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Core and safety design for France–Japan common concept on sodium-cooled fast reactor

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Abstract. France (CEA and FRAMATOME) and Japan (JAEA, MFBR, and MHI) teams have carried out collaborative works to have common technical views regarding a sodium-cooled fast reactor (SFR) concept. This paper mainly describes the capabilities of ASTRID 600 to demonstrate SFR technologies of both countries. Japan has studied the feasibility of an enhanced high burnup low-void effect (CFV) core and fuel using oxide dispersion-strengthened steel cladding in ASTRID 600, considering Japan's target for core performance. The neutronics design of the core has satisfied most required design targets and conditions. Regarding passive shutdown capabilities, Japan team has performed a preliminary numerical analysis for ASTRID 600 using a complementary safety device (CSD), called a self-actuated shutdown system (SASS), one of the safest approaches in Japan. Japan's team used the SASS in place of the hydraulically suspended absorber rod, called RBH, one of the safest approaches of France, to investigate the potential of the SASS as a design measure common to the countries. The preliminary analysis has shown that the SASS can satisfy the countries' main requirements. This study has also revealed that the mitigation measures of ASTRID 600 against a severe accident are promising to achieve in-vessel retention for both countries.

1 Introduction

A France–Japan collaborative work started in 2014 for plant design and three R&D areas – severe accident, reactor technology, and fuel – for the Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) to contribute to future sodium-cooled fast reactor (SFR) development [1–3]. With great support from CEA and Framatome, the team of Japan (Japan Atomic Energy Agency (JAEA) partnered with Mitsubishi Heavy Industry (MHI) and Mitsubishi FBR systems (MFBR)) conducted design studies and evaluations for ASTRID 600, including the designs of active decay heat removal system [4], curie-point electro-magnetic shutdown devices [5], and seismic isolation systems [6]; fabricability studies of above core structure (ACS) and polar tables [7]; improvement of core catcher design [8]; plant thermal transient evaluation; design analyses of the main vessel and an inner vessel; and core design evaluation.

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Since 2017, the researchers of both countries have conducted studies to further deepen the cooperation and achieve a common view on SFR concepts that possibly form the basis for future collaborations and standardization for SFRs in the countries. Based on the initial ASTRID design (600 MWe), both teams examined ways of developing a feasible, common design concept acceptable to both countries (hereinafter the common SFR concept). To understand design requirements common to the two countries, they discussed Top Level Requirements of design and conducted technical studies of all plant systems.

To achieve in-vessel retention (IVR), the following need to be demonstrated: a high burnup core with oxide dispersion-strengthened steel (ODS) cladding, a self-actuated shutdown system (SASS) that serves as a complementary safety device (CSD), and severe accident mitigation measures. The core and safety technologies have been investigated using Japan's core concept [9–12]. To develop the common SFR concept, the teams need to identify target items to be examined and determine if

ASTRID 600 concept can demonstrate the core and safety technologies.

In this study, French and Japanese teams examine their requirements and approaches for the designs of core and fuel, the reactor shutdown system, and mitigation measures against a severe accident to have technological views in common. Then, the teams examine whether ASTRID 600 can demonstrate the core and safety technologies of Japan. Finally, R&D items common to both countries are identified to achieve the qualification data required in the two countries.

2 Core and fuel designs

The purposes of this study are the following three: to develop a view common to the countries on the designs of a core and fuel by comparing design requirements for the components, target performances, development issues, and qualifications for licensing; to evaluate the possibility of harmonizing specifications of the designs, and to find design conditions that meet the design requirements in both Japan and France.

2.1 Common views on design requirements

The study based on ASTRID 600 revealed that several key parameters such as electric power, thermal power, fuel types, coolant, and primary coolant flow can be harmonized. There are, however, some differences in requirements for target burnup, breeding ratio, and reactivity control methods. The target breeding ratio is nearly 1.0 in France, whereas it is 1.1 to 1.2 in Japan. For a core including all fissile and fertile regions, Japan's target average burnup using ODS cladding is about 80 GWd/t. Although the average burnup for the whole core of ASTRID 600 is about 80 GWd/t [13], the burnup would be less than 80 GWd/t when radial blanket fuels are added to achieve the breeding ratio target, namely, 1.1 to 1.2.

France uses innovative architecture for the control and shutdown rods, called RID architecture, as a reactivity control method [14]. The RID architecture is composed of two groups of diverse control rods, named RBC and RBD, both of which contribute to operation and rapid shutdown. Japan, on the other hand, uses a conventional reactivity control method composed of two groups of control rods. Of these groups, the primary rod group is used for the operation and rapid shutdown, whereas the secondary rod group is required only for the rapid shutdown.

Thus, the feasibility of an enhanced high burnup core using ODS cladding based on the design of ASTRID 600 should be assessed to find the design conditions that meet core and fuel design requirements in both Japan and France.

2.2 Feasibility of high burnup CFV core design

The reference core design for ASTRID 600 is CFV (French abbreviation for "Coeur à Faible effet de Vide sodium", meaning low void effect core)V4 core with an austenitic steel cladding [14,15]. The design study of an enhanced high burnup CFV core using ODS cladding was performed to examine its feasibility, focusing on the average discharge burnup for the whole core and the breeding ratio. Examples of the core performance targets based mainly on Japan's targets are a whole core average discharge burnup of 80 GWd/tHM or higher and a breeding ratio of 1.1, as shown in Table 1. Two types of fuel composition were considered: a Pu vector from MOX (U–Pu vector, LWR) reprocessing, and a vector accompanied with minor actinides (TRU vector, LWR) from the reprocessing (see Tab. 2).

Table 1. Core design target.

Item	Design target
Average discharged burnup (core/total)	150 GWd/t/>80 GWd/t
Breeding ratio	1.1
Operation cycle length	>13 months
Max. linear heat rating	<430 W/cm
Fast neutron fluence	$< 5.0 \times 10^{23}$ n/cm^2
Cladding mid-wall temperature	<700 °C
Cladding cumulative damage fraction (steady state)	<0.5
Shutdown margin (cold/hot)	0.4%Δk/kk′/0.0%Δk/kk′
Sodium void reactivity (total)	< approx. 0 $

Table 2. Fuel composition of enhanced high burnup CFV core.

Nuclide	U–Pu (%)	TRU (%)
^{235}U	0.8	0.8
^{236}U	0.6	0.6
^{238}U	98.6	98.6
U-total	100	100
^{238}Pu	2	1.7
^{239}Pu	54	46.7
^{240}Pu	28	23.6
^{241}Pu	7	2
^{242}Pu	8	6.7
^{237}Np	–	6.2
^{241}Am	1	11.5
^{243}Am	–	1.4
^{244}Cm	–	0.2
TRU-total	100	100

Table 3. Fuel sub-assembly (F/A) design – Main differences.

Item	CFV V4 core*	Enhanced High Burnup CFV core**
F/A gap (mm)	3.0	4.4
Wrapper tube		
Material	EM10	PNC-FMS
Width across flats (mm)	161.5	159.3
Thickness (mm)	3.6	4.0
Cladding		
Material	AIM1	ODS
External diameter (mm)	9.7	9.6
Internal diameter (mm)	8.7	8.34
Thickness (mm)	0.5	0.63

(*) Beck, 2016; Venard, 2017. (**) All the different conditions from CFV V4 core have been decided based on the Japanese demonstration core conditions.

The dimension of the fuel pins and fuel sub-assemblies (F/As) were modified to achieve the high burnup without affecting the F/A pitch (see Tab. 3). The scheme of the control rod operation considered in this study is the use of the RID architecture. To fulfill the core performance, the Japan team made several modifications to the core. As shown in Figure 1, the Japan team added twelve inner core fuel assemblies and six outer core assemblies, increased the fissile height of the outer core by 7.5 cm to achieve the linear heat rating requirement (<430 W/cm), and added three RBCs and three RBDs to fulfill the control and safety reactivity requirements of Japan. Moreover, the Japan team added one fertile row to achieve the target breeding ratio and increased the total circumscribed diameter of the core by 34 cm, while maintaining the thickness of radial neutron protection for the CFV V4 core [13].

Japan team evaluated the characteristics of the enhanced high burnup CFV core by power distribution calculation and reactivity coefficient calculation, as well as burnup calculation using JENDL-4.0, the seven energy groups, and 3D-diffusion theory. Japan team obtained the power distribution using pin-wise flux interpolation based on mesh-wise flux. The γ-heat of the wrapper tube and thermal axial expansion of fuel pellets were considered in the linear heat rating calculation. Based on 70 energy groups and 3D-diffusion calculation, the Japan team evaluated sodium void reactivity according to the exact perturbation theory.

For the core neutronics design, most of the required design targets and conditions have been satisfied as shown in Table 4. The TRU vector has more minor actinide (MA) nuclides than the U–Pu vector does. In general, sodium void reactivity of an SFR core with TRU vector fuel increases because of spectrum hardening and an increase in MA nuclides fission rate compared with a core with U–Pu vector. As a result, sodium void reactivity is slightly positive in the core with the TRU vector.

3 Reactor shutdown capability

The purpose here is that the countries have a common view about reactor shutdown capability, especially about passive systems, which has been achieved by comparing Japanese and French safety requirements and design concepts. The teams studied whether the SASS can be applied to ASTRID 600 and examined its potential as a common design concept.

3.1 Common views on design approaches and requirements

France and Japan have similar safety design approaches, which are two active and one passive shutdown system (namely, CSD), and similar requirements for them. The main difference in the approaches is the type of CSD: the SASS for Japan, and a curie point electro magnet (CPEM)/a hydraulically suspended absorber rod, called an RBH, for France, ASTRID 600. The SASS is a passive safety system that automatically inserts control rods by the force of gravity, where rods will be detached when the coolant temperature rises under an anticipated transient without scram (ATWS) condition. Various out-of-pile tests have been carried out to investigate the basic characteristics of the SASS, demonstrating its holding stability under the reactor operation condition [11]. The CPEM is a device that can detach control rods to shut down a reactor after the coolant temperature has excessively increased under an unprotected loss of heat sink (ULOHS), which is typically caused by failures of a secondary coolant system or a tertiary water/steam system [5]. An RBH is a passive shutdown system actuated by a hydraulic suspension mechanism for unprotected loss of flow (ULOF) transients [13].

To assess the applicability of the SASS to ASTRID 600, the Japan team performed preliminary thermal-hydraulic analyses with CATHARE and computational fluid dynamics (CFD) codes, assuming that an RBH was replaced with the SASS. The purpose of these analyses is, in particular, to confirm if the SASS can work in ASTRID 600 without any additional structures, such as flow collectors, for shortening response time.

3.2 Applicability evaluation of the SASS to ASTRID 600

First, the Japan team calculated the allowable response time that satisfies safety criteria using CATHARE, a one-dimensional plant dynamics analysis code with point kinetics [16]. The model used consists of core channels (35 average F/As, 3 control rods, and 1 reflector) and a primary coolant circuit as shown in Figure 2. For the reactivity feedback, the Japan team took into account fuel

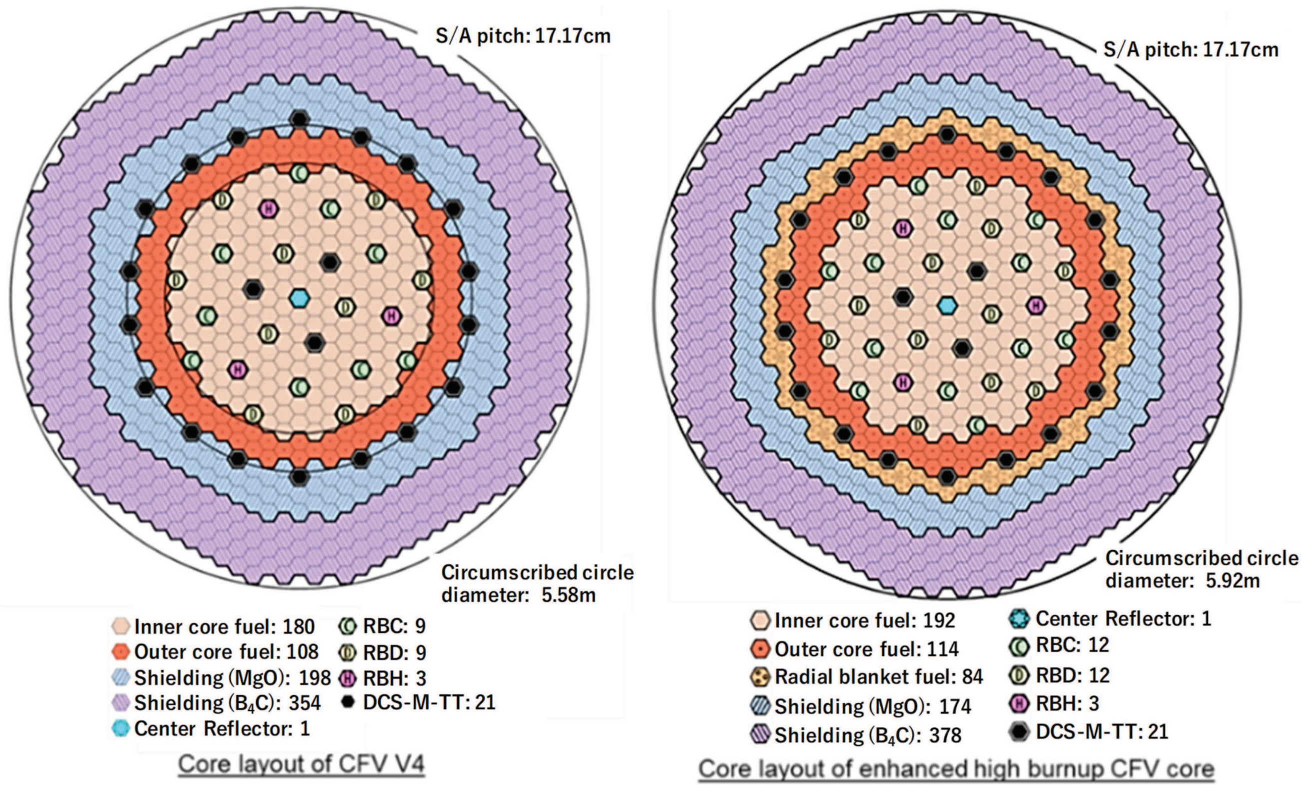

Fig. 1. Core configuration of CFV V4 core and enhanced high burnup CFV core.

Table 4. Main performances of enhanced high burnup CFV core.

Item	U–Pu	TRU
Average discharged burnup (core/total) (GWd/t)	147.1/80.7	147.8/80.7
Breeding ratio	1.11	1.13
Fuel residence time (months)	6×14.2	6×14.2
Max. linear heat rating (inner/outer) (W/cm)	427/409	413/398
Fast neutron fluence (n/cm^2)	4.03×10^{23}	4.06×10^{23}
Shutdown margin (cold/hot) (%Δk/kk′)	1.8/0.6	2.4/0.8
Sodium void reactivity ($)	−0.5	+0.3

doppler, fuel pin thermal expansion, coolant density variation, fuel cladding expansion, duct expansion, control rod insertion, and core support plate expansion [16,17]. A previous similar calculation on the SASS shows that the sodium temperature in a core drops with several seconds delay after the control rods have been delatched as soon as a temperature sensing part (TSP) of the SASS is heated by surrounding sodium up to a prescribed temperature, called de-latching temperature [12]. The delay can be estimated by the safety analysis using CATHARE. Of anticipated transients without scram events, a ULOF accident is a typical severe event; this paper, therefore, focuses on a ULOF analysis. For simplification, in this paper, the allowable response time for the SASS is defined as the delay from when the average outlet temperature of six F/As around the control rods reaches the de-latching temperature to when the maximum core sodium temperature reaches the sodium saturation temperature, 930 °C (safety criterion). The safety analyses were carried out parametrically with de-latching temperatures of 610, 630, 650, 670, and 690 °C. For a ULOF condition, a flow coastdown curve was given with a flow halving time of 10 s as a design requirement of the ASTRID primary pump. The CATHARE results show that the allowable response time for the SASS is around 26 s in the case of the de-latching temperature of 650 °C as shown in Figure 3.

Next, the Japan team calculated a response time from the outlet of F/As to the temperature sensing part for the SASS using CFD, which will be validated in the future through experimental analyses with experimental data on the response time. The CFD analysis was to evaluate the three-dimensional flow around the SASS. In addition, the Japan team calculated the response time for the SASS using CATHARE results of F/As outlet temperature and flow rate reduction as boundary conditions. The CFD analysis was for the lower part of the outer sleeve of a control rod drive mechanism (CRDM), which is originally requested for ASTRID 600. Figures 4 and 5 show the analysis model. Using the half circle of the inner

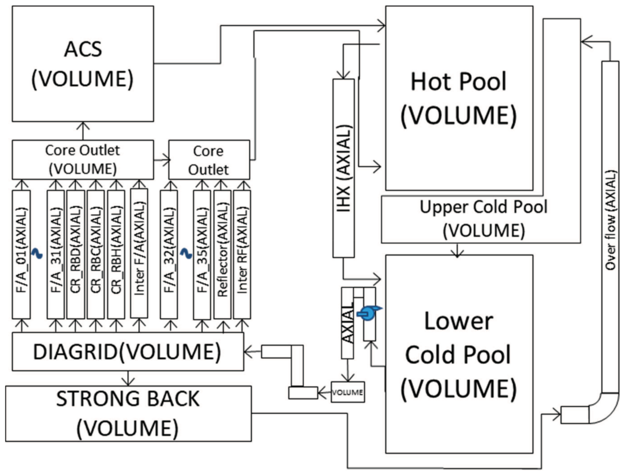

Fig. 2. CATHARE analysis model.

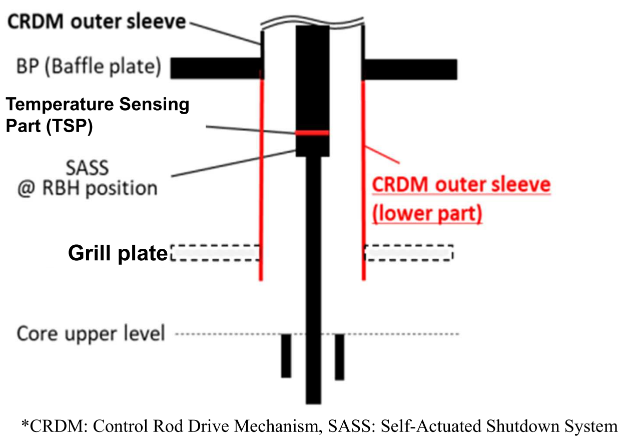

Fig. 4. Analysis model for CFD code.

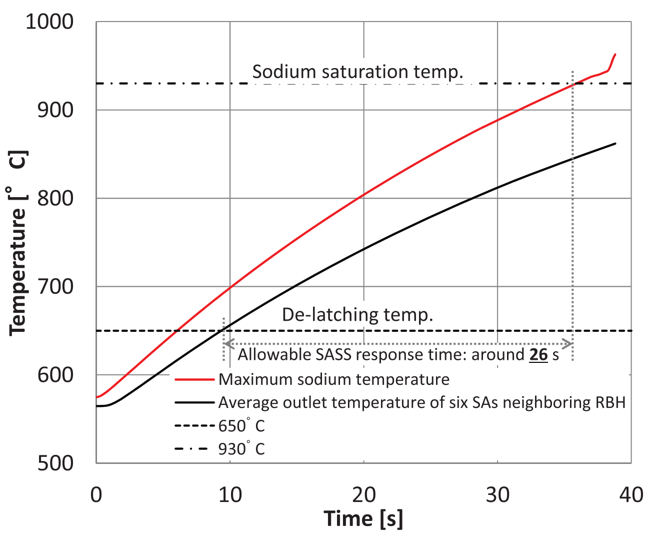

Fig. 3. Calculated temperature in ULOF.

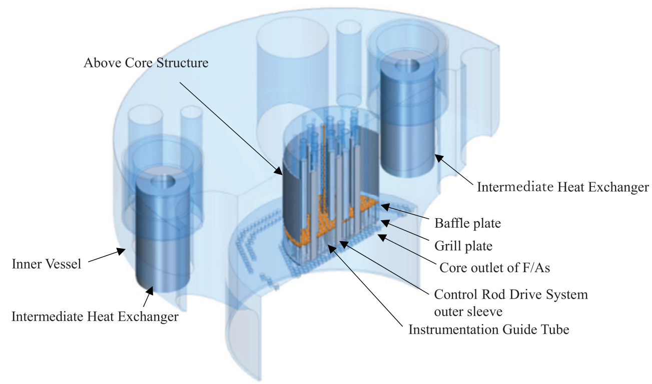

Fig. 5. Bird's eye view of CFD model.

vessel (Fig. 5), the Japan team performed the CFD analysis with the software, STAR-CCM+, which models the structure of the SASS in detail to assess the performance of the SASS. STAR-CCM+ modeled the components into hexahedral and polyhedral computational meshes. To provide more computational resources to the regions of interest, namely, the SASS and the region between F/A outlets and the baffle plate (BP), the Japan team used fine mesh for the region surrounding the SASS and coarse mesh for the hot pool.

Figures 6 and 7 show the analysis results of velocity distribution and temperature distribution during normal operation ($t = 0$ s), suggesting that the temperature distribution depends on the velocity distribution. These results show that both axial and lateral flows occur between the core outlet and BP. Regarding the control rods with the SASS, the lower part of the CRDM outer sleeve cuts off low-temperature sodium flowing from the neighboring RBD. The analysis revealed that low-temperature sodium flowing from the outlet of the RBH assembly is mixed with high-temperature sodium from the outlet of the neighboring F/As, resulting in a flow of high-temperature sodium into the CRDM outer sleeve. This flow would heat the SASS.

Table 5 summarizes the CFD-calculated SASS response times, which can be defined as the time from when the average core outlet sodium temperature of F/As around the RBH reaches the de-latching temperature to when the TSP solid temperature reaches the de-latching temperature.

The comparison between the allowable CATHARE-calculated SASS response time and the CFD-calculated SASS response time shows that the CFD-calculated time is much smaller than the other. France and Japan teams, thus, concluded that the SASS is applicable to ASTRID 600 without any additional structures such as a flow collector.

4 Severe accident mitigation

France and Japan teams studied the effectiveness of mitigation measures against core damage mainly for ASTRID 600. The purposes here are to review the feasibility and further study needs from the viewpoint of the acceptabil-

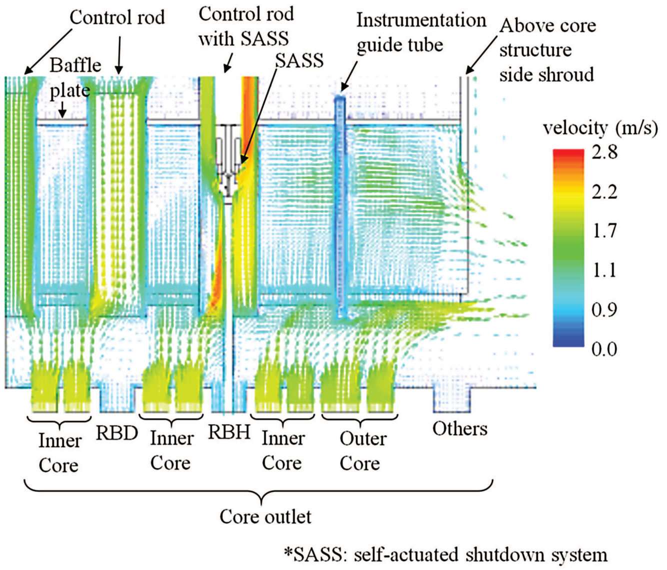

Fig. 6. Velocity distribution during normal operation.

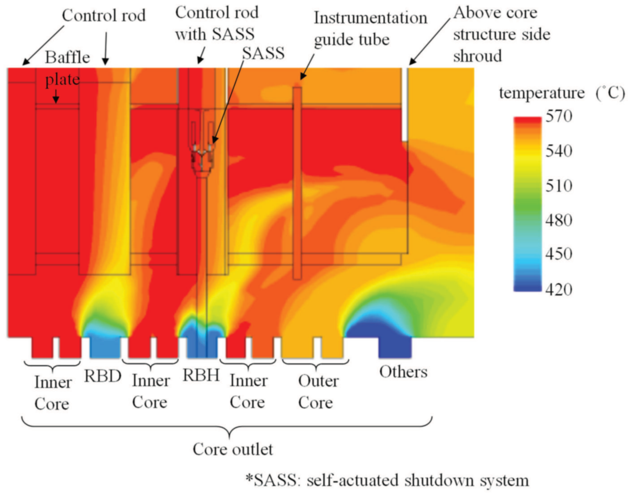

Fig. 7. Temperature distribution during normal operation.

ity of ASTRID 600 considering safety design requirements in Japan.

4.1 Common views on design approaches

The following are views common to France and Japan on safety objectives and general approaches for a severe accident:

- mitigating the consequences of hypothetical severe accidents by design.
- Maintaining containment functions against possible mechanical energy release due to a core damage accident (CDA).
- Limiting possible radiological off-site release.
- Reaching a steady safe state after a CDA by maintaining corium in a sub-critical, cool able condition through the IVR strategy.

The countries have a safety design approach in common, that is, the installation of design measures to mitigate consequences of core damage, which is postulated in a beyond-design basis accident domain. Specifically, the approach is adopted to prevent excessive energy release in the course of core damage, to discharge the core materials, if melted, through a steel duct structure, called a DCS-M-TT in this study, installed in the core, and to retain and cool it on a core catcher in the reactor vessel so that IVR can be achieved. France and Japan teams discussed the effectiveness of the mitigation measures represented by the CFV core, DCS-M-TTs, and a core catcher to determine their potential in light of existing evaluation results and knowledge from past experience. Regarding mitigation provisions and the expected influence on ASTRID 600, the common understandings of the items to be investigated to judge its acceptability in Japan have been built as follows.

Table 5. Self-actuated shutdown system (SASS) response time.

De-latching temperature (°C)	CFD-calculated SASS response time (s)	Allowable SASS response time (s)
610	3	30
630	3	28
650	4	26
670	5	24
690	6	22

- **Sodium void reactivity features**: the occurrence of a severe accident that can result in core melting can be schematically divided into four phases: the primary phase, the transition phase, the secondary phase, and the post-accidental cooling phase [18]. Large positive reactivity insertion due to coolant boiling will be prevented by the effects of the upper sodium plenum and internal fertile zone, which are specific characteristics of the CFV core. Steel reactivity worth will be also reduced in the CFV core. Therefore, positive reactivity due to molten steel motion will be limited, preventing power excursion during the primary phase. Japan team suggests possible relaxation of sodium void reactivity requirement that is zero or negative value in the core design, commenting such void reactivity is not requisite.
- **Fuel bundle features**: molten fuel in the fuel pin is expected to move upward through the central hole of fuel pellets, causing negative reactivity effects to be introduced in the primary phase in ASTRID 600, especially in unprotected transient overpower (UTOP). The following features of the fuel pins of the inner core of the CFV facilitate the in-pin fuel motion: larger linear power rating and annular pellets of the upper fissile zone, and solid fertile pellets below it. To take advantage

of the in-pin motion, this research needs model development and its validation with experimental results.

- **Measures to remove molten corium out of the core**: for severe accident mitigation, special guide tubes, namely DCS-M-TTs, that can reduce corium mass in the core region were added in the CFV design to shorten the duration of the second phase so that total energy produced during the phase can be reduced. One example evaluation that compared the effects of the DCS-M-TTs based on SIMMER calculations with the 2D-RZ model of the CFV core exhibited mitigation effects in power excursion [18]. The total mechanical energy in the secondary phase of ULOF will be significantly reduced. Effective fuel discharge can be obtained after mild recriticality. Other DCS-M-TTs installed between the outer core and blanket region are also expected to prevent internal fuel-storage damage. Molten fuel discharge through control rod guide tubes can be facilitated by modifying a flow orifice structure. In addition, radial reflector assemblies that can limit the radial propagation of molten materials to the internal fuel storage can be an option to prevent internal fuel storage damage.
- **Sacrificial box:** a structure, called a sacrificial box, jointly studied and devised by both teams is expected to support the functions of the DCS-M-TT and the core catcher. In a severe accident situation, it works as a buffer between the DCS-M-TT and the cold plenum above the core catcher to mitigate excessive molten material jet on the core catcher. It also helps prevent a coolant flow from bypassing the intermediate heat exchanger through the DCS-M-TT assemblies in normal operation. There are, however, still considerable uncertainties in the operation of the sacrificial box in a severe accident situation. The feasibility must be addressed by design and R&D studies in the future.
- **Cooling provisions**: the core catcher is installed at the bottom of the main vessel to terminate an accident within the main vessel, which is called an IVR strategy. To use the top-down approach [19] that is based on the IVR strategy, the core catcher has to collect the corium which is the whole melted core and reflectors located inside the DCS-M-TT row. One effective measure for promoting fragmentation and solidification of core debris falling on the core catcher is to extend the sodium region with enough sodium inventory to reach above the core catcher. The size of a core catcher must be such that it allows the spread of core materials to be maintained in a sub-critical state. Although further study is needed to establish the design of the core catcher, it may be feasible to achieve IVR through evaluations using loading conditions based on phenomenological consideration. The consideration also includes limiting mechanisms of molten materials discharged from the core to the core catcher in a short time.
- **Containment provisions**: the teams mainly discussed confinement issues, focusing on the response of the primary containment boundary under possible energetic events. Gas bubbles result from the vaporization of fuel, and of steel and sodium, which is combined with the release of fission gases contained in the fuel. Those bubbles are characterized by a bubble law simplified considering the conservativeness of the top-down approach. The bubble law explains the relationship between the pressure acting from the inside of the bubble toward the outside and its volume during its expansion.

$$P = P_0 \left(\frac{V}{V_0}\right)^{-n}$$

where:

P: bubble pressure at time t,
V: bubble volume at time t,
P_0: initial pressure of bubbles,
V_0: initial volume of bubbles,
n: coefficient representative of bubble contents and energy transfer between bubbles and their environment.

Assuming the initial volume and the bubble law on the expansion, the initial pressure was parametrically changed to simulate different mechanical energy. As a result, the teams can assess the maximum mechanical energy that the main vessel would endure. Here, it should be noted that the conditions such as the bubble law chosen for the top-down approach should not be excessively hypothetic to avoid unrealistic results.

The France team will perform systematic R&D for analysis code development and phenomenological data acquisition. The teams have identified additional R&Ds as common topics to both countries, such as studies of the failure behavior of irradiated, axially heterogeneous fuel pins for the CFV core, and the release process of fission products in a reactor vessel. To choose events to be evaluated,

- probabilistic insight should be considered in selecting CDA initiators, although the France team considers CDAs in the frame of defense in depth and requests that all types of initiating events and their sequences be assessed;
- UTOP events should be classified into either severe accidents to be mitigated or practically eliminated [19]. In the French approach, those events are classified by their consequences based on parametric evaluation considering reactivity insertion rates and the amount of reactivity inserted. The classification of situations to be practically eliminated can be justified, although the Japan team suggests that probabilistic assessment results also be considered when choosing to initiate transients.

5 Conclusions

France and Japan teams examined the countries' design requirements and approaches for the designs of a core and fuel, reactor shutdown systems, and mitigation measures against a severe accident to have a common technical view on the common SFR concept acceptable to both countries. Furthermore, the teams examined the acceptability of ASTRID 600 in Japan as a common SFR concept.

Regarding the core and fuel, the teams studied the feasibility of the enhanced high burnup core and fuel using oxide dispersion-strengthened steel cladding for ASTRID 600, considering Japan's core performance targets. For the core neutronics design, the teams have established the ASTRID 600-based enhanced high burnup core that satisfies most of the required design targets and conditions.

Regarding reactor shutdown capability, the teams have confirmed that they have similar safety approaches and main requirements for active and passive shutdown systems for ASTRID 600. The main difference is the type of CSD: the SASS for Japan, and the CPEM and RBH for France. The preliminary numerical analysis shows that the SASS can be used in place of the RBH device for ASTRID 600, satisfying their main requirements. The results of thermal-hydraulic studies with CATHARE and a CFD code show that the SASS can prevent core damage without any additional structures such as a flow collector and can be used as a CSD in both countries.

Regarding severe accident mitigation, France and Japan teams have investigated the effectiveness of the mitigation measures against core damage mainly on ASTRID 600, such as mitigation of energetic response thanks to sodium void reactivity features of the CFV core, molten corium that would come out of the core through DCS-M-TTs, IVR, the sacrificial box, and support structures. Both countries have a common safety design approach, that is, the installation of design measures to mitigate consequences of core damage postulated in the beyond-design basis accident domain. Regarding mitigation provisions, the teams have identified a common understanding of the items to be investigated for the common SFR concept.

This study has shown that the common design concept based on ASTRID 600 is feasible to demonstrate the SFR core and safety technologies for both countries. These collaborative activities can form the basis of SFR technology for future demonstration and collaboration.

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Conflict of interests

The authors declare that they have no competing interests to report.

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Data availability statement

This article has no associated data which cannot be disclosed due to legal/ethical/other reasons.

Author contribution statement

The related authors are the main contributors to the corresponding chapters: Chapter 2 Core and Fuel Designs: Kazuya Takano, Shigeo Ohki, Takayuki Ozawa, Masashi Ogura, Christophe Venard, Denis Verrier, Chapter 3 Reactor Shutdown Capability: Hidemasa Yamano, Yumi Yamada, Laurent Costes, Benoit Perrin, Chapter 4 Severe Accident Mitigation: Shigenobu Kubo, Kazuya Koyama, Laurent Costes, Bernard Carluec.

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