Study on Safety Design Concept for Future Sodium-cooled Fast Reactors in Japan

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Abstract. This paper describes safety design concept for future sodium-cooled fast reactors (SFRs) in Japan, which is based on the safety design criteria and safety design guidelines developed in the auspices of the international forum of generation IV nuclear energy systems. The safety design of future SFRs should be advanced taking the feedback of operating experiences, achievement of existing technology, and innovative technology into account. Inherent and/or passive design features are utilized based on SFRs characteristics such as low pressure, high thermal inertia of the system. Lessons learned from the Fukushima Dai-ichi accident is one of important issues to be incorporated into the safety design concept. In order to realize commercial SFRs in the future, robust and rational safety design should be pursued by integrating various factors into the design and limiting additional specific systems, structures and components. Existing engineering principle for the design and manufacturing of SFR's components, and innovative technologies introduced in the FaCT project are keys to achieve the safety concept.

Key Words: Generation IV SFR, safety design, loop-type reactor, pool-type reactor

1. Introduction

JAEA has conducted a feasibility study (FS) on commercialized fast reactor cycle systems since 1999 until 2004, in cooperation with electric power companies and plant makers, aiming at commercialization of fast reactors in Japan. In the FS, a sodium-cooled fast reactor (SFR) and oxide fuel have been selected among other types of reactor and fuel as promising concepts that can be sustainable energy source in the 21st century. JAEA has specified a design concept for Japan Sodium-cooled Fast Reactor (JSFR) that will adopt advanced loop-type design, and has advanced development of innovative technologies introduced in JSFR in the Fast Reactor Cycle Technology Development (FaCT) project [1]. One of core design engineering companies in Japan, Mitsubishi FBR Systems (MFBR), has joined the FaCT project. JAEA together with MFBR have improved safety design measures for SFRs based on lesson learned from TEPCO's Fukushima Dai-ichi nuclear power plants accident [2][3][4]. JAEA has joined safety design criteria task force (SDC-TF) of Generation IV International Forum (GIF) to contribute to the establishment of safety design criteria (SDC) and safety design guidelines (SDGs) for Generation IV (Gen-IV) SFR with the confidence of the safety design, evaluation techniques, and concrete design of safety facilities accomplished in FS, FaCT and the following study. JSFR is a large-scale reactor loaded with oxide-fuelled core. JSFR's coolant system consists of two loops, in which primary pump and intermediate heat exchanger (IHX) are integrated. The safety features including reactor shutdown system, decay heat removal system, and containment facility, are applicable to other design concepts including pool-type.

This paper describes the current perspectives of safety design concepts for future SFRs in Japan based on operating experiences of Joyo and Monju, accomplishments made through conceptual design studies for JSFR and associated research and development.

2. Development Targets for Future SFRs

Although situations and concerns surrounding nuclear power and its safety-related issues have changed after the Fukushima Dai-ichi accident, the basic policy of Japan continues to pursue the promotion of nuclear fuel cycle and fast reactor development. Fast reactors and related fuel cycle are expected to reduce the amount and toxicity of high level radioactive waste and improve utilization efficiency of nuclear fuel resources. Since it is assumed that SFRs are deployed as base load electric power source, development targets of GIF and FaCT project are basically followed.

In Japan, new regulatory requirements reflecting the lessons from the Fukushima Dai-ichi accident have been set for nuclear reactor facilities and applied to all existing light water reactors (LWRs) for commercial use. They prescribe to plant operators to take design measures against natural disasters (e.g. earthquake), fire, internal flooding, loss of electric power, as well as severe accident measures, which had been voluntary efforts by the operators before. The requirements have not been completed yet for a prototype reactor Monju, however, they call for implementing necessary tasks to meet the requirements in the fast reactor development and making continuous efforts to reduce accident risks by integrating cutting-edge insights from around the world. In the meantime, JAEA has established safety requirements for Monju, in which domestic and international experts' opinions were reflected [5]. Proactive evaluations for beyond design basis accidents were conducted in the licensing procedures of Monju in 1980s, considering specific features of SFRs and comparatively less operating experience. In addition, Monju, as well as existing LWRs, is required to comprehensively enhance safety based on the lessons learned from the Fukushima Dai-ichi accident, by enhancing measures against earthquake, tsunami, and other external hazards leading to common cause failure, alternative electric power sources in dispersed locations for long-term accident response, establishment of accident management strategies including operation and management of the facilities under loss of all AC power, detailed consideration to ensure containment function, enhancement of communication and instrumentation under loss of power and so on. Furthermore, Monju is required to utilize risk information in determination of postulated initiating events and event sequences to improve the safety according to probabilistic safety assessment that has been performed for the reactor.

The SDC-TF established by GIF has been examining safety criteria to establish Gen-IV SFR SDC aiming to become global standards. Gen-IV SFR SDC refers to SSR-2/1 (design requirements for LWRs) issued by IAEA and reflects lessons learned from the Fukushima Dai-ichi accident. The criteria require taking prevention and mitigation measures against core damage caused by design extension conditions (DECs) in addition to the safety design against design basis accidents (DBAs). SDC-TF has also been establishing two SDGs providing recommendations to support practical application of SDC to actual designs. SDG on safety approach and design conditions has already been developed [6]. Development of another SDG on structures, systems and components for reactor core, reactor coolant systems and reactor containment systems is on-going.

Future safety design concepts of SFRs in Japan should fulfill these elements, i.e., development targets of GIF and FaCT, Japanese new regulation and Monju requirements, and the SDC and SDGs.

3. Safety Design Approach

Design features of SFR differ from that of LWR because SFR uses liquid sodium as coolant and runs on energy released by fission reaction by fast neutron; however, fundamental safety functions are the same with LWR, i.e. control of reactivity, heat removal from the reactor core and fuel storage, confinement of radioactive material, shielding against radiation and control of planned radioactive releases as well as limitation of accidental radioactive releases.

The future SFRs should adopt built-in safety design, in which safety systems based on experiences and proven technologies obtained through operations of Joyo, Monju and foreign reactors and safety features to respond DECs are integrated. Active components in combination with passive mechanisms should be equipped with redundancy to further improve the safety functions and its reliability. While these features are incorporated in the design conceptual or basic stage, prior consideration should also be made for accident management, for example, connection and operation of portable equipment at sites where facilities are installed. Based on the defence in depth principle, the safety design approach should have multiple design measures to prevent core damage not only against DBAs but also DECs. Referring to risk information, the approach evaluates event sequences that can result in core damage and provides effective design measures against risk dominant accident sequences. Measures against natural disasters, fire, internal flooding, and loss of off-site power should refer conditions postulated for existing domestic LWR sites SFR-specific design features should be considered because these features could change the consequences of events.

DBAs should be postulated as malfunction and failure in a single component causing severe consequences that are assumed based on design features of the reactor and experience of existing reactors. Safety functions should be verified whether the core coolability is sufficiently ensured.

DECs should include events causing core damage, most of which can be classified into two types: initiating events followed by failure in reactor shutdown and, initiating events with successful reactor shutdown but followed by failure in decay heat removal. Under a DEC, multiple failures of reactor shutdown systems or decay heat removal systems should be assumed, therefore, diversity in these systems is essential to prevent common cause failure involving loss of function of these systems.

Consequences of a postulated event vary depending on its origin (internal or external); accordingly, affected facilities differ. The vital safety features must maintain their function regardless of the origin of events and under any severe conditions. On the other hand, addition of redundancy should be done in a rational way, specifically, introduction of passive features without relying on power source or instrumentation and control systems, because it can avoid increasing not only in the number of plant equipment and whole cost but in complication of operation procedures.

As for event sequences involving failure of reactor shutdown systems, a preventive measure by utilizing inherent reactivity feedback and passive shutdown mechanism in the reactor core should be installed against core damage. Furthermore, mitigation measures against core damage should be provided so that severe mechanical energy release due to re-criticality following relocation and/or phase change of core materials is prevented and that In-Vessel Retention (IVR) to retain and cool the degraded core materials inside the reactor vessel is achieved.

Event sequences involving failure of decay heat removal systems are addressed by design measures aiming that core exposure from the coolant and complete loss of decay heat removal function are practically eliminated. Since an SFR requires no pressurization to keep sodium in the liquid phase, the coolant level to submerge the core in liquid sodium can be maintained by static structure (e.g. guard vessel), and the decay heat can be removed by the circulation of the coolant between the heat sink and the core. The reactor vessel should be designed, manufactured, and installed with highest level of reliability and should have sufficient margin to ensure the coolant level so that failure of the boundaries can be prevented even under a beyond-design-basis earthquake. The guard vessel, i.e. back-up structure to keep the coolant level, will take the same manner for the reactor vessel. Common cause failure in the both structures should be prevented. The heat removal facilities will be designed to have redundancy and diversity to maintain the decay heat removal function even under severe external and/or internal events e.g. long-term loss of off-site power and air craft impact. Heat removal using natural circulation of sodium should be proactively provided to enhance reliability of the decay heat removal function. Capability of safety systems for the decay heat removal should be extended to be able to utilize even in DECs. Application of accident management strategies should be considered as such functional extension. Additionally, alternative cooling measures independent from the safety systems will be introduced.

Design of SFR should inhibit the influence of chemical reactions of sodium on the reactor core, for instance, by implementing mitigation measures for leaked sodium combustion and by controlling sodium-water reactions in steam generators. The design should have sufficient margin to maintain the function of sodium containing components under external events including extreme earthquakes. Protection barriers and water-proof measures should be provided if needed.

Measures to prevent fuel damage and to mitigate sodium chemical reactions should be taken for fuel handling and cooling outside the reactor vessel in the same manner as the reactor coolant systems.

4. Design of Safety Facilities

The safety design concept described here is schematically shown in the figure 1.

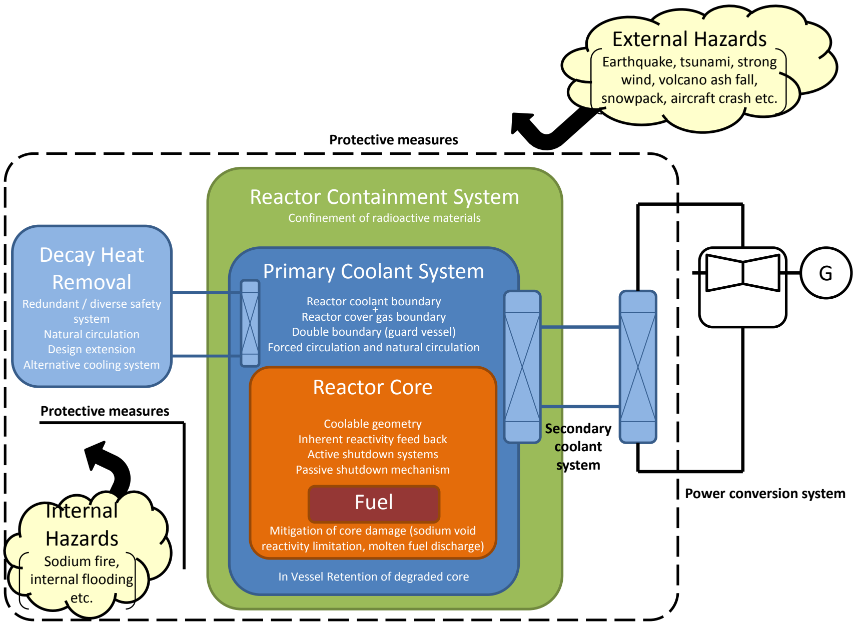

Figure 1 Safety design concept for the future SFRs in Japan

4.1.Core and Fuel

Development of the core and fuel has been carried out for the oxide fuel through Joyo and Monju in Japan. Large-scale mix oxide fueled core is one of promising concepts for demonstration reactor and commercial reactor. Fuel cladding tube should be designed as leak-tight and stability of fuel pellets should be maintained and also, thermal limit of fuel and the cladding tube is not exceeded under operational states so that these can be the first barrier against radioactive materials release. Also coolability of the core should be maintained in DBAs. Reactivity characteristics are affected along with enlargement of core size, extension of fuel burn-up and loading MA contained fuel etc. In the operational states and accident conditions, fuel burn-up and MA contents should be limited within a proper range to be acquired negative inherent reactivity feedback. Total sodium void reactivity are positive in a homogeneous MOX fuel core from a middle scale to a large scale. In DBAs, sodium boiling should be prevented. Thus, effect of sodium void reactivity by boiling would not be seen directly. However, since void reactivity is proportional to a coolant temperature co-efficient, if void reactivity is excessive, deterioration of the reactivity characteristics of the core is caused.

In DECs, core damage should be prevented by the inherent reactivity characteristics of the core and the passive shutdown mechanism such as SASS for the sequences with failure of the reactor shutdown systems. Here enhancement of spontaneous negative reactivity feedback and mitigation of positive reactivity are important because the passive shutdown mechanism requires a certain response time. Since limitation of positive sodium void reactivity leads to mitigation of the positive coolant temperature co-efficient, proper limitation of void reactivity is necessary to prevent the core damage the combination of the inherent reactivity characteristics and the passive shutdown mechanism.

In an initiating phase of the core damage considered as DECs, the positive reactivity with coolant boiling can be main cause of power excursion. On the other hand, dispersion of failed fuel provides strong negative reactivity. The positive sodium void reactivity should be limited within a range that counteracting negative reactivity effect by fuel dispersion can prevent prompt criticality.

The design for flattening, i.e., increasing core diameter to height ratio, is effective to limit the total sodium void reactivity since it enhances neutron leakage from the core. In JSFR, the void reactivity of a 1,500 MWe core is limited about six dollars by flattening the core.

Massive compaction of degraded core fuel should be prevented to avoid the prompt criticality at the core damage. For this purpose, molten fuel discharge outside from the core is effective.

In JSFR, FAIDUS, in which steel duct for the molten fuel discharge is installed in each fuel assembly, is adopted. When steep power is inserted at an early stage of the core damage, FAIDUS clearly takes effect after unprotected loss of flow initiating phase of the core with the positive sodium void reactivity. Other situations, e.g., in the case of progressing of the core damage in relatively low power level, discharge effect is expected depending on extent of fuel melting surrounding the inner duct. It is also thought of utilization of control rod guide tube (CRGT) or installation of specific assemblies for fuel discharge which is adopted in the CFV core of ASTRID [7].

In addition to above, design measures should be carried out for prevention of excess reactivity insertion in an intact core. Rapid control rod withdrawal, sudden core geometry change and large gas bubble passage through the core are thought as factors of reactivity insertion. Rapid control rod withdrawal can be managed by reducing reactivity change due to core burn-up, i.e., burn-up swing, and limiting reactivity worth of control rods, and limiting driving speed of control rods within a necessary range for operation and control. Excess deviation of each fuel assembly position should be prevented for prevention of excess reactivity insertion due to sudden core geometry change. For this reason, movable range of fuel assembly should be limited. Gaps between fuel assemblies are narrowed within a range that its re-loading does not interrupted and the whole core is restrained by the reactor structures. Large gas bubble passage through the core, which causes excess reactivity insertion should be prevented by control of gas entrainment from the free surface and prevention of the gas accumulation in the reactor coolant system.

4.2.Reactor Coolant Systems

JSFR adopts loop-type design advanced from Joyo and Monju which are also loop-type SFRs. The fundamental functions of the reactor coolant system in loop-type and pool-type SFRs are the same those are to form reactor coolant boundary and cover gas boundary, and to transfer heat generated in the core to the secondary coolant system by circulating sodium.

The reactor coolant system in an SFR operates under low pressure and has thin-walled structure to reduce thermal stress because it is exposed to high temperature during operation, in which thermal expansion and creep effect should be taken into account. The design of the system should, therefore, ensure intactness against thermal loads and seismic loads at the same time, considering the structural characteristic, especially if the size of an SFR is enlarged. Application of modified 9Cr steel with less thermal expansion coefficient to structural materials in the components and piping of the reactor coolant system can improve the structural integrity. Sufficient seismic margin is required to prevent simultaneous failures of plural components. Major components and piping of JSFR have reduced impact of earthquakes thanks to horizontal seismic isolation system combining laminated rubber bearings and oil dampers, which are installed on the base of the reactor building. They can maintain their integrity against seismic conditions in Japan. The smaller diameter of the reactor vessel than that of pool-type design is also favorable as a seismic-resistant structure [8].

4.3.Reactor Shutdown

Monju and JSFR have main and back-up reactor shutdown systems as active systems with rapid reactor shutdown function. Additionally, JSFR is equipped with Self Actuated Shutdown System (SASS) in its back-up reactor shutdown system, which is a passive control rod insertion mechanism with electro-magnets utilizing magnetic characteristic that can detach the control rods at Curie point temperature. These features can be applied in any types of reactor and core designs.

The main and the back-up reactor shutdown systems should be designed to prevent simultaneous failure due to common cause by diversifying their components (detectors, signal processing circuits, actuators and insertion mechanisms) and shapes of control rods and guide tubes to the extent practicable. The design should also consider physical separation of individual systems.

Mechanical causes leading to control rods' insertion failure should be prevented with sufficient margin because the both reactor shutdown systems are primarily composed of control rods. Hence, displacement of CRGTs located in the core and axis of the control rod drive lines supported by the reactor roof should be far below than the allowable limit to achieve control rod insertion.

Effectiveness of SASS can depend on the core reactivity characteristics and the design of reactor coolant systems; however, basically it can be applied to any reactor design concepts. Many other types of passive shutdown systems have been developed. Among them, SASS excels in its effectiveness against various kinds of transients such as loss of flow, transient over power, and loss of heat sink as well as its re-settable feature after actuation and its testability to some extent. The dominant factor of the response of SASS is a coolant transportation phenomenon occurs between the core outlet and the temperature sensing part. Our study has shown that the response can be improved by introducing flow collectors [9].

4.4.Decay Heat Removal

Decay heat removal is achieved by maintaining the enough coolant level to submerge the core and transferring the heat via coolant circulation between the core and the heat sink.

(1) Ensuring the reactor coolant level

The reactor vessel and the guard vessel of JSFR have no nozzles for piping connection. They are manufactured under strict quality control and be confirmed that non-negligible initial defect does not exist by inspection. No interaction exists between the two vessels under normal operation; consequently, no load causing damage is affected to the guard vessel. The vessels are supported by walls of reactor pit, which are made of steel plate-reinforced concrete and thus rigid structure with high tolerance against seismic loads. The internal concrete of the structures are cooled to avoid exceeding the operational limit temperature. With such design, double leakage from the vessels is expected to be practically eliminated. In addition to that, the primary piping and components in JSFR adopt double boundary structure. The gap space between them is limited and partitioned to maintain the coolant level as a design measure to cope with a leakage accident in the reactor coolant boundary.

In JSFR, cut-off of sodium flow in the loop (i.e. siphon break) in the primary coolant system is prevented by the above design measures. The coolant level for circulation can be also ensured, even if a double leakage from the primary piping and the guard pipes occurs, which is postulated as a DEC because of its relatively complicated structure of guard pipes connected to small diameter pipes for inner gas sampling and pressure control.

(2) Transportation of decay heat

Two primary reactor auxiliary cooling systems (PRACS) and one direct reactor auxiliary cooling system (DRACS) are introduced in JSFR for decay heat removal. These systems don’t rely on the secondary coolant system and steam-water system and contribute to simplify the system design. Using a feature of loop-type design that can be located IHX and a reactor vessel separately, JSFR has superior natural circulation capability of sodium by having a large height difference between the core and the primary heat exchanger of PRACS. For a pool-type, PRACS is favorable to remove decay heat by the natural circulation; when DRACS is employed in it, for instance, inter wrapper flow and penetration-type design can be introduced to ensure the natural circulation. In any cases, the natural circulation capability for the decay heat removal systems should be proactively incorporated into the design to the extent practicable depending on the characteristics of the reactor coolant system.

For practical elimination of complete loss of decay heat removal function, an SFR should be designed to have redundancy and diversity together with separation and independence of equipment in the decay heat removal system to make the system available even under severe plant conditions including beyond-design-basis external events.

Aside from these design measures, functional extension of the system, such as extension of the heat removal capacity, should be considered. Plus, the design should be applicable to accident management strategies, for example, manual operation of air coolers for air flow rate control by plant personnel under station black out.

Alternative cooling measures, independent from the decay heat removal system, should be provided; a reactor auxiliary cooing system separated from the main cooling systems can be a good option.

IVR in a core damage accident can be achieved by ensuring reactor coolant circulation path(s) and providing systems serve as heat removal in a reactor vessel depending on the damage condition, which should have resistance to pressure and thermal loads generated as consequences of core damage.

4.5.Measures against Sodium Chemical Reactions

(1) Mitigation of sodium leak and combustion

The double boundary structure is effective to control combustion of leaked sodium. Although the structural measure with nitrogen-filled and limited gap space between the structures is implemented in the primary coolant system from the perspective of ensuring the sodium level, it is also advantageous to the combustion control. The measure can be applied to facilities containing sodium (the secondary coolant system, the decay heat removal system, and the like). It allows to detect initial small leaks by sampling the atmosphere in the gap.

Large-scale sodium leaks and combustions under severe conditions beyond design basis should be prevented. The whole building containing sodium-related facilities, therefore, should have rigid structure with seismic isolation so that failure of the facilities can be avoided even in case of extreme earthquake. The secondary systems and a steam generators should be protected from external missiles including airplane impact. For this purpose, outer walls of buildings enveloping those systems and components should be strengthen. Selection of the height of ground level, provision of protection dikes, and enclosure of the buildings with water-proof design are carried out against water ingress caused by water-related events like a tsunami. Water systems and sodium systems are located separately and provided physical barriers against internal flooding from water supply facilities.

Furthermore, as design measures for a DEC, concrete floors and walls of rooms where sodium contained components and piping are installed are covered by steel liners to prevent hydrogen generation induced by large-scale sodium-concrete interaction.

(2) Mitigation of sodium-water reaction

Reliability of steam generators is one of crucial issue not only for higher plant availability but also for operational safety. Taking the past development and operation experiences of steam generators into account, proper design and material selection should be made to prevent steam-water leak from heat transfer tubes of a steam generator. Application of double walled heat transfer tube is a possible measure. Effective inspection technology should be developed and applied to periodically check the integrity of heat transfer tubes.

Sodium-water reaction caused by steam-water leak from heat transfer tubes of a steam generator should be safely terminated by detecting the leak, shutting off the steam-water and releasing the pressure of the secondary coolant system to prevent large-scale failure propagation, postulating a wide range of leak rates from very small leak to double-ended break of a tube. Large-scale leakage equivalent to double-ended break of a tube is suppressed by the pressure release through a rupture disk installed in the secondary coolant system side. Middle-scale leakage needs to be controlled to avoid event progression to large-scale failure due to overheat rupture. Small-scale leakage should be detected and terminated in the early stage. Water leak rates for design-basis leak should be set considering such detection and mitigation measures. The design should prevent rupture of a heat transfer tube of IHX that plays a role of a boundary between the primary and the secondary coolant systems against pressure loads generated by a design-basis water leakage accident. Moreover, the design should have sufficient margin to prevent such boundary failure of the IHX tube even under multiple failures of the detector and the mitigation systems.

4.6.Containment Functions

An SFR can keep the reactor core submerged in sodium by maintaining the sodium level with static components such as the guard vessel in the case of coolant boundary failure. Coolant vaporization doesn't occur in the containment; therefore, unlike LWRs, high pressure-resistant containment vessel and associated facilities for coolant injection are not required. In addition, a part of the containment functions can be served by the reactor vessel and guard vessel with IVR strategy. The main function of the containment system is to confine the radioactive materials inside and to protect the reactor coolant system from aircraft impact and other external events.

Event sequences leading to core damage are evaluated based on a best estimate analysis. A reactor vessel and necessary structures and systems for IVR (e.g. core catcher) should be designed to be effective for bounding consequences of various event sequences and their uncertainty.

Although gaseous fission products (FP) and sodium may be released into the containment vessel while IVR is achieved, no cause of hydrogen generation exists in the reactor core (Hence hydrogen-containing materials, for example ZrH, should not be used in the core.). The containment function should surpass the consequences of sodium combustion and heat generated by gaseous FP. For hydrogen generation due to sodium-concrete interaction, the contact between them should be prevented by taking the preventive measures in the design.

Furthermore, effect of liquid sodium that can retain volatile FPs, (iodine, cesium, and so on) and effect of sodium aerosol plate-out in the containment atmosphere should be appropriately quantified and incorporated into the design of containment vessel. Releases of noble gas FP is controlled by limiting the leak rate from the containment, and of volatile FP is dealt with an emergency gas treatment system.

5. Conclusion

In Japan, development SFRs is continued for commercial use as a sustainable base load power source in the future even though situations and concerns surrounding nuclear power and its safety-related issues have changed after the Fukushima Dai-ichi accident. Large-scale SFRs with oxide fuel is the subject of this paper. Development targets of GIF and FaCT project are taken into account. The new regulatory requirements in Japan and SDC and SDG for Gen IV SFR are referred. Based on operating experiences in Joyo and Monju in addition to accomplishments made through the development of design concept of JSFR and related techniques, this paper shows current perspectives of the safety design concept for Japanese future SFRs featuring safety design issues for rector core and fuel, coolant systems, reactor shutdown, decay heat removal, measures against chemical reactions of sodium, and containment function. In light of what we have learned from the Fukushima Dai-ichi accident, it is essential to make efforts to improve safety by incorporating comprehensive, effective, and rational severe accident measures into the design, although safety evaluations for beyond-design-basis core damage accidents have been conducted before the accident.

6. Acknowledgement

Present study includes the result of "Technical development program on a fast breeder reactor, etc." entrusted to Japan Atomic Energy Agency by Ministry of Economy, Trade and Industry of Japan (METI)

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