CURRENT STATUS AND PROSPECTS OF R&D ON GENERATION IV SFR SAFETY AND OPERATION PROJECT

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I. INTRODUCTION

Generation IV Nuclear Energy Systems are being developed under the initiative of Generation IV International Forum (GIF) begun in 2000. The SFR was selected as one of the promising concepts together with other five concepts.[1] The System Arrangement for the International Research of 15 February 2006 constitutes a framework to carry out the research and development work necessary to establish the viability and to optimize the performance of the GIF SFR, and to facilitate (but not undertake) the eventual demonstration of the SFR system. For the purposes of coordinating the collaborative R&D among the member countries, the Safety and Operation Project Management Board (SOPMB) was organized under the SFR SSC based on the SFR System Arrangement. The member countries of SOPMB are France, Japan, Republic of Korea, and the United States of America. The Project Arrangement was concluded in June 2009 for the implementation of collaborative R&D. This project includes 1) analyses and experiments that support approaches and assess performance of specific safety features, 2) development and verification of computational tools and validation of models employed in safety assessment and facility licensing and 3) acquisition of reactor operation technology, as determined largely from experience and testing in operating SFR plants. This paper describes the current status and prospects of R&D on SFR safety and operation project.

II. GOALS AND DEVELOPMENT TARGETS RELATED TO SAFETY AND OPERATION

II.A General Goals for Generation IV Nuclear Energy Systems Related to Safety and Operation

Three goals for the Generation IV nuclear systems have been defined in the safety and reliability as listed below.[1]

- *Safety and Reliability-1, Generation IV nuclear energy systems operations will excel in safety and reliability.*
- *Safety and Reliability-2, Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.*
- *Safety and Reliability-3, Generation IV nuclear energy systems will eliminate the need for offsite emergency response.*

II.B Key SFR Development Targets on Safety

To effectively meet the Generation IV systems goals, the SFR R&D plan will focus on particular technology development efforts. Critical issues for the SFR technology are examined for achieving the enhancement of safety. Some key development targets for the SFR are summarized below.

With regard to reactor safety, technology development efforts focus on two general areas: assurance of passive safety response, and techniques for evaluation of bounding events. Advanced SFR designs exploit passive safety features to increase safety margins and to enhance reliability.

The system behavior will vary depending on system size, design features, and fuel type. R&D for passive safety will investigate phenomena such as axial fuel expansion and radial core expansion, and design features such as Self-Actuated Shutdown Systems (SASS) and passive decay heat removal systems. The ability to measure and verify the performance of these passive features must be demonstrated. Associated R&D will be required to identify bounding events for specific designs and investigate the fundamental phenomena necessary to prevent severe accident progression.

The favorable passive safety behavior of fast reactors is expected to reduce the probability of severe accidents with potential for core damage. Nevertheless, design measures to mitigate the consequences of severe accidents are being considered. This approach is consistent with the defense-in-depth philosophy of providing additional safety margin beyond the design basis. A common safety approach incorporating the physical and chemical characteristics of the materials handled in the reactor (chemical activity and radio-toxicity, etc.) and unique SFR design features and phenomena should be established. The goal is to render the risk of installing SFR systems much lower than the risk of energy alternatives. Achieving this level of safety should result in licensing and regulatory simplifications that may in turn result in reduced system cost. To do this, probabilistic safety evaluations will be needed to identify design tradeoffs that assure very high levels of public health and safety.

III. TECHNOLOGICAL SAFETY ISSUES INDENTIFIED BY GENERATION IV SFR DESIGNS

Three reactor systems have been proposed as options for Generation IV SFR system. These concepts are based on the significant knowledge and experience accumulated so far, but they also adopt innovative technologies.

- A large size (600 to 1 500 MWe) loop-type sodium-cooled reactor with mixed uranium-plutonium oxide fuel, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors.[2]
- A medium or large size (600 to 1 500 MWe) pool-type system also supported by a fuel cycle.[3]
- A small size (50 to 150 MWe) modular-type sodium-cooled reactor with uranium-plutonium-minor-actinide-zirconium metal alloy fuel, supported by a fuel cycle based on pyrometallurgical processing in facilities integrated with the reactor.[4]

Assurance of passive safety response of a medium or large size pool-type SFR and a small size modular-type SFR in bounding events or Anticipated Transient Without Scram (ATWS) is one of the predominant developmental issues. For accurate predictions of the passive response of the reactor, it is necessary to develop advanced modeling of the transient thermal-hydraulic, and mechanical behavior of the reactor components that are the basis for inherent reactivity feedback estimates. To enhance the reactivity feedback models, an accurate three-dimensional core thermal-hydraulic model should be developed and fully linked to the reactor physics model. Validation of these models is an integral part of the development process.

The characteristics of metallic fuel behavior in a medium or large size pool-type SFR and a small size modular-type SFR during a hypothetical core disruption accident are an issue to be studied to ensure the expected low probability of the CDA occurrence in a metal fuel core and to prepare an adequate design for managing severe accidents.

The liquidus, solidus, and mobilization temperatures of metallic fuel and steel mixtures will be investigated. Molten fuel relocation behavior is also an important research area for the development of the predictive models.

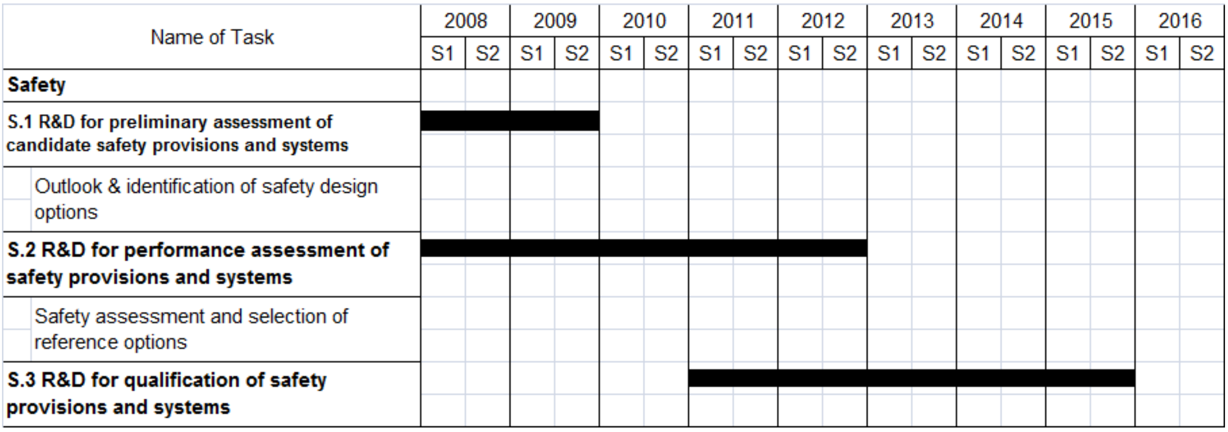

Figure 1: Safety Work Package Schedule.

In the safety approach for a large size loop-type SFR with mixed uranium-plutonium oxide fuel, the technology gaps center around two general areas: assurance of passive safety response, and the technology for evaluation of bounding events. With appropriate design features, passive safety response can be assured in large size reactors, and the plant utilizes passive safety measures to increase its reliability. R&D for passive safety will investigate relevant thermal-hydraulic and mechanical phenomena and design features such as self-actuated shutdown system and passive decay heat removal systems. R&D for bounding events will investigate the fundamental phenomena and design features for severe accident consequence mitigation. One approach is to introduce design measures to enhance the fuel discharge. The EAGLE[5] project is aimed at demonstrating the effectiveness of inner duct structure to enhance the fuel discharge without the formation of large molten fuel pool and to obtain an insight for the prevention of severe re-criticality in a sodium-cooled MOX fuel core.

IV. OUTLINES OF SAFETY AND OPERATION PROJECT

IV.A Safety

The safety design and its assessment must be coordinated with the overall design activity.

In particular, the time schedule of safety R&D work must also take place in concert with corresponding design work.

As shown in Figure 1, the safety project includes three stages corresponding to three work packages (WPs): (S1) R&D for preliminary assessment of candidate safety provisions and systems, (S2) R&D for performance assessment of safety provisions and systems, and (S3) R&D for qualification of safety provisions and systems. Each stage includes R&D of passive/active safety issues, severe accident issues and framework and methods of safety architectures that are necessary to support the system integration and assessment.

Work package S1 provides preliminary assessment of candidate safety provisions and systems introducing innovation. Because innovative and/or new design features will be adopted, critical issues in those features should be reviewed from the viewpoint of safety. Innovative/new features include new types of fuel such as recycle transuranic fuels (oxide, metallic, nitride, and carbide), and new types of plant designs aiming at economic advantages. Various passive/active safety features are considered in the designs, such as the SASS, the Passive Decay Heat Removal Circuit (PDRC), and the inherent shutdown and natural circulation shutdown heat removal features. Passive/active safety analysis tools will be developed. The feasibility of the innovative safety features will also be reviewed, and severe accident manage–ment measures will be assessed. Results of severe accident tests for MOX fuel and inherent safety tests for metallic fuel will be utilized for safety review.

Work package S2 provides performance assessment of safety provisions and systems in order to evaluate whether the design meets the safety requirements. Various R&D activities are planned to prepare the analysis tools needed for the performance assessment of the safety design options. Passive safety analysis tools will be validated through Joyo ATWS simulation tests. Additional results of severe accident tests for MOX fuel will be obtained and used for the validation of the models in severe accident analysis tools, including the SIMMER code. Post accident heat removal and in-vessel retention for metallic fuel systems will be investigated and assessed to establish Post Accident Materials Relocation (PAMR)/Post Accident Heat Removal (PAHR) scenarios. All of these R&D results will be used for establishing accident scenarios in the performance assessment of the design options. And a performance assessment for the proposed design of the safety architectures will be implemented.

Work package S3 provides a technical basis for safety assessment aiming at design optimization. A safety assessment of the reference designs will be interactively continued in order to refine the designs. For this purpose, further R&D for qualification of passive/active safety and severe accident analysis tools are considered, and the results will be reflected in the safety assessment.

IV. B Reactor Operation and Technology Testing

Operation technologies, experiments/ testing for computational tools validation, and demonstration of innovative technologies by using existing SFR plants are essential for SFR design and technology viability demonstration. This project is aimed at gathering the contributions from existing reactors to SFR design. Such contributions include: data acquisition, code validation, operation feedback (on safety, operation and maintenance), and innovative technology testing. The time schedule of Reactor Operation and Technology Testing (ROTT) project is shown in Figure 2. Additional ROTT tasks should be added later as the SFR project makes progress. (Utilization of Monju start-up within this testing program will be discussed when the schedule for these events become better determined.)

Tasks related to System Integration & Assessment (SIA) project are performed in Work package Op.1 in order to acquire validation data through the Phenix operation period that includes end-of-life tests and lifetime extension project, Monju start-up tests, and decommissioning of SPX. Plant behavior experiments will be performed at Phenix at its end-of-life time. Test data of thermal hydraulics including natural convection, neutronics, and investigation of negative reactivity transient will be used for validation of the MARS-LMR and SSC-K codes. Thermal-hydraulic data from Monju start-up tests will be used to verify and validate both computer models and design capabilities. Testing and validation could include design capabilities such as natural convection. The CEA CATHARE-ML general system code for a loop type reactor will be validated by using Monju transient recordings during commissioning tests and operation.

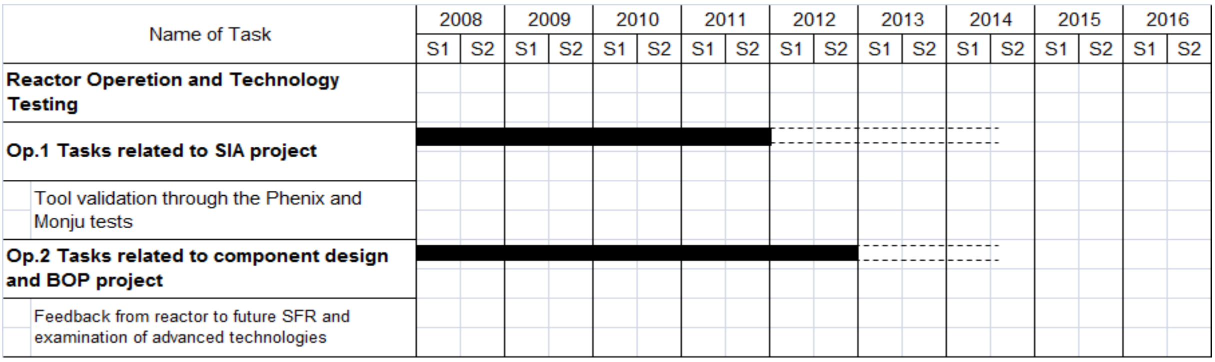

Figure 2: Reactor Operation Technology Testing Work Project Schedule.

The US will share the results from validation of new fuel assembly thermal hydraulic models with EBR-II Shutdown Heat Removal Test (SHRT) data[6]. The analyses will employ detailed sub–channel temperature measurements from the XX09 instrumented fuel assembly. The new models are being implemented in multiple computer code frameworks and solution architectures that include traditional subchannel modeling, commercial CFD modeling, and developmental advanced simulation capabilities. In order to contribute to a better design of future SFRs, the knowledge of decommissioning obtained from LMFR in the aspects of both technology and economics will be accumulated for integration and feedback.

Tasks related to Component Design and Balance-Of-Plant (BOP) project are performed in Work package Op.2 in order to issue recommendations for future SFRs in the areas of In-Service Inspection (ISI) methodology, sodium quality control and monitoring, radioactive impurities behavior, and tritium transfer based on the experience of existing reactors. Evaluation of ISI methodology will be based on results of its application to existing reactors. Analysis of the coolant chemistry operating feedback from sodium circuits in Phenix and Monju will be carried out during start-up and subsequently in steady-state. Additionally, tritium measurements at the Monju reactor will be analyzed to improve understanding of the reactor's tritium source and its subsequent transfer and accumulation on the reactor system and components. The improved understanding of radioactive material transport (activated corrosion products, fission products) and other behaviors in the primary circuit of an SFR that results from these activities will help develop design requirements for impurities control, oxygen control, and sodium purification system in future SFRs.

V. DETAILED DESCRIPTION ON SOME TOPICS IN THE SO PROJECT

In this section some topics in the SO Project are described in more detail.

In the US, research activities sponsored by the U.S. Department of Energy focus on fulfilling the Generation IV safety and reliability goals with a technology development program that includes demonstrative reactor concept safety performance evaluations. These analyses employ models validated with experimental data, and emphasize defense-in-depth for accident prevention with assurance of passive safety response, and for accident consequence mitigation with characterization of phenomena. Within the framework of the SOPMB, shared activities include US testing experience, documentation of design measures for accident prevention and consequence mitigation, passive safety performance assessment in conceptual designs, and developmental methods for uncertainty quantification.

Recently, analyses of passive safety performance in oxide and metallic-fueled reactor concepts for 1 000 MWt and 2 000 MWt core sizes have indicated the potential for prevention of accident progression in ATWS. The modeling employed in these analyses is based on reactivity feedback and decay heat removal mechanisms demonstrated in EBR-II SHRT conducted previously.[6] The analysis results show that significant margins to reactor upset and damage (coolant boiling, cladding failures, fuel melting) can be assured by selection of appropriate design features, promising enhanced safety performance and improved economics.

In the Korea, a large scale sodium thermal-hydraulic test facility sponsored by the Department of Education, Science and Technology has been designed for verification of the design concept of the PDRC in a medium or large size pool-type SFR, focusing on assessing its cooling capability during the long and short term periods after reactor trip. Starting with the basic design of the test facility in 2008, its installation is scheduled to be completed by the end of 2011. The main experiments will commence in 2013 after the startup test in 2012.

The main test section of the experimental facility is composed of a primary heat transport system and a passive decay heat removal circuit which are scaled-down from the target design. The test section includes all major components in the primary heat transport system reflecting the real configuration. The preliminary concept of the main test section is shown in Figure 3.

Auxiliary fluid systems such as an intermediate heat exchanger gas cooling system, a sodium supply/purification system, a heat loss compensation system, and a gas supply system are included in the experimental facility. In order to represent important thermal-hydraulic phenomena in the PDRC as well as the reactor system, the main test section has been designed complying with proper scaling criteria for geometric, hydrodynamic and thermal similarities. Overall scaling of the facility is 1/125 for volume and 1/5 for height. The reactor vessel height and diameter are about 3.6 m and 2.3 m, respectively. The reactor core is simulated by electrical heaters of 1.9 MW capacity which corresponds to a 7% of the scaled full power. Sodium is used as a working fluid and its inventory of the main test section is approximately 13 tons. Operating temperatures of the reactor system are preserved in the experiment.

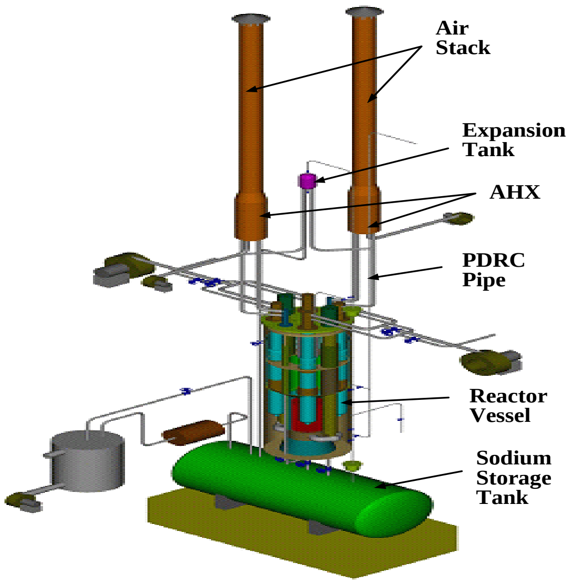

Figure 3: The preliminary concept of the PDRC test section

In the test, the natural circulation cool-down capability of the PDRC in conjunction with the reactor system will be investigated for various design basis events.

In the EAGLE-1 program,[5] several in-pile and out-of-pile tests were conducted under a co-operation between JAEA and National Nuclear Center of Republic of Kazakhstan. One of the main objectives of these tests was demonstration of effectiveness of the specific design concept to eliminate the severe re-criticality events in the course of core disruption accidents. Another important objective was acquisition of basic information on early-phase relocation of molten-core materials toward cool regions surrounding the core, which would be applicable to various core design concepts.

Figure 4 shows schematic of a typical in-pile test apparatus of the EAGLE-1 program. The geometry of this test apparatus is corresponding to a design concept equipped with a “discharge duct” within each fuel sub-assembly. The discharge duct of 2mm-thick stainless steel filled with liquid sodium was placed at the central part, and was surrounded by 75 UO2-fuel (BN350-type) pins with 400mm fissile height giving total fuel amount of ~8 kg. The test ID1 (Integral Demonstration test 1) was conducted with this test apparatus in IGR (Impulse Graphite Reactor). It was intended to produce a molten fuel-steel-mixture pool with the trapezoidal power diagram simulating the hottest part of the degraded core in an Unprotected Loss of Flow accident. This result showed a significant potential of core-material relocation even under a relatively low pressure difference (up to ~0.12MPa). These experimental data strongly suggested early fuel discharge with the inner duct equipped fuel subassembly design thereby eliminating large molten-pool formation which was the entry condition for severe re-criticality events. JAEA is presently conducting the EAGLE-2 program focusing on the long-term behavior of the degraded core with its stress on its coolability.

After 35 years operation, the 250 MWe (140 MWe since 1993) sodium cooled fast reactor Phenix was shut down on March 6th 2009. Before the decommissioning, the end of life tests are carried out during 2009 to collect relevant results in the fields of core physics, thermal-hydraulics fuel behavior under accidental conditions and negative reactivity transients occurred in ‘89-90’. The tests in the core physics field are: individual subassemblies reactivity worth, sodium void effect, control rod worth measurements by different methods on a low reactivity core and decay heat measurements. The thermal-hydraulics tests cover natural convection regimes in primary and secondary circuits and asymmetrical regimes in the primary vessel. One test concerns partial fuel melt in some experimental fuel pins.

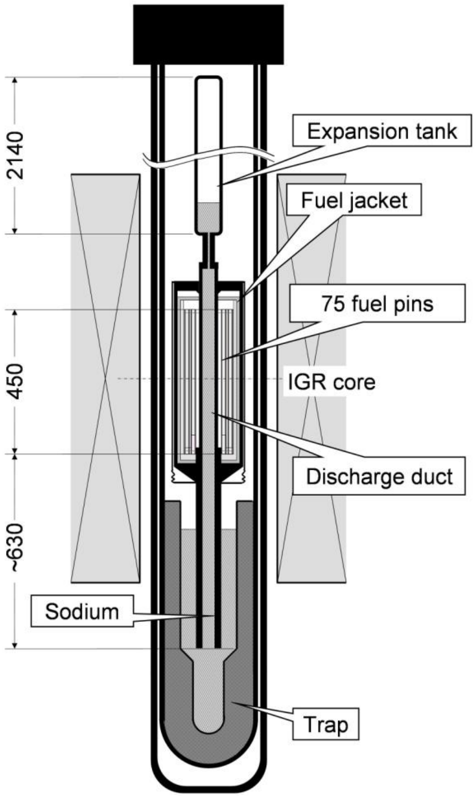

Figure 4: Typical EAGLE In-Pile Test Apparatus

Two tests are performed to help the comprehension of negative reactivity transients during the normal operation of the reactor, the first one concerns the neutronics and thermal-hydraulics coupling of an experimental moderated carrier with several adjacent blankets and the second is an artificially provoked core flowering to measure the reactivity effect. Two of these tests, thermal-hydraulic asymmetrical regimes ad core flowering are included in the scope of the SFR Safety & Operation Project. Main objectives are respectively, the validation of CATHARE ML system code and the understanding of safety related issues of the core neutronics/thermal-hydraulics/mechanics couplings.

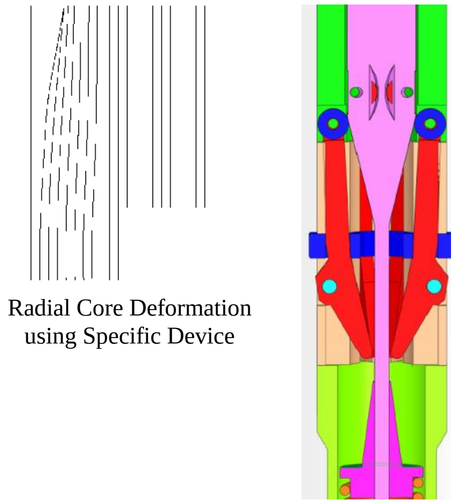

Figure 5: Phenix Core Flowering Test

VI. CONCLUSION

International collaborative R&Ds in the field of Safety and Operation Project are being implemented in the framework of GIF. The R&Ds include 1) analyses and experiments that support approaches and assess performance of specific safety features, 2) development and verification of computational tools and validation of models employed in safety assessment and facility licensing and 3) acquisition of reactor operation technology, as determined largely from experience and testing in operating SFR plants. Implementing these R&Ds with other project R&D results will effectively achieve the development of Generation IV SFR.

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