# Status of Sodium Cooled Fast Reactor Development Program in Korea

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**Abstract**. The Korea Atomic Energy Commission (KAEC) authorized the R&D action plan for the development of the advanced sodium-cooled fast reactor (SFR) and the pyro-processing technologies to provide a consistent direction to long-term R&D activities in December, 2008. This long-term advanced SFR R&D plan was revised by Korea Atomic Energy Promotion Council (KAEPC) in November 2011 in order to refine the plan and to consider the available budget for SFR. The revised milestones include specific design of a prototype SFR by 2017, specific design approval by 2020, and construction of a prototype generation IV SFR (PGSFR) by 2028. The prototype SFR program includes the overall system engineering for SFR system design and optimization, integral V&V tests, and major components development. Based upon the experiences gained during the development of the conceptual designs for KALIMER, the conceptual design of PGSFR has been carried out in 2012 and has been performing a preliminary design since 2013. The first phase of the development of PGSFR has been completed at the end of February 2016 and now going toward the second design phase in 2016. All the design concepts of systems, structures and components (SSCs) have been determined and incorporated into the preliminary safety information document (PSID), which includes basic design requirements, system and component descriptions, and the results of safety analysis for the representative accident scenarios. The PSID will be a base material for a pre-review of the PGSFR safety. The target of the second phase of PGSFR design is to prepare a specific design safety analysis report (SDSAR) by the end of 2017. The SDSAR is equivalent to the conventional preliminary safety analysis report (PSAR) but without the specific site information of the plant. To support the design, various R&D activities are being performed in parallel with design activities, including V&Vs of design codes and system performance tests.

**Key Words**: Sodium cooled fast reactor, Metal fuel, Generation IV, Pool type

## 1. Introduction

The light water reactor (LWR) has been operated and played a significant role of a stable electricity supply and economic growth in Korea since 1978. There are 21 LWRs and 4 CANDU type reactors currently in operation, 3 PWRs under construction and 4 additional PWRs planned by 2029 based upon the 7th National Electricity Demand and Supply Plan. The construction of nuclear power plant also support the Paris Agreement on new climate change agreed at COP21 in 2015, which targets nuclear share of 29% by 2035.

One of serious obstacle in constructing LWR is a problem on spent nuclear fuel (SNF) management because of high radio-toxicity and long half-life of SNF. Annually about 760 tons of SNF are discharged from PWRs and total stored SNF amounted to14,468 ton as of December 2015. In this context, the Public Engagement Commission on Spent Nuclear Fuel (PECOS) made 10 recommendations on future spent nuclear fuel management policy in 2015. One of key recommendation among them is to establish an R&D plan for volume and toxicity reduction of SNF. The SNF problem has been a common concern among the countries having utilized nuclear energy for a long time or having a plan to extend the utilization of nuclear energy. A SFR has been widely recognized as a technical alternative to effectively manage the SNF owing to its transmutation capability of long lived radio-toxic nuclides included in the SNF. It can be accomplished by using abundant high energy excessive neutrons in the core. For this reason, the SFR development plan is always accompanied with the policy for the extension of nuclear energy in many countries.

The Long-term Development Plan for the Future Nuclear Energy Systems was authorized by the Korean Atomic Energy Commission (KAEC) in 2008 and updated by the first Korea Atomic Energy Promotion Council (KAEPC) in 2011, which includes a construction of a prototype SFR by 2028, with preparation of preliminary safety (PSID) by 2015 [1], issue of specific design safety analysis report by 2017 and its approval by 2020.

In July 2016, the sixth KAEPC has approved the Basic Plan for High-level Radioactive Waste Management, which includes the secure of underground research laboratory, interim storage and final disposal sites of high-level waste, enactment of special law on the procedure for high-level waste management. The KAEPC has also approved the Demonstration Strategy of the Future Nuclear Energy System Development. The strategy provides basic directions for demonstration of the pyro-processing and sodium-cooled fast reactor technologies.

The national project to develop the Prototype Gen IV Sodium-cooled Fast Reactor (PGSFR) was initiated to achieve the national mission in 2012. For this, Sodium Cooled Fast Reactor Development Agency (SFRA) dedicated to the PGSFR development was established in the mid of 2012. Research and development works of the PGSFR project are mainly carried out by KAERI, KEPCO E&C and Doosan Heavy Industry. KAERI is in charge of the design and the validation of nuclear steam supply system (NSSS) and fuel development, and KEPCO E&C is responsible for balance of plant system design. Doosan Heavy Industry involves in the evaluation of a mechanical design and fabrication of major components. KAERI is closely working with Argonne National Laboratory (ANL) under an agreement on the joint development program approved as a work-for-others (WFO) contract. ANL supports KAERI with their experiences in SFR development and is jointly working on the developments of codes for fuel rod performance analysis and severe accident analysis. The collaborative activities through a generation IV international forum (GIF) and IAEA CRP also support the R&D activities for the PGSFR development.

## 2. Status of PGSFR Development Project

### 2.1. Design Status of PGSFR

Main goal of the PGSFR development is to demonstrate transmutation capability of transuranic (TRU) nuclides which are the major long-lived toxic elements included in the LWR spent nuclear fuel. The high level of safety and the efficient electricity generation are also one of the requirements of the PGSFR [2].

The initial core of the PGSFR is loaded with low enriched uranium metal fuel (U-10% Zr) for a reactor performance demonstration and as a driver fuel for TRU fuel irradiation test as shown in TABLE I. Several lead test rods (LTRs) and lead test assemblies (LTAs) containing TRU fuel recycled from LWR (LWR-TRU) will be loaded and qualified during this period. After qualification of LWR-TRU fuel, the U-TRU-Zr fuel will be loaded into the core as a batch. Until this stage, the back-end fuel cycle will be kept as once-through without self-recycling. The in-reactor performance of the self-recycled TRU fuel (MTRU) will also be demonstrated during the LTRU core operation. Then finally, the fully closed fuel cycle with the self-recycling will be demonstrated in the MTRU core.

TABLE I: EVOLUTION OF PGSFR CORE

| U core | LTRU core | MTRU core |
|---|---|---|
| • U-10%Zr fuel<br>• Open fuel cycle<br>• LWR-TRU (LTRU) fuel demonstration<br>• LTR and LTA test zone installation | • U-TRU-Zr fuel<br>• LWR-TRU equilibrium<br>• Open fuel cycle<br>• LWR-TRU and self TRU mixed (MTRU) fuel demonstration | • U-TRU-Zr fuel<br>• TRU core equilibrium with self-recycling + LWR recycling (MTRU fuel equilibrium) |

FIG. 1 shows the key design features and schematic diagram of the PGSFR [3]. The overall design features can be summarized as metal fueled, pool type sodium cooled fast reactor with active and passive decay heat removal and shutdown systems.

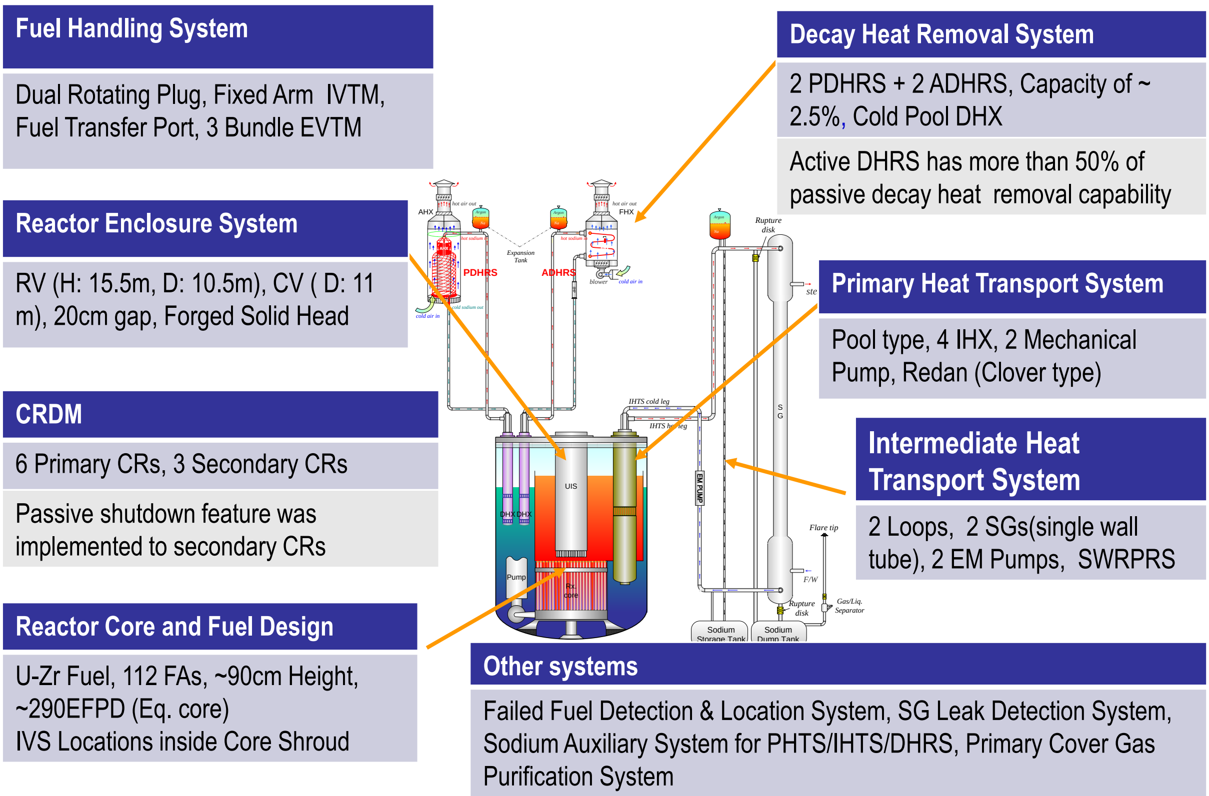


*FIG. 1. Key design features of PGSFR*

Total 217 fuel rods are arranged into the fuel sub-assembly with hexagonal configuration. Total 112 fuel sub-assemblies are loaded into the core in the hexagonal configuration. The active core height is about 90 cm. The cycle length of uranium equilibrium core is about 290 effective full power days (EFPDs). There is neither an axial nor radial blanket to prevent additional TRU production in the blanket region. The active core is directly faced into the steel reflector.

The primary heat transfer system (PHTS) of the PGSFR is a pool type. All the structures and components of PHTS, 4 intermediate heat exchangers (IHXs) and 2 mechanical pumps are submerged into a large sodium pool confined by double vessels; reactor vessel and containment vessel.

The intermediate heat transfer system (IHTS) consists of 2 loops with 2 steam generators. The annular linear induction pump is used in the IHTS and IHTS is connected into the sodium-water reaction pressure relief system (SWRPRS) to prevent over-pressure of IHTS loop when sodium-water reaction occurs in the steam generator. The IHTS pipe inside containment is double-walled. The gap between inner and outer pipes is filled with inert gas and continuously monitored by diverse leak detectors as well as the gap between reactor and containment vessels.

The decay heat is rejected to the atmosphere by decay heat removal system (DHRS). DHRS consists of 4 trains: 2 active decay heat removal systems (ADHRs) and 2 passive decay heat removal systems (PDHRs). The sodium to air heat exchangers are a finned tube type for the ADHRSs and a helical coil type for the PDHRSs to satisfy a diversity and redundancy requirements for the safety system. The active functions of ADHRSs are provided by air blower. The active circuit has also passive function by the natural circulation of sodium and air and its passive heat removal capacity of the active circuit is more than 50% of designed heat removal capacity even when air blowers are not operable.

The PGSFR has the independent and diversified safety shutdown systems which consists of 6 primary control rods (CRs) and 3 secondary shutdown rods. A passive shutdown mechanism is implemented into the secondary shutdown rods for additional shutdown capability during beyond design basis accidents. The major design parameters of the PGSFR are listed in Table II.

TABLE II: DESIGN PARAMETERS OF PGSFR

| Parameter | Value |
|---|---|
| Core power [MWt/MWe] | 392.2/150 |
| Coolant temperatures (inlet/outlet) [℃] | 390/545 |
| Total core flow rate [kg/sec] | 1,990 |
| Fuel type (initial and transition) | U-10%Zr |
| Cycle length [EFPD] | 290 |
| Fuel cladding material | FC92(FMS) |
| Number of batch (inner/outer) | 4/5 |
| Active core height [cm] | 90 |
| Pitch to diameter ratio (P/D) | 1.14 |
| Total heavy metal inventory [ton] | 7.33 |
| Discharge burnup (avg./peak) [MWd/kg] | 66.1/104.7 |
| Average/peak linear power [W/cm] | 159.7/323.7 |
| Sodium void reactivity at EOEC [pcm] | -900 |

### 2.2.Safety Analysis Results of PGSFR

The safety analysis for the first phase design has been carried out for the representative bounding accident scenarios [4]. The event classification and corresponding safety design acceptance criteria have been established in terms of cumulative damage fraction (CDF) and temperatures. All the results of the safety evaluation satisfy the acceptance criteria with a sufficient margin. The list of the representative events and the acceptance criteria are given in Table III.

Table III. Event scenarios included in PSID and acceptance criteria

| Classification | Events | Remarks |
|---|---|---|
| Reactivity Insertion | - Maximum velocity withdrawal of single control rod (DBA1)<br>- Reactivity insertion and pump trip by SSE | - Acceptance criteria<br>- AOO: $CDF_{\Sigma AOO} < 0.05$<br>- DBA1: $CDF_{event} < 0.05$<br>- DBA2: Fuel T < Solidus T, Cladding T < 1075ºC, No bulk sodium boiling<br>- DEC: Fuel T < Solidus T, No bulk sodium boiling |
| Undercooling | - Loss of Flow (LOF-AOO)<br>- Loss of Heat Sink (LOHS-AOO)<br>- One Pump Seizure (OPS-DBA1)<br>- PHTS Pipe Break (PB-DBA2)<br>- SBO (SBO-DBA2)<br>- 5 DEGs of SG tubes (DBA2) | |
| Increase or decrease of PHTS Inventory | - Reactor vessel leak (DBA2) | |
| DEC | - ATWS (UTOP, ULOF, ULOHS) | |
| HCDA | - Mechanical energy release evaluation | - Mechanical energy release during whole core melting (100$/s) ~ 16.1 MJ |
| PSA | - Level-1 PSA | - CDF by internal events ~ $1.2\times10^{-9}$/Rx-yr |
| Source Term | - In-vessel source term evaluation | - Non-mechanistic source term but consider fission product retention in sodium |

## 3. R&D Activities on PGSFR Technology V&V

The validation and verification (V&V) activities to demonstrate the system performance and safety of PGSFR are in progress in parallel with the design efforts for the PGSFR [5]. The major activities are as follows:

- Reactor mock-up physics test,
- Performance test for the DHRS heat exchangers (STELLA-1),
- Sodium thermal hydraulics integral effect test for PGSFR (STELLA-2),
- Hydraulic performance test of the PHTS pump,
- Performance test for a finned-tube sodium-to-air heat exchanger (FHX),
- Intermediate heat exchanger (IHX) flow characteristics test,
- Reactor flow distribution test,
- Core thermal-hydraulic characteristics test,
- Dynamic characteristics test of the upper internal structures,
- Performance test of the control rod drive mechanism,
- Drop test of the control rod assembly,
- In-service inspection tests of the waveguide sensor for reactor internals, EMAT for the reactor vessel, and the combined sensor for steam generator tubes,
- Irradiation programs on the advanced cladding and fuel materials.
- Fuel assembly mechanical and hydraulic tests

The supports of V&V activities are essential to demonstrate the PGSFR safety characteristics. The main items of the performance demonstration and computer codes V&V tests were introduced. The current status of each activity was explained with emphasis on the significant test results. A large-scale sodium thermal-hydraulic test program (STELLA) including the STELLA-1for separate effect tests and the STELLA-2 for integral effect tests are planned. The STELLA-1 produced satisfactory results of heat transfer characteristics of the decay heat exchanger (DHX) and natural-draft sodium-to-air heat exchanger (AHX) in the DHRS and they were used to validate both the heat exchanger design codes and safety analysis code. The STELLA-2 is under construction and it is expected to be the world unique integral effect test facility for SFR including all the heat transport systems of PHTS, IHTS, and DHRS. The scale of the STELLA-2 system is of 1/5 in length and of 1/125 in volume. The detailed design of STELLA-2 facility has been completed in 2016 and the procurements of several major components will be initiated during 2017.

The similarity test of the model mechanical pump which were performed in STELLA-1 under sodium condition showed a good agreement with the test results in water. Based upon this result, the performance test of a hydraulic model pump reflecting an impeller and a diffuser with a real-sized design data of PGSFR has been conducted in water to generate the test database which will be used for the safety analysis.

The SELFA facility for forced-draft sodium-to-air heat exchanger (FHX) performance test has been constructed and the test operation is ongoing. The intermediate heat exchanger (IHX) performance test is being prepared by utilizing existing STELLA-1 loop and test section of DHX. The test conditions of flow rate and operating temperature are refined by replacing sodium pump and controller in STELLA-1. The modification of the loop were already completed and the IHX performance test will be completed within 2017. The detailed design of the test facility for reactor pool flow distribution, called as PRESCO is underway which aims the completion by the first half of 2017.

The performance tests for both primary CRs and secondary shutdown assemblies have been completed. These include performance tests of the control rod driving mechanism such as an electromagnet, an abnormal withdrawal prevention part, a gripper, driving motors, and the verification test for the passive shutdown mechanism. The drop tests of the CRA under scram conditions were performed to be compared with those from drop analyses. The drop analysis methodology was verified with the test results and the optimal design is currently under way.

Three types of inspection sensors are currently under development for the safe operation of PGSFR and their basic performance tests have been conducted; the waveguide sensor for reactor internals, EMAT for the reactor vessel and the combined inspection sensor for steam generator tubes. Although the feasibilities of the developed inspection sensors are successfully demonstrated, more efforts should be made to improve the performance for the application to an actual inspection.

For U-Zr fuel, fuel design for PGSFR, and fabrication of all the fuel components and fuel assembly were performed [6]. Verification tests of U-Zr fuel are under way. The fuel cladding of ferritic-martensitic steel, FC92, which has a higher mechanical strength at high temperature than conventional HT-9 cladding was developed, fabricated and is being irradiated in the fast experimental reactor. Barrier such as Cr electroplating on inner cladding surface to prevent an interaction between the metal fuel and cladding during irradiation was fabricated and tested in the reactor showing satisfactory performance. As a first milestone, performance of U-Zr fuel will be verified and technical feasibility will be demonstrated by 2020.

Cladding tubes of FC92 and HT9 are subjected to irradiation tests in an experimental fast reactor, BOR-60. It is essential not only to demonstrate cladding performance under fast neutron environment but also to measure in-pile characteristics of cladding for fuel design. Irradiation creep and swelling tests are of utmost importance to obtain in-pile creep model of

FC92 for which out-of-pile creep data are used as supplement. Two irradiation rigs were used; Material Test Rig (MTR)-1 and MTR-2 for which nominal irradiation temperatures are 600℃ and 650℃, respectively. Peak irradiation doses at the end of 2019 are expected to reach 45 and 75 dpa for MTR-1 and MTR-2, respectively. In March 2015, MTR-1 and MTR-2 tests were initiated after the completion of the verification test. As of May 2016, peak irradiation dose reached 12.6 ± 1.5dpa in MTR-1 and 22.9 ± 0.7dpa in MTR-2. The first interim inspection was done for MTR-1 and -2.

PGSFR fuel verification is being made through in-pile and out-of-pile tests. Irradiation tests of fuel rods and fuel components are underway in both thermal and fast research reactor. Fuel behavior depending upon temperature and fission except fast neutron flux can be evaluated by irradiation test in thermal reactor such as HANARO (KAERI) and ATR (INL). The fuel irradiation rig was fabricated, and the irradiation has been begun at an instrumented irradiation position of BOR-60 in July 2016. In-reactor behavior of U-Zr fuel rod for PGSFR initial core is scheduled to be mostly confirmed around 2020. The present fuel tests at fast neutron environment will be extended beyond 2020 to reach the target burnup.

## 4. Summary

The primary goal of PGSFR is the demonstration of reduction of radioactive waste from spent nuclear fuel by transmuting highly radio-toxic and long-lived elements. The successful construction and demonstration of PGSFR will bring Korean nuclear industry a step closer to guarantee the sustainable operation of NPPs nationwide and finally most importantly will solve the issue on spent nuclear fuel management of Korea.

Based on the experiences gained through the development of past KALIMER designs, KAERI is developing PGSFR design that can better meet the Gen IV technology goals and the technologies necessary for its demonstration. Several advanced design concepts were developed to improve the economics, safety, reliability, and metal fuel performance of SFR in the areas of reactor core, fuel and materials, reactor systems and the balance-of-plant.

The safety design of PGSFR emphasizes accident prevention by enhancing inherent safety characteristics and passive safety features using natural phenomena. To support the development of PGSFR design and technologies, R&D activities are being performed for various topics including the validation of neutronics analysis codes, safety demonstration of DHRS in conjunction with primary systems, sodium technology development, and metal fuel qualification.

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