

Progress of Design and related Researches of Sodium-cooled Fast Reactor in Japan

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Abstract. Development of a sodium-cooled fast reactor has been implemented in Japan from the viewpoint of severe accident countermeasures in order to strengthen safety of a fast reactor since the Great East Japan Earthquake. This paper describes the progress of design study and research and development related to safety enhancement and the severe accident countermeasures. For the purpose of strengthening of decay heat removal function, several researches have been carried out on the decay heat removal in a core disruptive accident (CDA), diversity and applicability of decay heat removal systems, and thermal hydraulic evaluation methods. In order to elucidate the behavior of molten fuel during CDA, some in-pile and out-of-pile tests has been performed by international collaboration including basic experiments. Core design was also improved from the viewpoint of preventing the occurrence of severe accident.

Key Words: Severe Accident Countermeasures, Decay Heat Removal, Core Disruptive Accident, Self Actuated Shutdown System

1. Introduction

Fast reactor cycle technology has great importance from the standpoints of energy security, reduction of environmental load, and also the global warming. The Fast Reactor Cycle Technology Development (FaCT) project had been carried out in Japan to develop a demonstration reactor and subsequent commercial reactor of sodium-cooled fast reactor (SFR) until 2011. This project was aborted owing to change of Japanese energy policy and nuclear energy policy due to accidents in the TEPCO's Fukushima Dai-ichi Nuclear Power Plants, which occurred subsequent to the Great East Japan Earthquake. Currently, research and development (R&D) on the fast reactor cycle technology are conducted in accordance with the 4th Strategic Energy Plan, which have been approved by the Cabinet in April 2014. The main targets are to strengthen safety of a fast reactor system and to reduce volume and toxicity of high-level radioactive waste.

Following the above situation, several R&Ds on especially severe accident countermeasures are planned and carried out in order to strengthen safety of an SFR. In the Generation-IV International Forum (GIF), the development of international standards of safety design requirements for Generation-IV SFRs started in 2010. Safety Design Criteria (SDC) was approved by the GIF in May 2013 and is currently being reviewed by international organizations and regulatory bodies of countries which are developing SFRs. Safety Design Guidelines (SDG), which shows recommendations to apply the SDC to the actual design of an SFR, is also being developed in the GIF. Japan contributed to these activities as a key member. Our R&D activities on the safety enhancement of an SFR take into account concepts of these SDC and SDG developments.

In this paper, the followings are described as the progress of design study and related researches: (1) the R&D on the decay heat removal system during and following the core disruptive accident (CDA), the applicability and diversity of heat removal systems, and the evaluation methods for thermal hydraulics for the purpose of strengthening of decay heat removal function, (2) the in-pile and out-of-pile tests by international collaboration, and the basic experiments in order to elucidate the behaviour of molten fuel during CDA, and (3) the improvement of core design from the viewpoint of preventing the occurrence of severe accident.

2. Overview of Safety Approaches for SFR

In order to strengthen the safety of an SFR, several R&D programs have been carried out based on the following safety approaches which future SFRs are expected to take [1].

Regarding reactor shutdown system, at least two independent rapid and active reactor shutdown systems with diversity for design basis accidents (DBAs), and additional passive shutdown function for Design Extension Conditions (DECs) should be provided.

Concerning Decay Heat Removal System (DHRS), ensuring diversity in systems is essential for improving the overall reliability. An SFR should proactively utilize its natural circulation capability to an ultimate heat sink (atmosphere), since this can significantly contribute to improving the reliability of the heat removal capability, even under long-term loss of power supplies.

As for containment, robust and comprehensive approach for maintaining/enhancing integrity of containment vessel is recommended in future SFRs. For instance, significant pressure/temperature increases by sodium fire should be prevented by applying e.g. guard vessel and pipes and/or inert gas filling, and significant mechanical energy release via severe re-criticality in the course of the CDA should be prevented by prevention and mitigation built-in safety design measures. According to the lesson learned from the accidents in the TEPCO's Fukushima Dai-ichi Nuclear Power Plants, it is strongly recommended that inherent features and passive mechanisms should be incorporated into the prevention and mitigation measures as well as practical accident management measures.

3. Progress of Researches

3.1. Researches on Decay Heat Removal Function

R&D on the decay heat removal (DHR) including situations in CDAs are being carried out in JAEA for clarifying global and local thermal-hydraulic phenomena in SFRs, evaluating the applicability and diversity of decay heat removal systems, and developing safety assessment methods. When a CDA occurs, the damaged fuels and debris may be distributed in the core, on a core catcher and bottom of the upper plenum in the reactor vessel. Robust and diverse measures of DHR are of importance for the in-vessel retention concept for CDAs. In-vessel natural convection induced by cooling systems has a significant role for DHR in both of pool and loop type SFRs. Experimental study using three kinds of experimental facilities named PHEASANT, PLANDTL-II and AtheNa-RV has started to investigate thermal-hydraulic phenomena under natural circulation DHR conditions including the situations of CDA in SFRs and to evaluate the performance of DHRSs [2].

PHEASANT (PHenomena clarification Experimental Apparatus for Severe AccideNT) is a water experimental facility based on 1/10 scale model of a 10m-class reactor vessel of advanced loop-type or pool type reactor [3]. PHEASANT consists of the reactor vessel and

three cooling loops with heat exchangers to simulate several types of DHR systems as shown in Fig. 1. Two of them are direct heat exchangers (DHXs) installed in the upper (hot) plenum or penetrated into the lower (cold) plenum. Each of them can simulate a dipped type DHX in the upper plenum, a penetrated type DHX connecting the upper and lower plena or a cold-pool DHX which has both of inlet and outlet windows in the lower plenum, depending on the position of primary side inlet and outlet windows. Flow path of IHX in a pool-type reactor can be simulated using the penetrated type DHX. The last one simulates an ex-vessel cooling system. The cooled flow along the reactor vessel wall by the ex-vessel cooling system can be also simulated by the copper tube heat exchanger set near the acrylic wall of the reactor vessel. These three loops and heat exchangers can be used solely or as combinations to investigate separate effect of each DHR system and also integral effects.

PHEASANT has several electric heaters, which simulate the heat from a damaged core and accumulated debris due to core melting. The capacities of micro-heaters are 30 kW, 5 kW and 25 kW, in the core region, on the bottom of the upper plenum and on the core catcher respectively. Two baffle plates at the top and bottom of the core can simulate increase of flow resistance due to the damage of core (blockage) depending core regions.

Figure 2 shows an example of the visualizations related to the behavior of cold flow from the dipped-type DHX at start-up of DHX operation. The flow behavior in the upper plenum and lower plenum is visualized by injecting dye ink from the primary side of the DHX. It is observed that the path of cold flow penetrates from upper plenum to lower plenum through the core. On the other hand, the temperature field in the vessel is measured by the multipoint thermocouple trees, which have 100 thermocouples in the upper plenum and 61 thermocouples in the lower plenum. These multipoint thermocouple trees, which can rotate in circumferential direction and move in vertical direction, enable to obtain three-dimensional time-averaged temperature fields. Several preliminary experiments have started in this year.

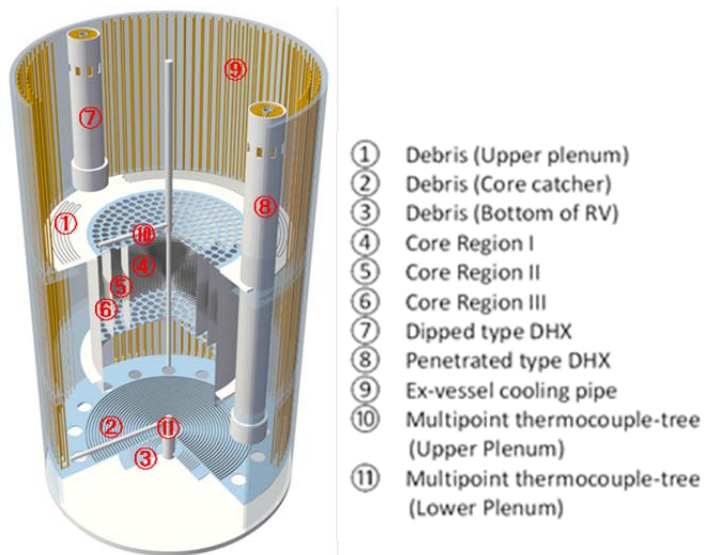


FIG. 1. Schematic diagram of water experimental apparatus, PHEASANT.

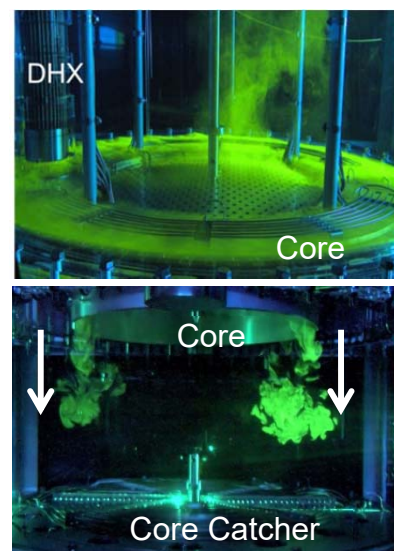


FIG. 2. Cold Flow from DHX in upper and lower plenum.

PLANDTL-II (PLANt Dynamics Test Loop) is a middle-scale sodium test facility which is remodeled for a reactor vessel of SFR focusing on thermal-hydraulic behaviors in the core and upper plenum regions, especially inter wrapper flow and radial heat transfer in the core and core-plenum thermal-hydraulic interaction. PLANDTL-II consists of the core region including fuel assemblies, neutron shielding assemblies and a control rod channel, the upper

plenum including the upper core structure and the DHX, the primary and secondary heat transport systems, the IHX, and related components in the heat transport system as shown in Fig. 3. The reactor core is modeled by 30 heated channels simulating fuel assemblies, 24 unheated channels simulating neutron shielding assemblies and one unheated channel simulating a control rod channel at the center of the core region. Figure 4 shows the schematic image of PLANDTL-II test section. The heater power in the core region is 1 MW and the regional flow rate through the core is around 0.9 m³/min. Additionally, each outlet of channel can be plugged independently to simulate damaged core conditions though the inter wrapper flow through the gap of the channels remains in the core region. PLANDTL-II facility is now under construction and will be completed at the beginning of 2017.

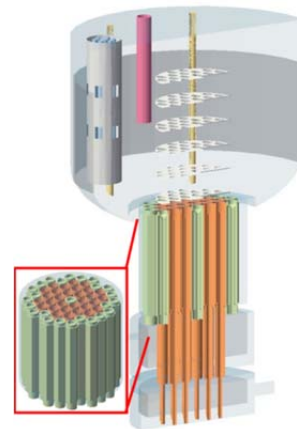
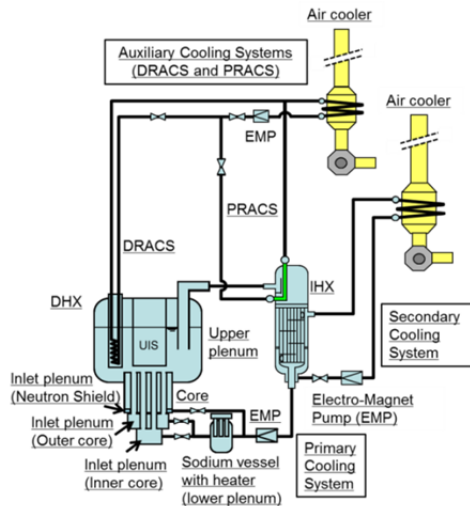


FIG. 3. Schematics of PLANDTL-II.

FIG. 4. Schematic of PLANDTL-II test section.

AtheNa-RV is a large-scale sodium test facility to clarify thermal hydraulic phenomena in a reactor vessel of pool-type or loop-type SFR. Its design study is in progress for comprehensive DHR tests under intact/damaged core conditions. Figure 5 is an example of the reactor vessel conceptual designs. Electric heaters in the core and debris regions can simulate several different heating conditions of the core (damaged core) and the debris on the core catcher. A multi-function DHX is installed which simulates dipped-type DHXs in the hot and cold pools and penetration-type DHXs of the DHRs. Ex-vessel cooling system can be also installed on the outside of safety vessel wall. The temperature field in the vessel is measured with hundreds of thermocouples and the flow rate in the core region is measured with eddy current flow meters. In the experiments, cooling performance of each DHRS will be evaluated and demonstrated, and complicated thermal hydraulic behaviors under various operating conditions of DHRs will be clarified including the interaction effect on cooling performance when various DHRs are operated simultaneously.

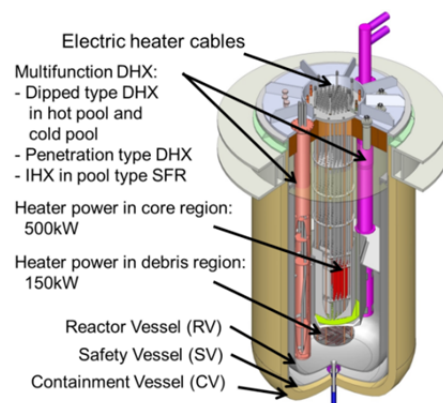


FIG. 5. Conceptual Design of AtheNa-RV (Tentative)

The experimental results obtained from above three test facilities will also contribute to model improvement and validation of various safety analysis codes.

3.2. Researches on Countermeasure against Core Disruptive Accident

In order to elucidate the behavior of molten core, two types of experimental studies have been conducted. The first one is demonstrative experiments using molten-fuel or oxide material. The second one is basic experiments addressing key phenomena in detail using a simulant.

The demonstrative experiments have been carried out under a joint-research program with National Nuclear Center of the Republic of Kazakhstan (NNC/RK) named EAGLE (Experimental Acquisition of Generalized Logic to Eliminate re-criticalities) program over more than 15 years, in which both out-of-pile and in-pile test facilities of NNC/RK are used. The past periods of the EAGLE program called EAGLE-1 and EAGLE-2 provided unique experimental database for molten-fuel discharge [4]-[6], and evaluations using these database showed that the countermeasure against CDA, in which a dedicated duct is installed in the fuel subassembly (i.e., the fuel subassembly with an inner-duct structure (FAIDUS) [7]), would be effective to eliminate power excursions during the early phase of CDA [8] in which the core melting progresses rapidly driven by insertion of positive reactivity.

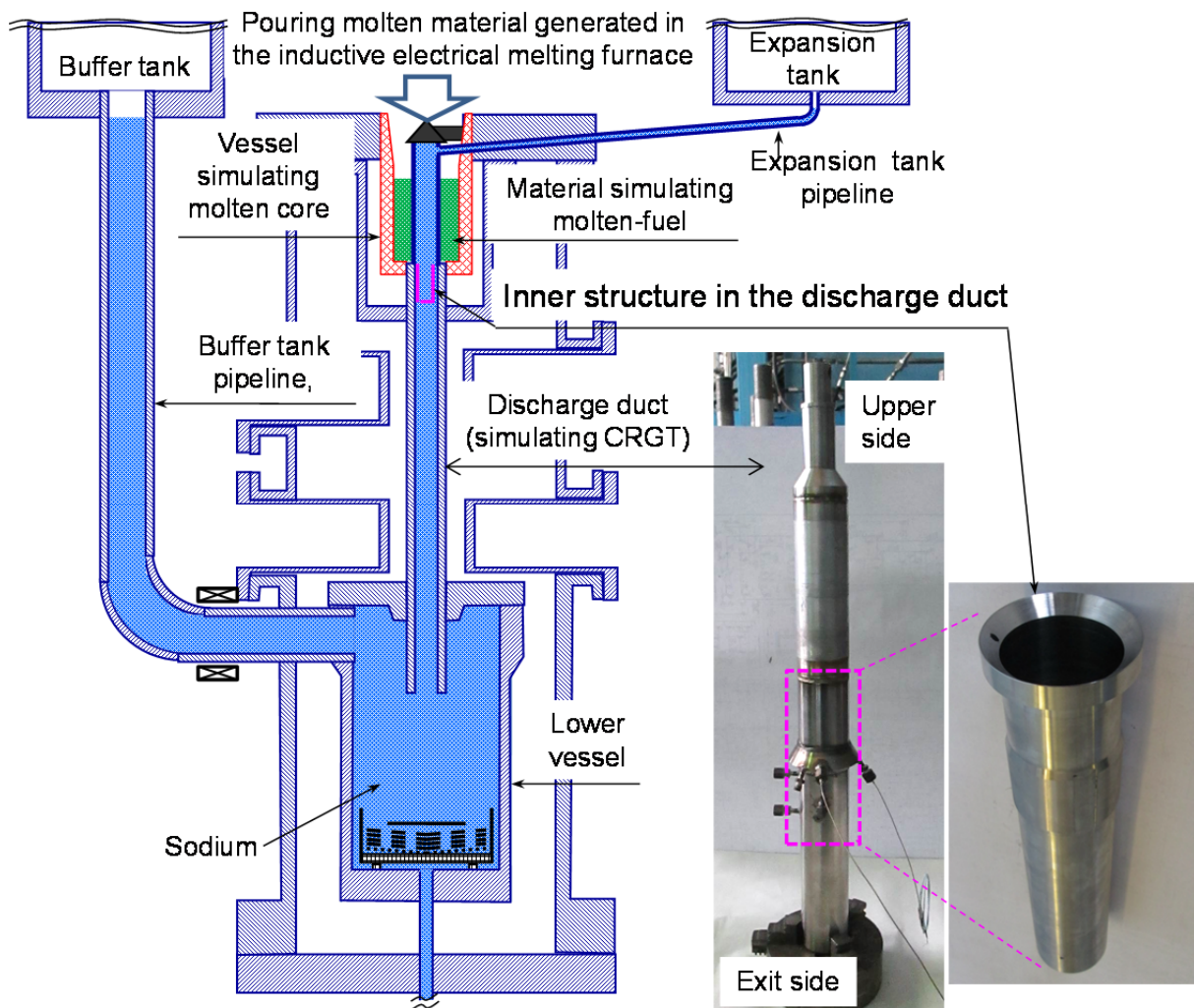


FIG. 6. Experimental devices for out-of-pile series experiments.

Based on the knowledge obtained through past studies, current major interests are focused on molten core behavior in the late phase of CDA, in which the degraded core material could be melted by its own decay heat and thus gradually relocated into the coolant plenums located below the core region mainly through control rod guide tubes (CRGTs). Based on above accident sequence, the EAGLE-3 is dedicated to clarifying the following three issues: The

first one is molten fuel discharged through CRGT, the second one is coolability of discharged fuel in the coolant plenum and the third one is coolability of remaining fuel in the degraded core [9]. In this program, in order to obtain experimental data to clarify the effects of various parameters, such as an inner structure of CRGT, fuel discharge diameter, etc. on the relocation and cooling behavior of core-materials, a series of experiments is conducted at first using the out-of-pile test facility of NNC/RK (Fig. 6). Then, in order to confirm the behavior of actual reactor materials, in-pile experiments are performed using the impulse graphite reactor (IGR) of NNC/RK. At present, a series of out-of-pile test is being carried out to optimize a design of CRGT for accelerating fuel discharge.

The basic experiments addressing key phenomena in the late phase of CDA, such as fragmentation of fuel discharged into the coolant plenum, formation of fuel debris bed, etc., are being conducted in Japan (JAEA and Universities [10]). Figure 7 shows an example of an experimental result which simulates fragmentation of discharged fuel in the coolant plenum [11]. In the present experiment, molten aluminum (Al) heated up at approximately 1723 K was used as a fuel simulant, and it was poured into a sodium pool through a nozzle. Fragmentation behavior of molten Al in the sodium pool was observed using X-ray imaging system which was composed of an X-ray generator, image amplifier and high-speed camera. X-ray images in Fig. 7 show that the liquid column of molten Al was intensively fragmented nearly simultaneously with a rapid expansion of sodium vapor in the vicinity of the column and penetration of the liquid column was limited due to the intensive fragmentation. At this moment, a series of experiment using molten-steel is being carried out to expand knowledge and database for the fragmentation in sodium. Knowledge obtained through these basic experiments and an evaluation methodology will be utilized to consider effective designs for containing the degraded core materials in the reactor vessel.

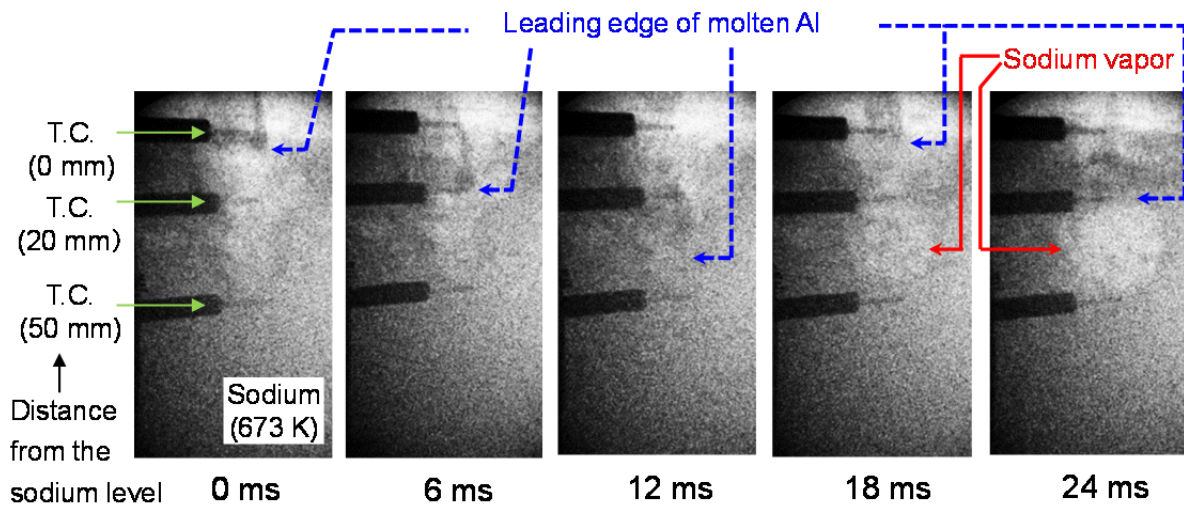
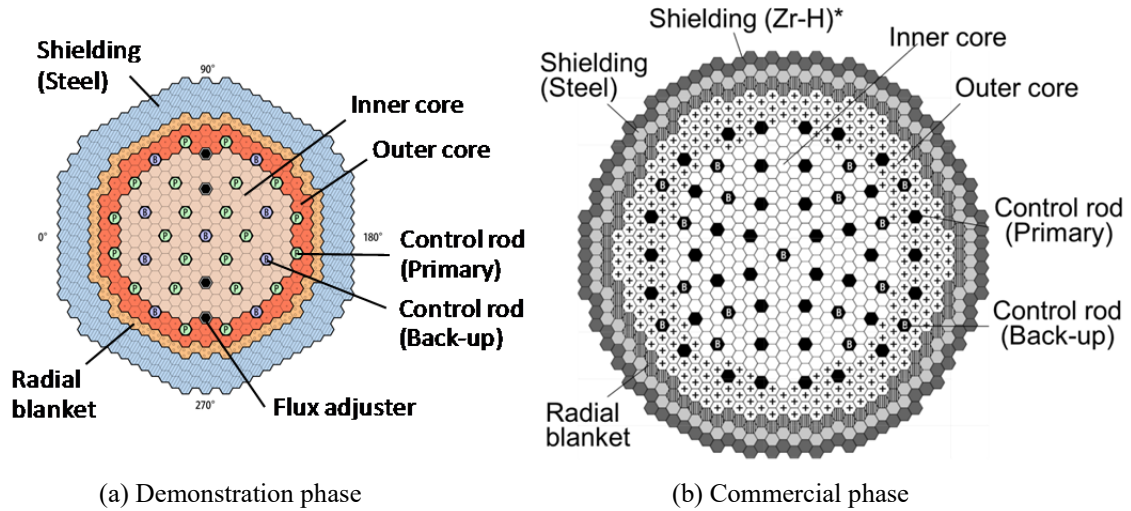


FIG. 7. Typical example of X-ray images for fragmentation in a sodium pool.

4. Progress of Design Study

A reference SFR core concept with mixed oxide fuel, called a “high-internal conversion” core, was developed in FaCT [12-15]. Figure 8 shows the core configuration, and Table 1 summarizes main specification. Power outputs in the demonstration and the commercial phases are 750 MWe and 1500 MWe, respectively. The core outlet and inlet temperatures are 550 °C and 395 °C, respectively. This core concept applies large-diameter fuel pins to increase the internal conversion ratio and, therefore, to reduce the amount of blanket to the greatest extent possible. It has the economical advantage of achieving high average

discharged burnup for the entire core (including blanket) and long operation cycle length, in addition to sufficient breeding capability with a small amount of blanket. To realize high burnup under high temperatures, oxide-dispersion-strengthened (ODS) steel is being developed for application as the cladding material [16].



* Under reviewing to avoid the potential risk of hydrogen release

FIG. 8. The Reference SFR Core Configurations.

TABLE I: MAIN SPECIFICATION OF THE REFERENCE SFR CORES.

Item	Demonstration Phase	Commercial Phase		
Power Output [MW _e]	750	1500		
Coolant Temperature (outlet / inlet) [°C]	550 / 395	550 / 395		
Breeding Ratio	1.1	1.03	1.1	1.2
Core Height [cm]	100	100	100	75
Axial Blanket Thickness (upper / lower) [cm]	20 / 25	15 / 20	20 / 20	40 / 50
Number of Fuel Subassembly (inner core / outer core / radial blanket)	184 / 94 / 66	288 / 274 / 0	288 / 274 / 96	288 / 274 / 96
Fuel Pin Diameter [mm]	10.4	10.4	10.4	9.3
Subassembly Pitch [mm]	206	206		
Core Diameter [m]	3.8	5.4		
Average Discharge Burnup [GWd/t]	148	150	145	149
	(83)*	(115)*	(89)*	(60)*
Operation Cycle Length [month]	18	26.3	26.3	21.0
Fuel Exchange Batch	6	4	4	4
Pu-Fissile Inventory [t/GW _e]	6.2	5.8	5.8	5.1
Average MA Content in Core Fuel [wt%]	≤3	≤3		

* Total average discharge burnup (including blanket)

In the safety design of fuel and core, three aspects shall be considered, i.e., 1) ensuring fuel integrity with sufficient margin in the normal operation, 2) ensuring core coolability against DBAs, and 3) core damage prevention and mitigation under DEC. The reactor core, which has inherent power control capability mainly due to Doppler effect, and active reactor shutdown systems, which consist of control rods, are designed not to exceed design limit for core coolability against DBAs. Self Actuated Shutdown System (SASS) is installed as a design measure under a DEC, in which a main and back-up reactor systems failure is assumed in case of anticipated operational occurrences (AOOs) (hereafter “Anticipated Transient Without Scram, ATWS”). Furthermore, unprotected transients, in which SASS is not actuated in ATWS, are assumed as a DEC for mitigation of core damage. Some design measures,

which include limitation of positive sodium void reactivity, installation of steel duct in each fuel assembly for molten fuel discharge in early phase of a CDA are taken to prevent prompt criticality.

SASS is a passive control rod insertion mechanism, which utilizes a physical property of magnetic material, i.e., Curie point temperature. The control rod is suspended by an electromagnet installed just above the core, in which a temperature sensing alloy is embedded. When the core outlet coolant temperature abnormally increases up to the Curie point, SASS loses its holding force. Hence, the control rod is detached and inserted by gravity. Points of this technology are 1) to ensure response time necessary to prevent core damage against ATWS, and 2) to ensure stable suspension of control rod during normal operation. Based on these basic requirements, quantitative design conditions such as holding force of the control rod and detach temperature were identified and mechanical design of SASS was materialized. Transient analysis to evaluate SASS response against ATWS was conducted and showed that core damage is prevented [17].

SASS is designed to respond to temperature increase of coolant transported from the outlet of 6 fuel assemblies, which surround a back-up control rod with SASS. Transient thermal hydraulic behavior in ATWS around the core outlet and heat transfer to the temperature sensing alloy of SASS are important factors. The flow and temperature fields around the SASS will change drastically in a transient of ATWS. Especially in loss of flow type ATWS (LOF-type ATWS), relatively cold sodium in the core peripheral region tends to flow into the core central region and pass through the SASS. In order to limit influence of such radial motion of coolant, a flow collector was installed, which covers the outlets of 6 surrounding fuel assemblies and guides the exiting flow toward the temperature sensing part of SASS. The severest case of ATWS from the point of view of core temperature is the LOF-type among LOF-type, TOP-type (transient over power type) and LOHS-type (loss of heat sink type). Partial power operation leads to more severe condition, since its longer response time due to lower initial core outlet temperature. Core damage can be prevented by SASS with the flow collector and decreasing the detach temperature of SASS to 650 degree C for the LOF-type ATWS even in a case of the partial power operation.

In the initiating phase of core damage sequences, positive sodium void reactivity due to coolant boiling could be a cause of power excursion. It was shown from past studies that prompt criticality during the initiating phase can be prevented by limiting sodium void worth of the total core less than 6 dollars, although it depends on the core and fuel design. Initiating phase analyses of ULOF (unprotected loss of flow) and UTOP (unprotected transient over power) were conducted using SAS4A for the demonstration phase core [18]. In a conventional calculation with SAS4A, lumped core geometry, in which several numbers of fuel assemblies are represented by one fuel channel, is used. This time, each fuel assembly was modeled as one channel, although core symmetry was taken into account. Control rod positions in axial direction were also taken into account for the axial power distribution.

For the ULOF analysis, sensitivity of core burn-up state, initial power level and sodium void worth uncertainty was investigated, while sensitivity of initial power level and reactivity insertion rate of control rod withdrawal was investigated for the UTOP. It was shown that prompt criticality is prevented in these calculation cases, where the detailed core channel model was effective.

5. Concluding Remarks

Research and developments are conducted in Japan from the viewpoint of severe accident countermeasures to strengthen safety of an SFR. For the purpose, the safety approaches for future SFRs are taken into account in the design study and the related researches.

Experimental studies using three kinds of experimental facilities named PHEASANT, PLANDTL-II and AtheNa-RV has started to investigate thermal-hydraulic phenomena under natural circulation DHR conditions in SFRs and to evaluate the performance of DHRs. PHEASANT is a water experimental facility based on 1/10 scale model of a 10m-class reactor vessel of SFR. PLANDTL-II is a middle-scale sodium test facility which is remodeled for a reactor vessel of SFR focusing on thermal-hydraulic behaviors in the core and upper plenum regions, especially inter wrapper flow and radial heat transfer in the core and core-plenum thermal-hydraulic interaction. AtheNa-RV is a large-scale sodium test facility to clarify thermal hydraulic phenomena in a reactor vessel of pool-type or loop-type SFR. Its design study is in progress for comprehensive DHR tests under intact/damaged core conditions.

The demonstrative experiments and the basic experiments have been carried out in order to elucidate the behavior of molten core in the late phase of CDA. The demonstrative experiments have been carried out by utilizing both out-of-pile and in-pile test facilities under a joint-research program with NNC/RK. At present, a series of out-of-pile test is being carried out to optimize a design of CRGT simulation for the molten fuel discharge. The basic experiments are also conducted; a series of experiment using molten-aluminum and also steel is being carried out to expand knowledge and database for fragmentation in sodium.

The quantitative design conditions of the self-actuated shutdown system (SASS) such as holding force of the control rod and detach temperature were identified and the mechanical design was materialized. The transient analysis to evaluate SASS response against ATWS was conducted and showed that core damage is prevented. The initiating phase analysis of ULOF and UTOP were also conducted using SAS4A with improved core modeling methodology for the demonstration phase core, and showed that prompt criticality is prevented in the calculation cases.

6. Acknowledgement

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