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| Title | Plant system study of France-Japan common concept on Sodium-cooled Fast Reactor |
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| Citation | EPJ Nuclear Science & Technologies, 8, p.11_1-11_10 |
| Text Version | Published Journal Article |
| URL | https://jopss.jaea.go.jp/search/servlet/search?5074576 |
| DOI | https://doi.org/10.1051/epjn/2022006 |
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Plant system study of France–Japan common concept on Sodium-cooled Fast Reactor

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Received: 2 February 2022 / Received in final form: 4 February 2022 / Accepted: 6 May 2022

Abstract. This paper provides an overview of plant system studies to establish a common technical view for Sodium-cooled Fast Reactor (SFR) concept (the common SFR concept) between France and Japan based on ASTRID 600 and the new concept with downsized output (ASTRID150). One of important issues on a reactor structure design is to enhance seismic resistance to be tolerable against strong earthquake such that postulated in Japan. A concept of High Frequency Design (HFD) is shared, in which the natural frequency of the reactor structure should be higher than that of peak acceleration of vertical floor seismic response with a horizontal seismic isolation system. The design options related to HFD have been examined and design recommendations are established. ASTRID 600 adopted a gas power conversion system to strictly eliminate the chemical reaction risks due to the proximity of sodium and water in the steam generator units. On the other hand, a steam generator (SG) is thought to be a concept with high technical readiness level and is a reference option in Japan and a backup option in France. Then, design comparison of the SG with single-walled helical coil tubes was mainly conducted in this study from the viewpoint of safety and so on. A common concept of a decay heat removal system is discussed to achieve practical elimination of loss of decay heat removal function. A fuel handling system studies are performed to eliminate and ex-vessel storage of spent fuels in sodium to reduce a construction cost. An adequate confinement system is investigated to achieve practical mitigation of large radiological release to the environment even under the condition of core destructive accident.

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 [1,2]. With valuable support from CEA and Framatome, the Japanese team (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 (DHRS), curie-point electro-magnetic

shutdown devices, and seismic isolation systems; fabricability studies of the above core structure (ACS) and polar tables; improvement of core catcher design; plant thermal transient evaluation; design analyses of the main vessel (MV) and an inner vessel; and core design evaluation.

Since 2017, the researchers of both countries have conducted studies to deepen the cooperation further and achieve a common view on SFR concepts that possibly form the basis for future collaborations and standardization for SFRs in the countries [3]. Based on the initial ASTRID design (600 MWe), French and Japanese teams examined ways of developing a feasible, common design concept to be able to be built in both countries (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. After that, both teams started to study mainly in

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the area of “safety approach and core concept” and “a plant system”. In this study, four plant concepts are under consideration to draft common plant specifications for each system and component. The first three plant concepts are the basis of the study, and the last one would be a target of this study.

- ASTRID (600 MWe) [1].
- ASTRID (150 MWe) [3].
- Japan’s pool-type SFR design [4].
- a common SFR concept.

The following chapters the report results of the studies for the plant systems. Typical points in the studies of plant system design are how to harmonize differences in both countries as the common SFR concept, in fact differences in seismic condition (mainly in reactor structure and primary system areas), differences in technical experiences and maturities of each country (mainly in secondary and tertiary system, fuel handling system areas), differences in safety approach (mainly in decay heat removal system and containment of the reactor areas). Each chapter basically consists of four parts of French concept introduction, Japanese concept introduction, commonization discussion for 600 MWe grade plant, and impact of downscaling to 150 MWe. Technological decision would be made based on similarity of a system and a component, commonality of physical phenomena, design technologies to be able to enhance designs in both sides, priority on the top level requirements.

2 Reactor structure and primary design

ASTRID 600 is a pool-type SFR that mainly consists of a main vessel (MV), a conical inner vessel, an ACS, three primary mechanical pumps, four intermediate heat exchangers (IHXs) (shell- and tube-types), an internal core catcher, and support structures that support these components [5,6]. The Marcoule site in France is the reference site in setting the design basis earthquake for ASTRID, and the beyond design basis earthquake is 1.5 times the design basis.

In contrast, Japanese SFRs such as Joyo, MONJU, and JSFR are loop-type because loop-type structure offers seismic advantages. The design basis earthquake (called Ss earthquake in Japan) of JSFR is equivalent to or larger than the average Ss earthquake conditions for most light water reactor sites in Japan and higher than design basis earthquake conditions in France. Plant safety must be assured even under the condition of 1.8 times as large as Ss earthquake [7]. In Japan, it is required that earthquake resistance be improved and a horizontal seismic isolation system be installed for a pool-type reactor. The features of the advanced seismic isolation system for SFR are as follows: (1) it mitigates the horizontal seismic force through the thicker laminated rubber bearings with a longer period and improves damping performance by adopting oil dampers instead of steel bar dampers, (2) it mitigates the vertical seismic force through the thicker laminated rubber bearings with a longer period.

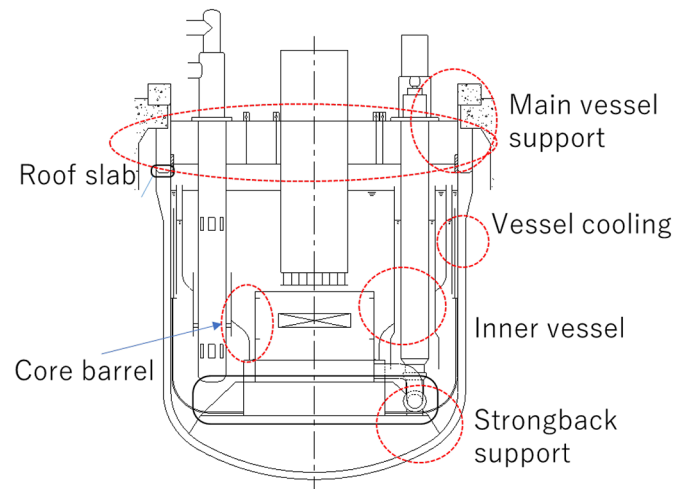


Fig. 1. Discussion parts to improve structural intactness.

In parallel, the main technological challenge is to assure structural integrity against a certain level of seismic loads concerning the effect of the horizontal seismic isolation system. To improve the earthquake resistance for ASTRID 600, Japanese team proposed several modifications for the design, and they were examined. Points of study are shown in [Figure 1](#); MV support, vessel cooling, inner vessel, strongback support, core barrel, and roof slab. Eigen frequency of the reactor structures shall be designed not to overlap the peak of the horizontal and vertical floor response spectrum with the horizontal seismic isolation system. In case that the floor response spectrum has high vertical acceleration for thin-walled cylindrical shell such as a large-sized MV of a pool-type reactor, it is appropriate that the structure is designed to enhance its stiffness to suppress the response low sufficiently. Earthquake levels and Japanese regulations lead the Japanese team to seek a High-Frequency Design (HFD) for the reactor block, i.e., the natural frequency of the reactor structure should be higher than that of peak acceleration of vertical floor seismic response with a horizontal seismic isolation system, for improvement of earthquake resistance ([Fig. 2](#)). For example, the targeted natural frequency of the MV of Japan’s pool-type SFR design is more than 10 Hz (less than period 0.1 s) concerning eigen frequency and a peak of the vertical floor response.

Japanese team shared the logic of the HFD with their counterpart and presented that the shapes of reactor structures with the HFD could be a common design between the countries against severe earthquakes. At the end of study, a shape of ASTRID 600 design was adopted for the reactor structures of the common SFR concept.

Regarding downscaling of output power, design requirements and discussion points are the same as ASTRID 600. Thanks to the comprehensive understanding of HFD, the design of ASTRID (150 MWe) achieved an eigen frequency of approximately 10 Hz (HFD) for the reactor block, thanks to the HFD that reduces the

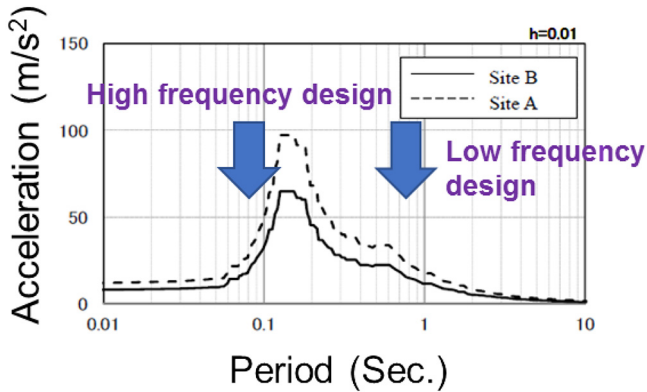


Fig. 2. Concept of High Frequency Design against Floor Response Spectrum.

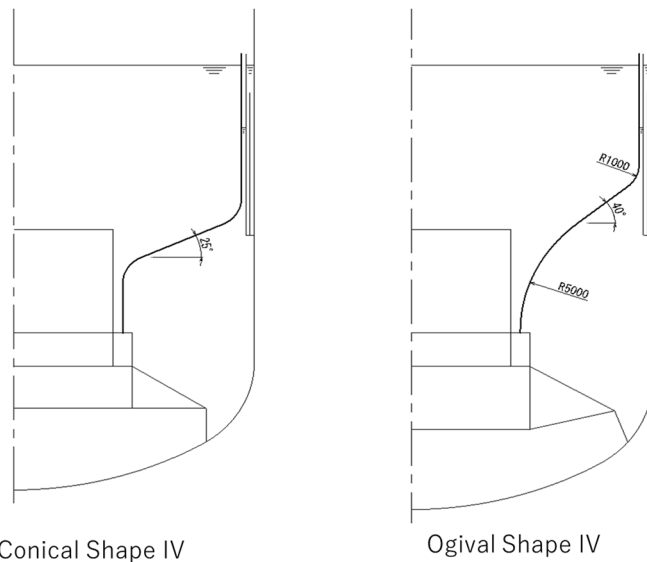


Fig. 3. IV shape to improve seismic tolerance.

diameters of the upper closures and the distance between the MV and the reactor pit and that anchors the reactor-block supporting skirt onto the intrados (the internal metallic plate of the steel concrete structure) of the reactor pit. Moreover, a preliminary study of the inner vessel dynamic behavior and its seismic buckling strength revealed that an ogival inner vessel design (Fig. 3) is highly suitable for ASTRID 150 to reduce the wall thickness and keep acceptable safety margins against buckling under seismic loads. However, it must be noted that these analyses of inner vessels are based on the ASTRID 150 design, and these results could be unsuitable for larger reactors or commercial reactors.

One other discussion point is the difference in the concept of the roof slab. The roof slab is cold in the Japanese design with an active gas cooling system, whereas it is warm in the French team's design. Since the differential displacement between the roof and reactor pit are to be minimized to achieve high stiffness, Japanese

design adopts the cold slab concept. For a common view built in France, although the temperature of the slab could be maintained at around 120 °C, the cold roof option is still open. Table 1 compares the advantages and disadvantages of the cold and warm roof slabs. This topic will be discussed in a future France-Japan SFR R&D collaboration program.

3 Secondary and tertiary systems

ASTRID 600 adopts a sodium-gas (Nitrogen) heat exchanger with a compact panel type module (Fig. 4) [8], Brayton cycle (a thermal efficiency of about 38%), and an electrical, magnetic pump for a secondary pump. The secondary loop consists of four loops and eight sodium gas heat exchangers. This power conversion system is the main option in order to strictly eliminate the risks of sodium-water reaction and sodium-water-air reaction in between secondary and tertiary circuits, which potentially could have a significant impact on safety systems. A backup option is a monolithic steam generator (SG) with a helical coil tube made of Alloy 800 and Rankine steam cycle [9], which can achieve a thermal efficiency of about 42%.

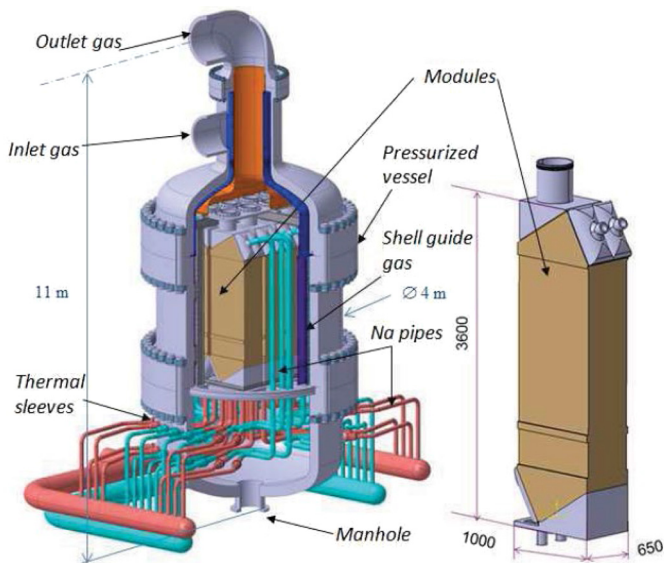
Japanese team adopts Rankine steam cycle and mechanical pumps for secondary systems prioritizing higher yields and higher technical readiness levels [10]. To provide a higher reliability to sodium-water reaction measures, the Japanese team initially sought an SG concept using a straight double-walled tube made of Mod. 9Cr-1Mo steel to improve the reliability of the plant against sodium-water reaction caused by SG tube failure [11]. However, the pool-type design proposed by Japan adopts an SG with a monolithic helical coil tube made of Mod. 9Cr-1Mo steel concern the maturity of reliability and reduce future R&D loads.

Although there are fundamental differences in the secondary and tertiary systems between the designs proposed by French and Japanese teams according to their requirements, the backup concept of France is similar to that of Japan. Both teams decided to use helical coil SGs for the steam power conversion system for the common SFR concept. They then compared Alloy 800 and Mod. 9Cr-1Mo steel for the SG tubes on the basis of the countries' safety approaches to sodium-water reaction in SGs. Table 2 shows qualitative comparison of SG tube materials. This table shows that there is no strict advantage on one material compared to the other one. The selection, therefore, depends on which specification France and Japan place greater value. For example, based on PROPANA code calculations [9], Alloy 800 offers better resistance to wastage than Mod. 9Cr-1Mo (Fig. 5); however, the raw material of Alloy 800 is far more expensive generally. In fact, the two materials are applicable to the tubes for the SGs.

In terms of a safety philosophy in the case of sodium-water reaction (SWR), both countries require the leak tightness of primary-secondary coolant system interface (in fact IHX) for design basis events. Thus, the outer shell of the steam generator must be designed to remain leak tightness for design basis SWR events.

Table 1. Comparison between the cold and the warm roof slab

| | Cold roof slab | Warm roof slab |
|------------|---|--|
| Advantages | <ul style="list-style-type: none"> – Reduction of thermal expansions of the bottom plate – Reduction of uncertainty of horizontal locations of ACS – Reduction of temperatures of seals, concretes and electrical devices – NPP availability (shorter shutdown) – No restricted access to the roof slab due to temperature during normal operation | <ul style="list-style-type: none"> – Reduction of thermosiphons effects along components penetrations – Reduction of risk of Na deposit in components penetrations – Reduction of thermal power to evacuate through cooling system: allow to replace complex thermal insulation (Na compatible) by a simple baffle – Reduction of thermal stress within structures |
| Drawbacks | <ul style="list-style-type: none"> – Cost – Increment of thermal insulation and of roof slab total height – Impact of the active gas cooling system on the reactor building – Increment of the risk of thermosiphons effects along components penetrations – Risk of Na deposit in component penetrations and impossible removal of components for maintenance/ renewal – Increment of thermal gradients within structures (including the main vessel and components) | <ul style="list-style-type: none"> – Reduction of seals lifetime – Limitation of tightness performances of seals on the roof slab during accident (due to higher temperatures) – Height of the cylindrical skirt that supports the roof slab and the main vessel – Reduction of NPP availability due to cooldown transients before shutdown and heating transients before start – Needs of thermal insulation on top of the roof slab – Higher temperature of electrical devices – Increment of uncertainty of horizontal locations of ACS – Resistance of concrete to higher temperatures |

**Fig. 4.** Sodium-gas heat exchanger concept in ASTRID600.

For design extension conditions, the Japanese team considers severe SWR (that correspond to tube ruptures due to the leak propagation in the tubes bundle in case of multiple failures of countermeasures against SWR), and the leak tightness of IHX is also required. On the other hand, the French team considers cumulating loss of leak

tightness of the IHX (or loss of secondary loop envelope or loss of steam generator outer shell) to ruptures of multiple tubes event and aims at demonstrating the absence of core disruptive accident or any other severe accident scenario. This remaining issue of the difference in the safety philosophy will be studied in the R&D collaborative framework between France and Japan starting from 2020 [12].

For ASTRID 150, the Steam Rankine cycle with a mechanical pump was adopted to minimize future R&D costs. The SG is the same type as the one selected by the Japanese team. In addition, both teams share a common view that the number of components should be reduced, and the size of components should be enlarged to be qualified as commercial reactors. Consequently, with ASTRID 150, both teams reached a complete common view.

4 Decay heat removal system

Severe accidents resulting from loss of decay heat removal (DHR) cannot be reasonably mitigated; such situations, therefore, must be practically eliminated with a high level of confidence. The possibility of certain conditions arising may be considered to have been ‘practically eliminated’ if it would be physically impossible for the conditions to arise or if these conditions could be considered with a high level of

Table 2. Comparison between the cold and the warm roof slab

| Properties | | Advantages and Disadvantages | |
|----------------------|---|------------------------------|--------------|
| | | Alloy 800 | Mod. 9Cr-1Mo |
| Thermal conductivity | | Applicable | Superior |
| Thermal expansion | | Applicable | Superior |
| Cyclic behavior | | Superior | Applicable |
| Welding | | Superior | Applicable |
| Manufacturing | | Superior | Applicable |
| Material cost | | Applicable | Superior |
| Chemical behavior | Intergranular corrosion, SSC, corrosion in NaOH | Applicable | Superior |
| | Hydrogen embrittlement, Oxidation Hydrogen migration | Superior | Applicable |
| Wastage | | Superior | Applicable |

Superior: *Characteristics of a material is superior to the other material significantly.*

Applicable: *the material could be adopted technically, but the material is inferior to the other material.*

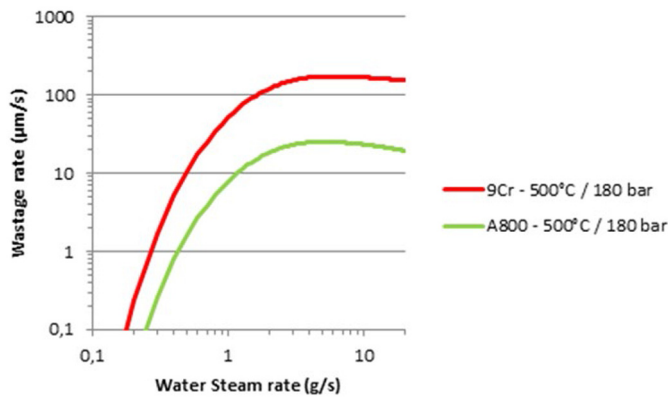


Fig. 5. Comparison of the wastage rates between 9Cr and A800 alloys.

confidence to be extremely unlikely to arise [13]. In DHRS design, since it is hard to achieve a physically impossible condition, the design shall reduce occurrence frequency as low as possible based on the probabilistic evaluation. As defined in the Safety Design Criteria (SDC) and Safety Design Guidelines (SDG) [14,15], the purpose of a DHRS is practically to eliminate complete loss of DHR functions with robust demonstration. In DHRS design, the importance of lessons learned from the Fukushima Dai-Ichi accident and the GIF-SDC/SDG were emphasized, such as analysis and design measures against external hazards, and also importance of passivity. At present, the demonstration of practical elimination relies on deterministic approaches complemented by probabilistic approaches because of the immaturity of the probabilistic approaches and the limited experience feedback on SFRs. Precisely, as we have to take the deterministic approach for DHRS design, diversity, independence, and redundancy are important for DHRSs. In its application, since there is flexibility to apply the deterministic approach, implementation methods which depend on experiences and technical maturities of Japan and France are different.

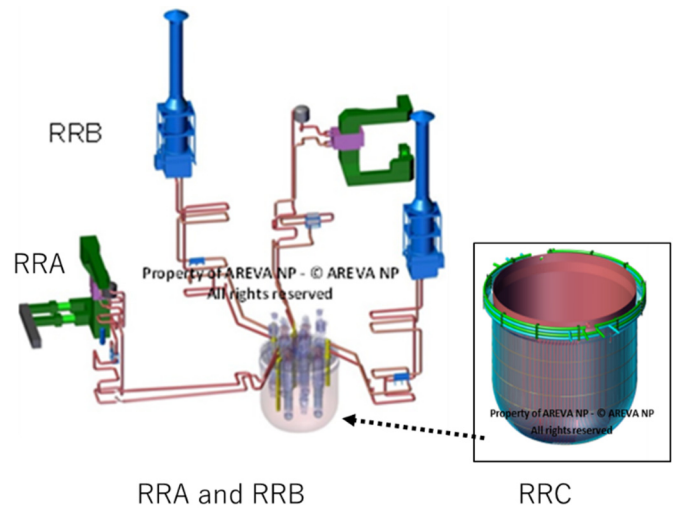


Fig. 6. DHRS in ASTRID 600.

The DHRS for ASTRID 600 is composed of two trains of active decay heat removal system with forced circulation (RRA) inserted in the MV, two trains of passive decay heat removal system with natural circulation (RRB) also inserted in the MV, one train of ex-vessel type decay heat removal system with forced circulation (RRC), and one train per a secondary loop of a dedicated system for normal shutdown with forced circulation (NDA) (a non-safety class component to maintain cold shutdown state) (Fig. 6). Comprehensively, the Japanese team proposed one dipped-type Direct Reactor Auxiliary Cooling System (DRACS) and one system that penetrated through the inner vessel-type DRACS, and four Intermediate Reactor Auxiliary Cooling Systems (IRACSs) (Fig. 7). The comparison studies showed that the redundancy of DHRS design for ASTRID 600 allows the safety requirements of both countries to be satisfied and that passive system with natural circulation capability need to be installed in the DHRS.

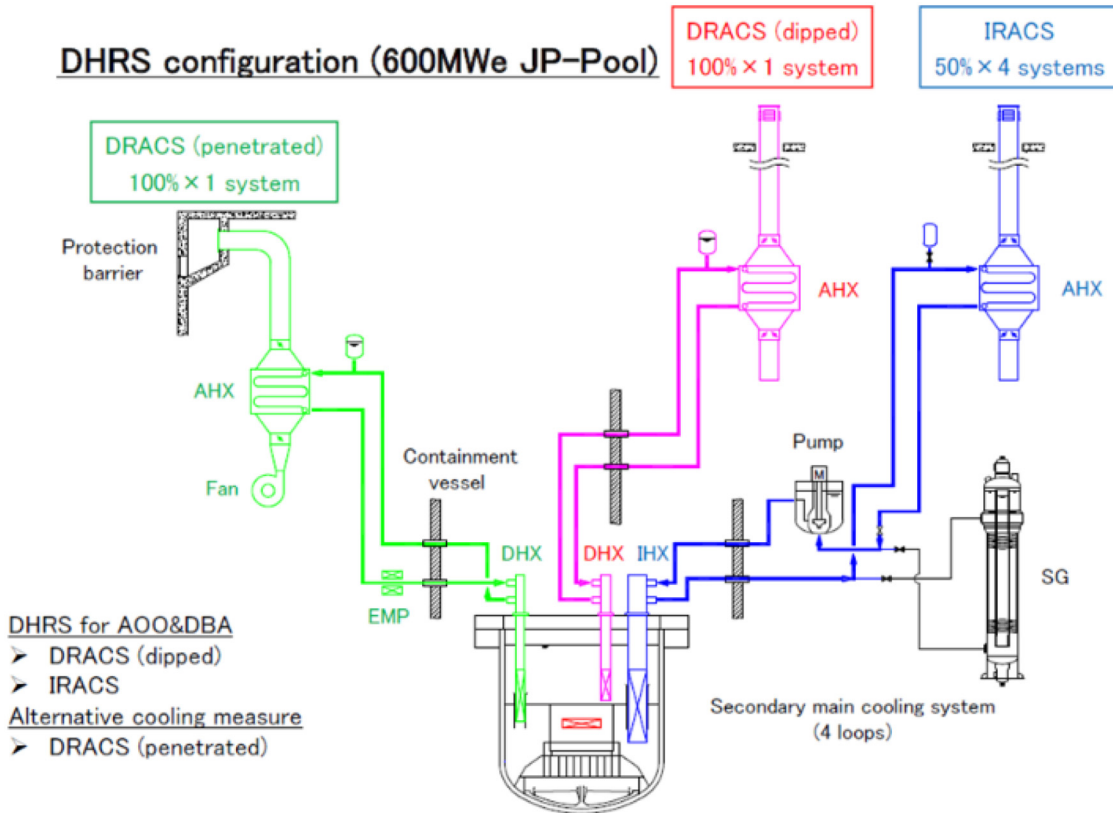


Fig. 7. Japanese proposal of DHRS (600MWe plant).

On the transition from ASTRID 600 to ASTRID 150, the French team proceeded to a Design to Cost methodology. Finally, the DHRS is composed of two RRBs, two RRCs, and an NDA in order to enhance the capability of natural circulation in the MV. The capability of technology demonstration is an essential matter in a downscaled reactor. These results confirmed that the following test conditions are essential for Verification and Validation (V&V) of design tools for the commercial reactor DHRS from the viewpoint of natural circulation capability (Fig. 8);

- higher primary flow: simple and sufficient core cooling via IHX (Type A)
- lower primary flow: complex and limited core cooling (Type B)

This remaining issue will also be studied in the R&D collaborative framework between France and Japan starting from 2020 [12].

5. Fuel handling system

The Fuel Handling System (FHS) of ASTRID600 is composed of the in-vessel storage of fuels (IVS), the ramp type ex-vessel transfer machine, the external buffer zone (EBZ) in sodium located in the containment vessel (CV) to uncouple the handling phase for fuel loading/unloading from those of washing and storage. Storage capacity is for one batch of the core plus the number of failed fuels. Cleaning is done by moist CO₂ with approximately 500 degree-C plus water rinse (Fig. 9).

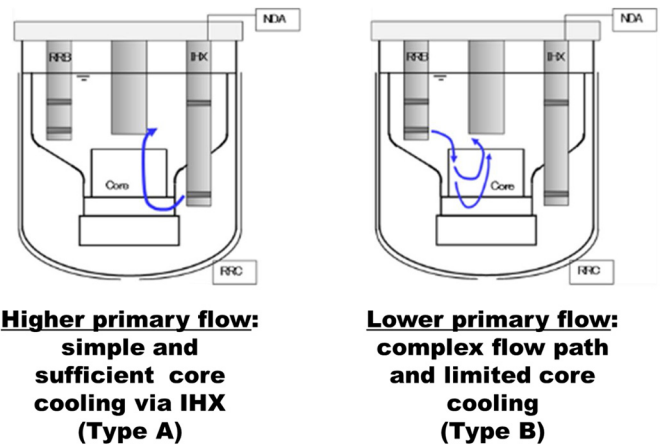


Fig. 8. DHR flow pattern in MV to be demonstrated.

Comprehensively, FHS in Japanese pool-type reactor (Fig. 10) composed of the external vessel sodium storage tank located outside the containment vessel to store a core (whole core discharge) plus one batch of the core and number of failed fuels, the cask car type ex-vessel transfer machine, spent fuel cleaned by Argon (Ar) gas blowing and inactivation by moist Ar gas without water rinse (direct storage of SF in the spent fuel pool), etc. These differences in system configuration and capacity come from the priority of development targets (safety, economic

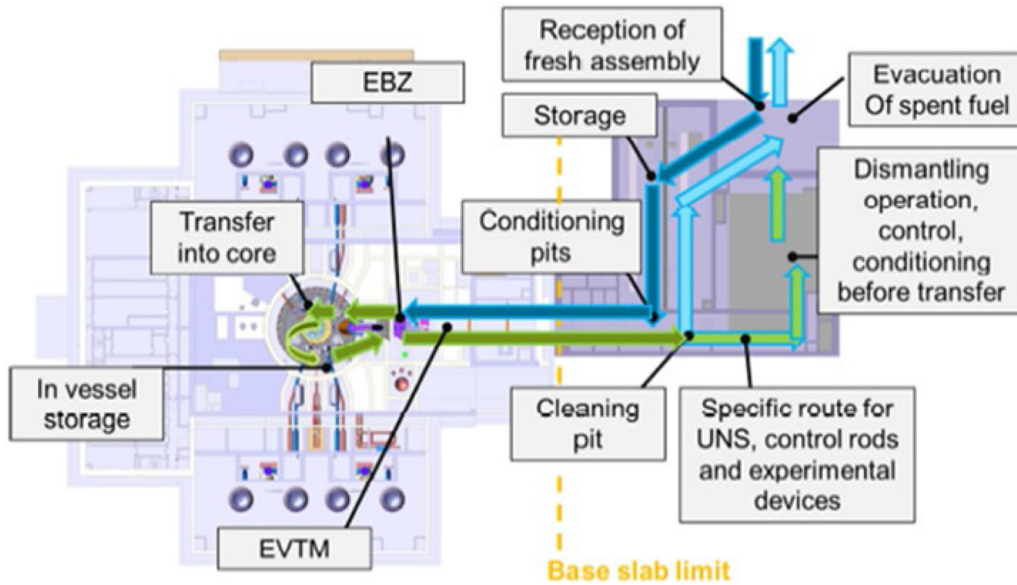


Fig. 9. FHS in ASTRID 600.

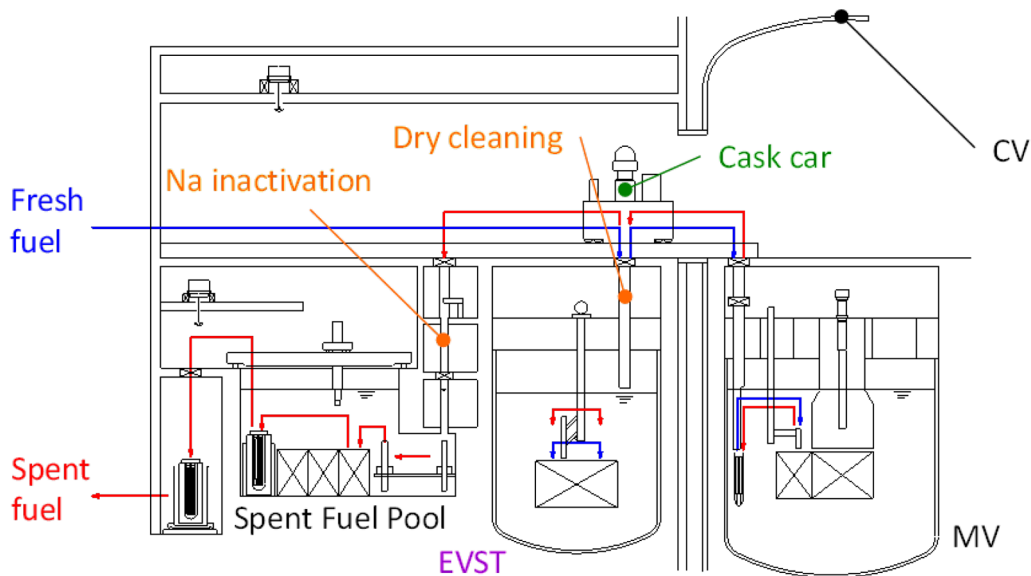


Fig. 10. FHS in Japan pool type SFR.

competitiveness in R&D phase and commercial phase, and user requirements such as electric utilities) and technological experiences in each country.

To reduce construction cost, the change from 600 MWe to 150 MWe was proposed by France, with an external buffer zone eliminated to reduce the commodities of the plant, and the type of an ex-vessel transfer machine was also changed with a cask car concept responding to the elimination (Fig. 11). The spent fuel from the core is transported directly to a water pool after washing and inactivation. In this concept, failed fuels are stored in the MV. The main technical challenges here were (1) to eliminate the Ex-Vessel Storage (EVS) system (This is especially challenging to manage failed fuel without an

EVS system.), and (2) to employ an ex-vessel transfer machine without a sodium pot. As for (1), France has high confidence about storing the open clad failed fuel in the reactor vessel thanks to the Phénix feedback where open cladding failed fuel was stored in its MV during 300 effective full power days. As a result, the amount of sodium entered the cladding became negligibly small, and a post irradiation analysis showed that no reaction between sodium and the fuel occurs. Regarding (2), Japan has already demonstrated a similar concept in MONJU. Those challenges have been addressed, and it has been reached significant results thanks to the collaboration of knowledge between France and Japan. In conclusion, France and Japan will focus R&Ds to achieve the common view of the

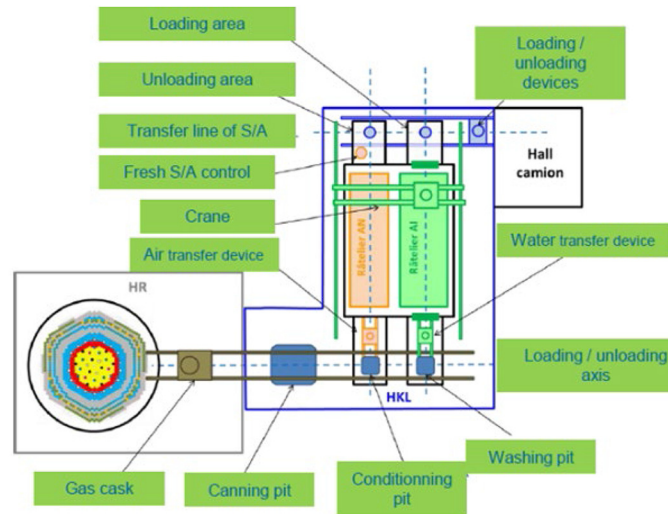


Fig. 11. FHS in ASTRID 150.

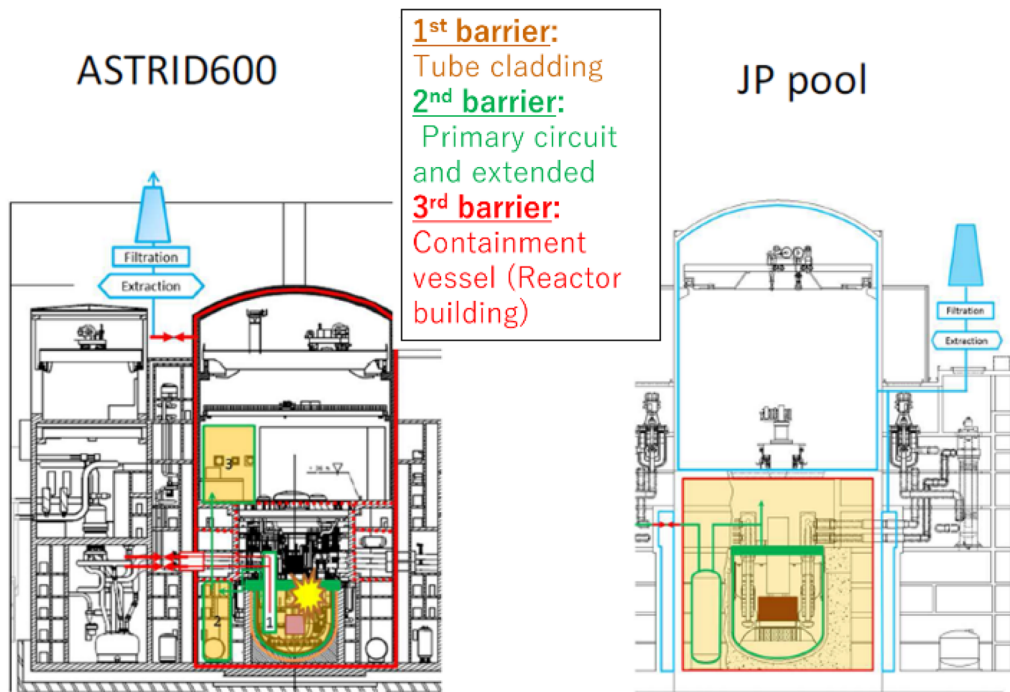


Fig. 12. Concept of containment.

FHS considering alternatively a limited Internal Vessel Storage (IVS) or an External Buffer Zone (EBZ) and cask car transportation as the main reference.

6 Containment of the reactor

The points needed to be considered for the containment and confinement of ASTRID that are common to the countries are the following: radioactive material release; external hazards relating to designs of the second and third barriers; the role of the reactor building (the third barrier in France's design) to protect the second barrier against

internal and external hazards; maintenance requirements for the reactor and its auxiliary systems; cost minimization; and application of Steel Concrete (SC) structure. Figure 12 shows the concepts of the containment systems of France and Japan. Regarding the main concept of containment, the three barriers concept is common between France and Japan basically, and non-volatile degraded core materials are retained inside 2nd barrier.

– ASTRID 600: CCV (concrete CV) + Polar table + Retention chamber, Isolation devices, assuming both energetic and non-energetic Severe Accident (SA) situations;

Table 3. Role of barriers

| Event | | Role of barrier | |
|--|--|--|--|
| | | ASTRID600 | Japan's pool type reactor |
| DBA | FP release inside 2nd barrier | 2nd barrier | Primary coolant boundary (2nd barrier) Cover gas boundary (2nd barrier) |
| | FP release outside 2nd barrier | 3rd barrier | Containment vessel (3rd barrier) Emergency gas treatment system |
| DEC | Core melt accident (Non-energetics) | 2nd barrier (retention chamber to be significant) | Containment vessel (3rd barrier) Emergency gas treatment system |
| | Core melt accident (Energetics, primary sodium ejection) | 3rd barrier Sodium fire consequences on the 3rd barrier limited by the above roof area | Not considered |
| Sodium fire in the above roof area with opened polar table | | Structural integrity of the building to be maintained | Not considered |
| Fuel handling accident in the reactor building | | 3rd barrier | Ex-vessel transfer machine (EVTM) Emergency gas treatment system |

- ASTRID 150: SCCV + Retention chamber, assuming both energetic and non-energetic SA situations;
- Japan pool-type SFR: SCCV + Emergency gas treatment system, assuming the non-energetic SA situation.

Japan pool-type SFR adopted the innovative design concept, which is intended to secure early discharge of molten fuel from the core region in a core disruptive accident preventing severe re-criticality in case of core damage, called fuel subassembly with an inner duct structure; FAIDUS [16]. Table 3 shows the roles of the barriers. The countries have differences in the following: the concept of the barriers between the core fuel material and the environment, the assumption of an SA (energetic or non-energetic), and the provision of isolation valves on the secondary sodium circuits (ASTRID 600). The French team suggested that ASTRID have three confinement barriers and no emergency gas treatment system, meaning no leakage collecting area. A confinement bypass should be practically eliminated to prevent significant radioactive material release, which is required by the French licensing authority. On the other hand, Japan's pool-type SFR design uses an SC structure for its containment vessel. confirmed that both concepts have the potential to expand design options such as the provision of the polar table and, or retention chamber, application of SC structure, and no provision of the emergency gas treatment system.

In the future, residual confirmation to harmonize the main difference between France and Japan could be done. For example, residual confirmation matters are as follows:

- confirmation for leak tightness of the second barrier including retention chamber against non-energetic CDA and integrity of the second barrier against energetic SA situations;

- assumption of mechanical energy in the case of energetic SA situation, leak tightness of the third barrier and integrity of polar table;
- public dose assessment for CDA (energetic and non-energetic SA situations) and DBA with/without emergency gas treatment system.

7 Conclusion

France Japan succeeded to reach the following points as a result of common specification for the common SFR concept; MOX fuel Sodium-cooled Fast Reactor based on pool-type architecture, with the applicability of a high-frequency design for the reactor structure;

- steam Power Conversion System (PCS) with helical coil Steam Generators (SG);
- decay heat removal system: diverse and redundant in-vessel systems with natural circulation capability, external vessel cooling, and water cooling at the SG;
- fuel Handling System (FHS) considering a limited Internal Vessel Storage (IVS) or an External Buffer Zone (EBZ) alternatively and cask car transportation as the main reference;
- containment implying the Reactor Building with or without a Polar Table.

Based on the results of this study, a new R&D collaborative framework has started between France and Japan from 2020 [12].

Conflict of interests

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

Funding

This study includes some results of the “Technical development program on a fast reactor international cooperation, etc.” entrusted to JAEA by the Ministry of Economy, Trade and Industry in Japan (METI”).

Data availability statement

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

Acknowledgements. Many people engaged in the joint project, and it is a good opportunity to thank them for the quality of their work and their involvement in this innovative project. The project team is very grateful to all engineers, researchers, and SFR specialists and experts from CEA, FRAMATOME, JAEA, MHI, and MFBR, without whom all this work could not be possible. We appreciate the valuable advice from the people involved.

Author contribution statement

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Cite this article as: Atsushi Kato, Tomohiko Yamamoto, Masato Ando, Yoshitaka Chikazawa, Hisatomo Murakami, Kazuhiro Oyama, Fumiaki Kaneko, Koichi Higurashi, Frédéric Chanteclair, Marie-Sophie Chenaud, Franck Dechelette, Edouard Hourcade, David Plancq, Gilles Rodriguez, Bernard Carlucc, Alexandre Dauphin, Jean-François Dirat, Julien Hirn, Plant system study of France–Japan common concept on Sodium-cooled Fast Reactor, *EPJ Nuclear Sci. Technol.* **8**, 11 (2022)