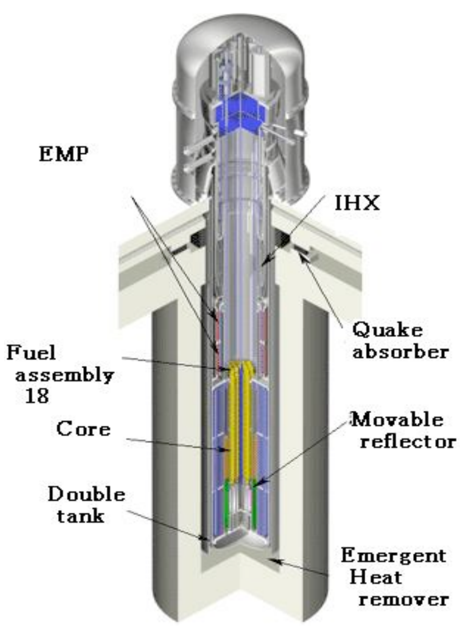


Fig. 2 Cut-view of the 4S Reactor Primary System (10MWe version)

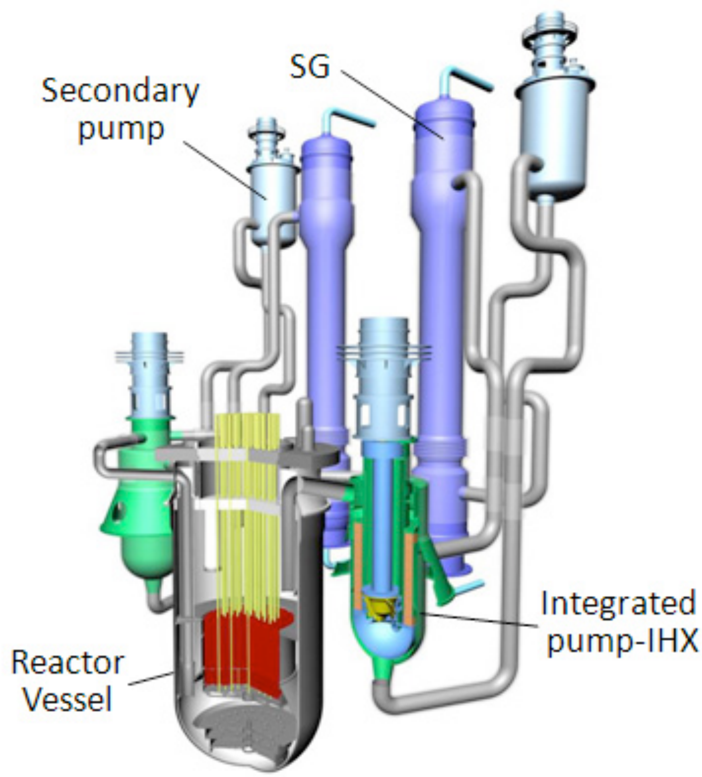


Fig. 3 Schematic of Reactor Vessel and Main Cooling Systems of JSFR

3. Current status of design study

3.1 4S fast reactor design study

4S reactor (Super-Safe, Small & Simple) represents an excellent result of the SMR efforts made in Japan. Its conceptual design study was initiated by Toshiba 1988. The 4S reactor was selected as one of the candidate reactors in GNEP SMR WG in 2006 and has been in the pre-review phase by USNRC since 2007. The design has been completed featuring 30 year of refuelling interval, passive safety, minimum moving mechanical parts and full security/safeguards system in the underground reactor building. A 10 MWe version uses (U-10%Zr) metallic fuel in the core consisting of 18 fuel subassemblies. Enrichment is 17% for the 6 inner core subassembly fuels and 19% for the outer core subassembly fuels. Each subassembly consists of 169 fuel rods of 14 mm in diameter, 2,500 mm active length. Fuel rods are tightly packed in a triangular array configuration with wire spacers and its pitch to diameter ratio is 1.08. The power density is very low with the average linear heat rate is 90W/cm (1/4 of the Monju q'). The coolant core inlet temperature is 355°C and its core outlet temperature is 510°C. Under the normal operating condition, the flow is driven by EM pump and Reynolds number in the core region is about 15,000.

A 30 MW double wall tube steam generator is employed. In case of off-normal shutdown and accidents, passive decay heat removals are assured with RVACS and IRACS (Intermediate Reactor Auxiliary Cooling System) requiring no AC/DC power.

3.2 JSFR enhancement of safety

3.2.1 Reactor shutdown system

Enhancement of safety is a key factor to be considered in the core and fuel design. As the passive shutdown system, JSFR adopts a SASS using a Curie-point electromagnet [3]. The SASS is characterized mainly by no movable parts, less uncertainty, and inspection capability through a measurement of drop-out current in the electromagnet. Several experimental studies for the SASS (e.g., thermal resistance test, thermal transient test, demonstration of the holding stability in JOYO [4], etc.) have been performed for past two decades, so that its development has been mostly finished. The SASS can be adapted to JSFR by determining control rod stable holding temperature, which is specified considering uncertainty of core outlet temperature, and detachment temperature. A full-scale mock-up test for the confirmation of holding force characteristic is also scheduled.

3.2.2 Decay Heat Removal System

In addition to the SASS, the passive decay heat removal function is introduced to enhance the prevention capability. The decay heat removal system (DHRS) for JSFR is designed to be operated by fully natural circulation, which requires no active components like pumps [5]. The DHRS consists of a combination of one loop of direct reactor auxiliary cooling system (DRACS) and two loops of primary reactor auxiliary cooling system (PRACS). Considering the innovativeness of the fully natural circulation DHRS, a plant dynamics calculation for a loss-of-offsite-power accident which is one of DBEs has been performed, and the results have fulfilled the safety criteria of cladding and coolant boundary to show that the natural circulation DHRS for JSFR is effective.

The natural circulation DHRS is utilized in almost all scram events including manual reactor trip. The steam generator is not used, even if off-site power is available. Then, cooling power of DHRS is increased by factor of 1.5 to reach hot standby operation mode within 3 hours by using only DHRS.

3.2.3 In-vessel Retention in CDA

Furthermore, an inner duct structure is provided in each of the core fuel subassembly to enhance discharge of molten fuel in case of CDAs, so that the severe power excursion due to recriticality could be avoided [6]. The molten fuel discharge behavior through the inner duct tube in a subassembly was experimentally confirmed. The material relocation and heat removal phases during CDA were preliminarily evaluated. The current evaluation led to the prospect of the in-vessel retention in CDA.

3.3 JSFR Compact reactor vessel

A compact reactor system is significant for the purpose of economic competitiveness of JSFR, with attention of the accessibility to as many in-vessel components as possible in view of capability of in-service inspection and repair (ISI&R), and the robustness even against the severest class earthquake condition. A compact reactor vessel (RV) is a notable feature for this purpose, and several design measures are taken to realize this concept [7].

A radial slit is opened in the upper internal structure (UIS) so that a pantograph-type arm of fuel handling machine (FHM) can move in the slit. Then, there is no need to move the UIS largely from the core top during refueling operation. This fuel handling system can use a single rotation plug and contribute to reduce the RV diameter.

"Hot vessel" concept using a thinner RV wall is pursued where the RV is equipped with no layer to cool down the RV wall. This is adaptable for a potential benefit of SFR as a low pressure system. In other words, the thinner RV wall to restrict the impact of thermal transient is feasible, however, specific care is given to the RV wall thickness so as to be well tough against even the severe earthquake. Recently the earthquake condition for the JSFR design was reviewed and modified to be severer in order to consider Chuetsu-oki earthquake occurred in 2007. A seismic evaluation of the reactor system was carried out and following revisions were made: 1) increase of RV wall thickness and 2) optimization of natural periods of seismic isolation system [8]. The RV wall thickness influences thermal load caused by the thermal stratification, however, the hot vessel concept can be maintained.

3.4 JSFR Two-loop cooling system

The two-loop cooling system is a key design feature of JSFR to enhance economic efficiency. Since a two-loop configuration is adopted even for a large-size plant design as 1,500MWe power output, the coolant flow rate in each loop increases and the diameter of piping is enlarged. The coolant velocity in the hot leg pipe reaches 9.2 m/s. Further, a simple L-shaped piping design with a short elbow, where curvature radius is the same as the pipe diameter, is used for a compact plant configuration. Higher coolant velocity and short-elbow may raise an issue on flow-induced vibration of pipes. Thus, several experimental studies using water test facilities have been carried out to understand the phenomena regarding the flow-induced vibration.

The primary and secondary piping layouts with drastically shorter length and fewer elbows than a conventional reactor like Monju are designed for JSFR, taking advantage of Mod. 9Cr-1Mo steel which has a high strength at the elevated temperature and also a low thermal expansion coefficient. These piping layouts result in a compact plant configuration through a close arrangement of components, as well as reduction of the amount of piping materials.

As for a welded joint of Mod.9Cr-1Mo steel, creep strength decreases comparing with base metal at the high temperature and long time duration, which is known as Type-IV damage. The creep strength reduction by this Type-IV damage [9] was taken into account in the hot-leg piping design of primary cooling system, although definite reduction has not been observed at the operation temperature of the primary hot-leg piping, i.e., 550 ˚C. Based on the available experimental data and considering enough conservativeness, a provisional value of allowable stress was proposed. This allowable stress is applied to the hot-leg piping design.

3.5 JSFR Steam Generator (SG) with double-walled straight tubes

In the SG unit design, prevention and mitigation of water leakage accidents is the most important issue. JSFR has a large output of 1,500MWe and consists of the two-loop cooling system. Therefore, one SG unit has a large capacity of 750MWe. This large capacity requires higher reliability of heat transfer tube, because a larger number of tubes may cause higher possibility of failure, and a longer time is required for water leakage detection than in conventional SGs with small capacity. Following targets were set for the SG conceptual design using double-walled straight tubes:

- For high reliability on sodium/water boundary

The commercial SFRs must have higher availability for economic performance. Sodium/water boundaries, especially heat transfer tubes, should have lower possibility of water leak. The tube is a double-walled type composed of inner and outer tubes, and both tubes are mechanically contacted. Each wall has a complete volumetric inspection; the outer and inner tubes are inspected by ultrasonic test and eddy current test, respectively.

- For mitigation function of sodium/water reaction

Sodium/water boundaries, especially the tubes, must have mitigation function of sodium/water reaction for property protection. Design target is non-propagation of tube failure. Mechanically contacted double-walled tube has a very small gap between the walls. This gap restricts water leak flow rate and the maximum leak flow rate is under the tube failure propagation limit.

Following trial manufacturing on the double-walled tube SG have been carried out in the FaCT project; mechanically contacted double-walled long tube, forged alloy for tube sheet, and tube-tube sheet joint. Another tube design study is also performed as an alternative to the double-walled tube.

4. Highlighted R&D on thermal hydraulics

4.1 4S tight lattice fuel assembly design study

As long-life core concepts with passive safety features including natural circulation decay heat removal options are considered for a candidate of the next generation nuclear reactors, tight lattice fuel pin subassembly design with the triangular pin array configuration (Fig. 4) attracts attention of thermal hydraulics communities. This configuration is often used in advanced nuclear reactors including high conversion light water reactors and sodium-cooled fast reactors. With lower pin pitch to diameter ratio P/D, i.e., less coolant volume fraction in the core, less neutron moderation assures harder neutron energy spectrum leading to more neutron population in the core and to a possibility of higher conversion of ^{238}U to ^{239}Pu. For example, in the Japanese prototype sodium cooled fast breeder reactor Monju with 280 MW electricity output JAEA website [10], 169 wire-wrapped fuel pins are encased in a hexagonal duct tube. P/D is 1.21 with P = 7.9 mm and D = 6.5 mm, which is much tighter than 1.3~1.4 of conventional light water reactors. Sodium flows in this core fuel subassembly at about 6.8 m/s (Re ~ 7.5×10^4) on an average. Also an inherently safe, small sodium-cooled fast reactor, 4S (Super-Safe, Small and Simple; see reference [11] with 10MW electricity output, employs even tighter P/D = 1.08 with P = 15.1 mm and D = 14.0 mm with wire spacers. Under the normal operating condition, the flow is driven by EM pump and Reynolds number in the core region is about 15,000, corresponding to the average flow velocity 2 m/s in fuel subassemblies of the average q'=90 W/cm.

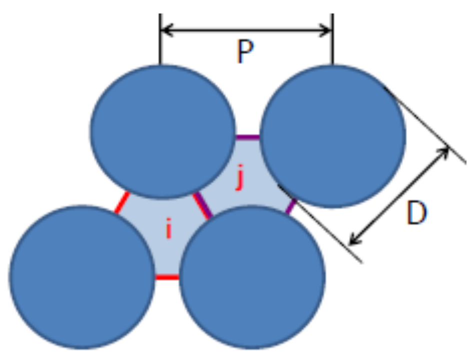


Fig. 4 Triangular pin array configuration
P is the pin pitch and D the pin diameter; i and j are the subchannel number

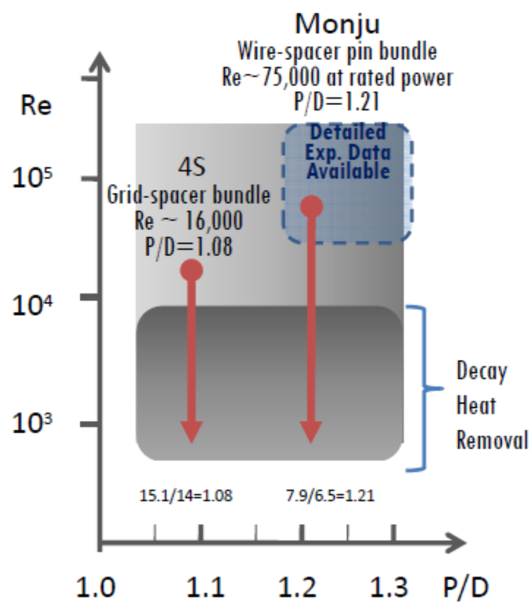


Fig. 5 Operational points of Monju and 4S fast reactors

Flows in fuel pin subassemblies (or in any inhomogeneous channel geometry) are characterized, in general, by divergence cross flow and turbulent mixing. In the sodium-cooled fast reactor development, a large thermal hydraulics database has been constructed for fuel subassemblies of relatively high P/D and under highly turbulent flow conditions (Fig. 5). In the case of tight lattice bundles where coolant flows at relatively low mass flow rates, however, very few experiments have been carried out. Therefore, no experimental data are available with respect to detailed velocity and temperature distributions inside fuel subassemblies or in the subchannel geometry configuration under low Re turbulent flow conditions. There, the flows of interest are characterized by unsteady and irregular flows which give something of the appearance of randomness; strong vorticity; stirring and diffusion of heat; dissipation of energy by momentum exchange. As the decay heat further decreases, natural circulation mass flow rate would approach an extremely low range. Without much experimental information, it is a big challenge to apply CFD to these flows and obtain any physical insights out of it.

In the fully developed turbulent flows in a fuel pin sub-assembly without spacer effects, there are a couple of interesting phenomena that were hardly captured in the past by the Reynolds-Averaged Navier-Stokes (RANS) equations approach with isotropic k-ε turbulence models but found theoretically or experimentally. They include i) turbulence-driven secondary flows in subchannels, ii) local transition between laminar and turbulent flows near a narrow gap between two adjacent fuel pins in particular in the case of tight lattice fuel pin subassemblies, and iii) the global pulsations leading to the coolant mixing between the two subchannels, all these phenomena being connected to the anisotropy of turbulence. It is noted that the secondary flow effects Also these particular phenomena have been considered to be less dominant for fuel subassemblies of high P/D and at high Re because the turbulence would be more isotropic. However, as P/D and/or Re is reduced, these phenomena become conspicuous but the mechanisms have not been fully understood.

For flows in a subchannel of a triangular pin array configuration with a fuel pin pitch to diameter ratio P/D=1.2, it has been revealed that calculated profiles of the main flow velocity, wall shear stress, Reynolds stress, …, etc. vary with decreasing Re, showing strong dependency on the angular position along the pin surface, while they show relatively flat distribution and become almost independent of Re in the highly turbulent flow regime where turbulence becomes more isotropic. For the flows in tight pin array configurations (P/D=1.1~1.05), flow profiles, etc. are a result of more anisotropic turbulence structure, significantly influenced by the pin wall in the narrower gap region.

It is found that the flow is fully turbulent at almost everywhere in the subchannel at Re~24,000 for a fuel subassembly with P/D=1.2. It has been shown numerically that a local laminar transition starts to develop in the gap region as Re decreases (< 20,000). This local laminarization is enhanced as P/D decreases and, as Re increased from ~3,000, the fully transition to turbulent flow will be delayed. These local laminar-turbulent transition phenomena have been also investigated by LES as well as by DNS for turbulent flows in eccentric annuli channels [12]. These results are considered to indicate similar phenomena taking place in the subchannel flows in a fuel subassembly. Detailed discussions are found in references [13-14].

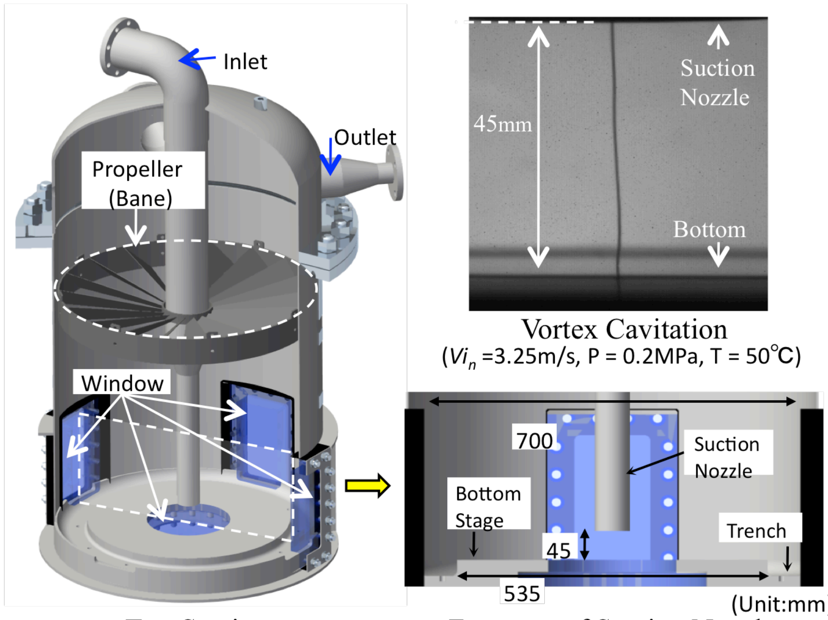
4.2 JSFR Compact Reactor Vessel

4.2.1 Vortex Cavitation

The coolant velocity in the hot leg (H/L) reaches 9.2 m/s in JSFR. This high velocity can generate strong vortices at H/L intakes [15]. Meanwhile, cavitation can occur due to pressure drop at the center of sub-surface vortex. This vortex cavitation is a significant hydraulic issue to be cared for the structural integrity of JSFR.

The cavitation due to sub-surface vortex (vortex cavitation) has been well researched for the pump sumps in the turbo machinery field. Standards for the model tests were provided by American National Standard Institute [16] and also Turbomachinery Society of Japan [17]. JAEA has performed a 1/10 scaled model test using water to evaluate the vortex cavitation based on the TSJ Standard. However, the coolant in JSFR is sodium, which has 1/3 smaller kinetic viscosity (n) than water, while those standards suppose to use the same working fluid as in the real plant. The decrease of n may influence the vortex structure. Thus, it is required to establish an evaluation method for the sodium system.

A fundamental experiment [18] was performed for vortex cavitation in Oarai research and development center of JAEA. The experiment was carried out in a simple cylindrical tank geometry using water as shown in Fig. 6. In order to see the influence of fluid viscosity, water temperature was varied from 10 to 80 °C. Occurrences of vortex cavitation were detected using a digital CMOS camera and image analysis under several different viscosity conditions.



Test Section Zoom-up of Suction Nozzle

Fig. 6 Test Section for Vortex Cavitation

The occurrences of vortex cavitation were evaluated using the cavitation factor σ and a yield fraction (Y.F.) of cavitation in time domain. Following findings were obtained.

The influence of ν was obvious under the condition of larger ν (>8.0x10-7). However, the occurrence of vortex cavitation was expressed by a nearly constant s condition under the relatively high inlet velocity, or small ν (<5.5x10-7) conditions. New non-dimensional parameter was proposed as an index to evaluate the occurrence of vortex cavitation.

4.3 JSFR Two-loop cooling system

As mentioned in the section 2.3, flow-induced vibration (FIV) in the primary loop hot leg piping is concerned from a view point of structural integrity. The pressure fluctuation in the elbow is the sources of FIV excitation force. In general, the pressure fluctuation and the velocity fluctuation are closely related. Thus, the information of the flow field and velocity fluctuation under full power condition of very high Reynolds number, Re = 4.2x107, is needed in order to grasp the mechanism of FIV. However it is difficult to conduct an experiment at such the high Reynolds number. Consequently, prediction of the flow structure is significant up to such a high Reynolds number, Re = 4.2x107 based on available experimental techniques and numerical simulations.

Thus, water experiments using the 1/8 scaled elbows (see Fig. 7) were conducted in order to investigate the mechanism of flow fluctuation induced by interaction between flow separation and secondary flow at the elbow [19]. The unsteady velocity fields were measured using a high-speed Particle Image Velocimetry (PIV) and the pressure fluctuation on the wall was measured using fiber optic pressure sensors in the Re range from 1.8x105 to 5.4x105. The elbow curvature ratio was chosen as an experimental parameter in order to investigate the influence of separation region on the velocity fluctuation in the elbow. The experiments were conducted using the two types elbows with different curvature ratio, r/D = 1.0 (so-called 'short-elbow') and r/D = 1.5 (so-called 'long-elbow').

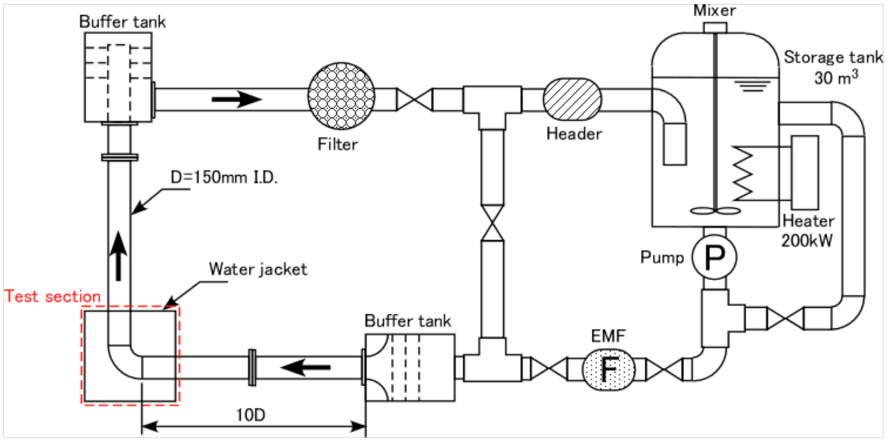


Fig. 7 Test Loop for Fluctuated Flow in Elbowa

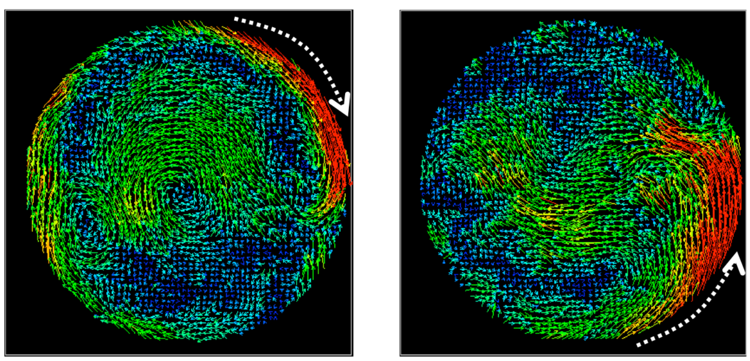


Fig. 8 Instantaneous Velocity Field at Elbow Outlet

Following insights were obtained from the experiments. The elbow curvature influenced the flow separation at the elbow outlet. The separation region was generated constantly only in the short-elbow case. The turbulent intensity near the inside wall downstream of the reattachment point was higher than in other area. Influences of reattachment of the separation flow and circumferential secondary flow were discussed for this high velocity fluctuation. Example of secondary flows at the elbow outlet is shown in Fig. 8.

A prominent power spectrum density of pressure fluctuation at St = 0.56 was found downstream of the short-elbow outlet. This pressure fluctuation was related to relatively larger motions of the flow separation region.

4.4 JSFR Natural circulation decay heat removal

The natural circulation is a significant issue on passive features of a sodium cooled fast reactor. Fully natural circulation system is adopted in a decay heat removal system (DHRS) of JSFR. The JSFR has two loops of the main heat transport system in order to reduce number of components and the construction cost. The DHRS of JSFR consists of two units of PRACS (primary reactor auxiliary cooling system), which has a heat exchanger in a primary-side inlet plenum of IHX in each loop and further one unit of DRACS (direct reactor auxiliary cooling system), which has a dipped heat exchanger in the reactor vessel. The decay heat after a reactor scram is removed solely by natural circulations in the main loops and DHRS including air flow paths of the air coolers. Such natural circulation DHRS is one of key issues of the JSFR development program. Start-up transient of the natural circulation is of importance for the decay heat removal. Straight tube type heat exchanger is adopted in the PRACS of JSFR. The heat transfer characteristics under natural circulation conditions are also significant issue of the DHRS design.

Sodium experiments [20] were carried out to study the heat transfer characteristics and transient behavior of natural circulation DHRS of JSFR. A partial model of the straight tube type PRACS heat exchanger was set in a sodium test loop named PLANDTL-DHX, which consists of a core simulator, a reactor upper plenum, the primary loop, IHX, the secondary loop, and a loop of DHRS. Schematic of the test loop is shown in Fig. 9. The straight tube type heat exchanger of PRACS was installed in the primary side inlet plenum of IHX. The heat exchanger tubes were set at one-side and larger flow area opened beside the tubes in the plenum to simulate the reactor geometry. A transient experiment was also carried out from forced circulation in the primary loop and standby operation of PRACS to the natural circulations in all systems.

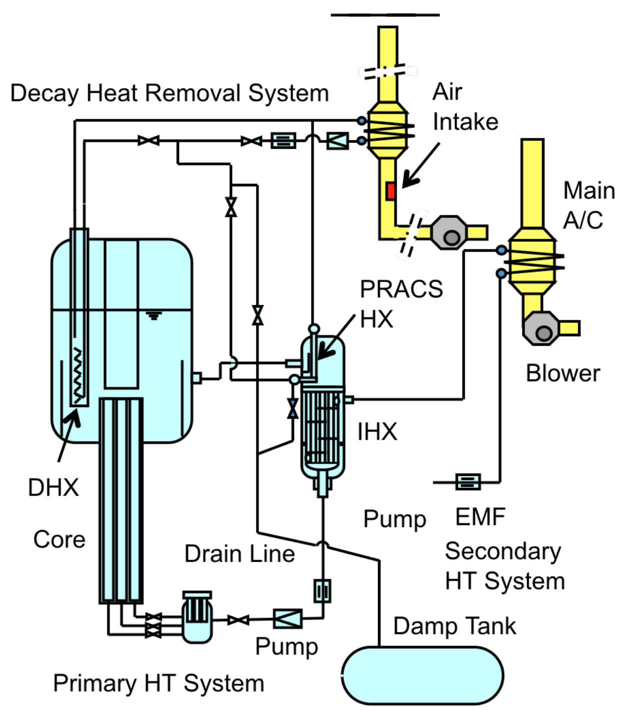


Fig. 9 Sodium Test Loop for Natural Circulation

Following results were obtained from the steady state and transient experiments. The heat transfer coefficient on the shell side of PRACS was estimated and in good agreement with a design correlation. The start-up of natural circulations was smooth in the air duct of the air cooler and in the primary loop. An opening procedure of the air cooler dampers is proposed to mitigate the thermal shock due to the difference of time delays of natural circulation start-up in the air duct and the PRACS sodium-loop.

4.5 JSFR CDA evaluation

4.5.1 Technical Database in the Unprotected Events for Level 2 PSA

A Level 2 probabilistic safety assessment (Level 2 PSA) including core damage and containment failure is indispensable to the comprehensive safety assessment of sodium-cooled fast reactors (SFRs). The core damage accidents in SFRs can be roughly divided into two categories: anticipated transient without scram (ATWS) and loss of heat removal system (LOHRS) events. Historically, the core damage accident in the ATWS events has been called as a core disruptive accident (CDA), where an energetic recriticality has been principally concerned.

In the transition phase during CDA, core material melting area is enlarged to a whole-core scale gradually following the fuel melting and dispersion within a fuel assembly by a power transient occurred in the initiating phase. Potential reactivity increase is caused by the compaction of molten fuel driven by gravity or pressure transient in the whole core scale, whereas reactivity decrease would likely occur by fuel discharge out of the core. To simulate such phenomena, an advanced computer code, SIMMER-III, has been developed in JAEA [5].

Toward commercialization of SFR, its safety features should be ameliorated by introducing adequate measures to mitigate the consequence of postulated CDA. An enhanced safety design aiming at the elimination of severe power excursion during CDA, which is called "recriticality-free concept", is adopted in JSFR. For this safety design, an inner duct is introduced into a fuel assembly (see Fig. 10) to enhance molten fuel discharge from disrupted core in the transition phase [6]. Therefore, a heading of the fuel discharge through the inner duct must be added to the conventional event tree in the transition phase.

The introduction of the recriticality-free concept allows us to likely skip the energetic evaluation during the post-disassembly expansion (PDE) phase in the licensing process. In the PSA, however, the PDE phase should be evaluated because the PSA should encompass all spectrums of event sequences.

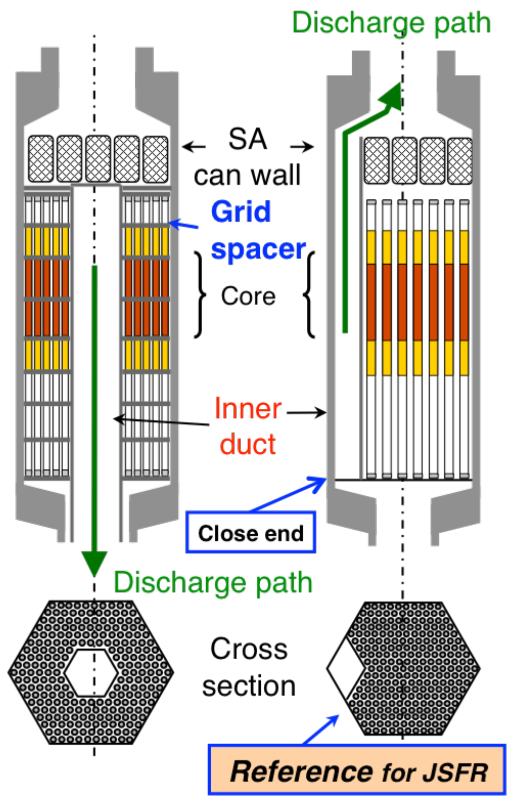


Fig. 10 Schematic of Inner Ducts in Subassembly

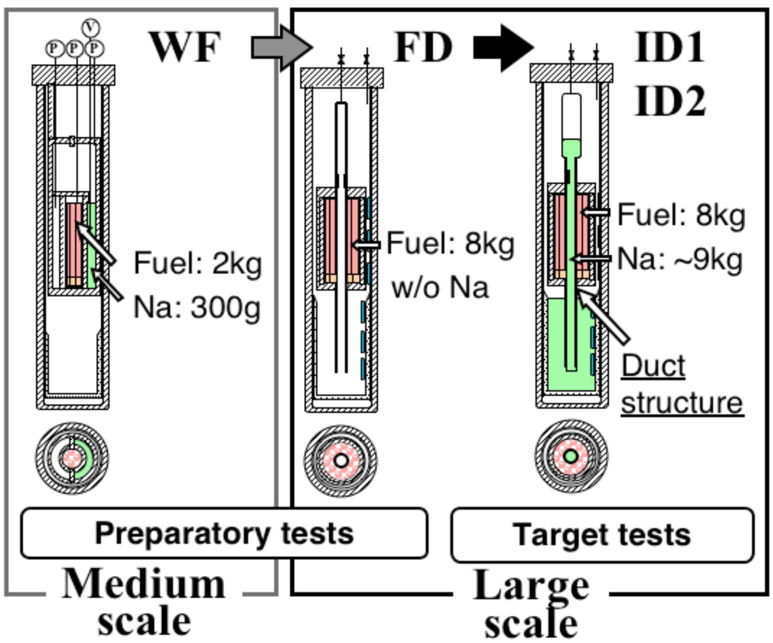
For the Level-2 PSA in SFRs, the technical database was developed for contributing to the quantification of branch probabilities in event sequences, focusing on the transition and PDE phases in the ULOF accident [21]. To identify the dominant factors, the parametric analyses were performed using the SIMMER-III code.

The dominant factor was the fuel discharge through the inner duct. Therefore, this phenomenon was added as one of headings of event tree in the transition phase. The experimental databases on the fuel discharge behavior and its driving force formation were summarized from mainly the EAGLE experiments. In the PDE phase, the parametric analyses showed that the energy dissipation effect of internal structures was a key issue.

4.5.2 Heat Transfer between Molten Core Fuel and Steel Wall Structure

Based on the fact that the sodium-cooled fast reactor (SFR) core is not in the highest reactivity configuration, the issue of severe power excursion driven by reactivity insertion during CDA has been addressed as one of the key safety issues. Such severe power excursion events cannot be ruled out if core melting progresses significantly and forms whole-core-size pool, since molten-fuel motion leading to compaction might cause severe reactivity insertion. However, in case of core melting, it is highly probable that molten core materials would discharge from the core region through some paths. In order to achieve the early fuel discharge from the core region and make it possible to prevent the large-scale molten-core-pool formation, which leads to severe power excursion events, fuel sub-assembly with an internal duct structure (FAIDUS, see Fig. 10) was adopted in JSFR design [22].

In the EAGLE-1 (Experimental Acquisition of Generalized Logic to Eliminate recriticalities) program [23], a series of in-pile and out-of-pile tests was performed utilizing the research facilities in the National Nuclear Center of the Republic of Kazakhstan as shown in Fig. 11. The purpose of this program was to obtain experimental evidences to show effectiveness of FAIDUS. The molten fuel discharge through the inner duct commences with its failure under the heat-transfer from the molten core materials, and thus it is quite important to simulate magnitude of heat transfer in CDA analyses in order to evaluate the timing of wall failure appropriately [24].



TEST NAME	MASS OF TEST FUEL	TEST OBJECTIVE
WF	~2 kg (12-pin bundle)	***Wall failure*** by the heat from molten fuel/ steel mixture pool.
FD	~8 kg (75-pin bundle)	Overall ***fuel discharge*** behavior through the inner duct structure ***without Na***.
ID1, ID2	~8 kg (75-pin bundle)	Overall ***fuel discharge*** behavior through a duct structure ***with ~9 kg of Na*** in the duct and lower plenum. In ID2, energy release in the test fuel is smaller than that in ID1.

Fig. 11 Schematic of Test Sections in EAGLE-1

Therefore, the heat-transfer characteristic from the molten core materials to the outer surface of the inner duct structure was studied through analyses of the experimental data which were obtained in the EAGLE-1 program [25]. Through comparison with this experimental information, applicability of the FBR safety analysis code SIMMER-III to the heat-transfer behavior was also evaluated.

Through the present study, it was understood that the pool-to-duct heat flux was so high in all the EAGLE-1 in-pile tests that the duct wall failed without coolant boiling in its behind. Under this high heat transfer condition, temperature increase of structures facing the molten-core materials is dominated by the thermal inertia of the structure. This knowledge is supporting the effectiveness of the inner duct design, since the thin duct will fail early and allow the early fuel discharge before the failure of thick wrapper tube which prevents the large-scale molten-core-pool formation. Influence of steel in the molten mixture was also evaluated for this high heat transfer.

5. Summary

Low Re single-phase flows in tight lattice fuel subassemblies exhibit quite interesting phenomena including the local laminarization that creates higher temperature spot on the cladding and the global flow pulsation that enhances coolant mixing between two adjacent subchannels reducing the hot spot temperature. Therefore fully understanding the phenomena taking place in the tight lattice fuel assemblies core of low power density, i.e., the 4S reactor core is necessary. There, the full use of simulation technology will play a key role with most updated CFD approaches based on DNS and LES.

Current status of design study of JSFR is summarized and several R&D activities related to the design are reviewed. CFD code is of importance to evaluate thermal hydraulics in a reactor component and piping. Verification using water and sodium experiments are undergoing. However, specific modeling is also required for the vortex cavitation due to the sharp reduction of pressure around the vortex center. The vortex structure depending fluid property, e.g., viscosity, is significant and investigated. Particle Image Velocimetry is strong tool to measure transient velocity fields of fluctuating or intermittent phenomena, e.g., flow separation at the elbow outlet and vortex cavitation. A sodium experiment of natural circulation started for sodium-sodium heat exchanger of DHRS and also start-up transients of DHRS through the primary system to natural draft in the air cooler. In-pile experiments were carried out for evaluation of CDA process and recriticality free concept. As for the level-2 PSA of JSFR, technical database is developed for contributing to the quantification of branch probabilities in event sequences, focusing on the transition and PDE phases in the ULOF accident based on several experiments and parametric analyses. Especially, it was identified that the fuel discharge behavior through the inner duct of FAIDUS was governed by the inner duct failure timing and the mobility of molten core materials.

Thus, the heat transfer characteristics from molten core material into the inner-duct wall were investigated based on the EAGLE-1 in-pile experiments. Very high heat flux between the molten pool and the inner duct wall was confirmed in the tests providing an evidence of the early fuel discharge using FAIDUS.

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