Review Article

Design Concept of Advanced Sodium-Cooled Fast Reactor and Related R&D in Korea

Yeong-il Kim, Yong Bum Lee, Chan Bock Lee, Jinwook Chang, and Chiwoong Choi

Korean Atomic Energy Research Institute (KAERI), 989-111 Daedeok-Daero, Yuseong-Gu, Daejeon 305-353, Republic of Korea

Correspondence should be addressed to Chiwoong Choi; cwchoi@kaeri.re.kr

Received 28 September 2012; Revised 16 February 2013; Accepted 26 February 2013

Academic Editor: Wei Shen

Copyright © 2013 Yeong-il Kim et al. This is an open access article distributed under the Creative Commons Attribution License, which permits unrestricted use, distribution, and reproduction in any medium, provided the original work is properly cited.

Korea imports about 97% of its energy resources due to a lack of available energy resources. In this status, the role of nuclear power in electricity generation is expected to become more important in future years. In particular, a fast reactor system is one of the most promising reactor types for electricity generation, because it can utilize efficiently uranium resources and reduce radioactive waste. Acknowledging the importance of a fast reactor in a future energy policy, the long-term advanced SFR development plan was authorized by KAEC in 2008 and updated in 2011 which will be carried out toward the construction of an advanced SFR prototype plant by 2028. Based upon the experiences gained during the development of the conceptual designs for KALIMER, KAERI recently developed advanced sodium-cooled fast reactor (SFR) design concepts of TRU burner that can better meet the generation IV technology goals. The current status of nuclear power and SFR design technology development program in Korea will be discussed. The developments of design concepts including core, fuel, fluid system, mechanical structure, and safety evaluation have been performed. In addition, the advanced SFR technologies necessary for its commercialization and the basic key technologies have been developed including a large-scale sodium thermal-hydraulic test facility, super-critical Brayton cycle system, under-sodium viewing techniques, metal fuel development, and developments of codes, and validations are described as R&D activities.

1. Introduction

In Korea, electricity demand has increased by about eleven times since 1980 with an average annual growth rate of 8.7% mainly due to economic growth. The anticipated average annual growth rate is estimated to be 2.2% during the period of 2010 to 2024, as shown in Figure 1 [1]. However, the available energy resources are extremely limited in Korea: no domestic crude oil, little natural gas, and limited sites for hydro power. Consequently, about 97% of energy resources come from abroad. Nuclear power plants currently generate about 40% of the total electricity, and the role of nuclear power plants in electricity generation in Korea is expected to become more important in the years to come due to Korea's lack of natural resources. The significance of nuclear power will become even greater, considering its practical potential in coping with the emission control of green-house gases. This heavy dependence on nuclear power eventually raise the issues of an efficient utilization of uranium resources, which

Korea presently imports from abroad, and of a spent fuel storage [2].

From the viewpoint that a sodium-cooled fast reactor (SFR) has the potential of an enhanced safety by utilizing inherent safety characteristics, transuranics (TRU) reduction, and resolving the spent fuel storage problems through a proliferation-resistant actinide recycling, an SFR is sure to be the most promising nuclear power option.

The Korean Atomic Energy Research Institute (KAERI) has been developing SFR design technologies since 1997 under a National Nuclear R&D Program. The goals of the SFR design technology development project are to secure strategic key technologies and develop the conceptual design of an SFR which are necessary for an efficient utilization of uranium resources and a reduction of a high level waste volume. The SFR design technology development project has been carried out as follows. From 2002 to 2005, the preliminary design concept for KALIMER-600 was developed [3]. The basic key technologies were developed according to a power level

FIGURE 1: Forecast for electricity demand.

increase based on the KALIMER-150 design concepts and the advanced concepts such as a nuclear proliferation resistant core, a simplification of an IHTS pipeline and reactor structures, have been developed. The experimental data were secured through basic experiments such as a verification experiment for the computational models and sodium detection experiments. The basic key computer codes and methodologies have been continuously improved, and additional ones have been developed as necessary. Recently, the long-term advanced SFR R&D plan has been set up again aiming at the construction of an advanced SFR prototype reactor in 2028.

2. Design Concept of the Demonstration Reactor

2.1. Top-Tier Design Requirements. The KALIMER-600 design served as a starting point for developing a new advanced design which is equipped with advanced design concepts and features. Various advanced design concepts have been proposed and evaluated against the design requirements which were established to satisfy the Gen IV technology goals.

The top-tier design requirements of a 600 MWe TRU burner are categorized by three criteria: general design requirements, safety and investment protection, and plant performance and economy. Details of these design requirements are given in Table 1. These requirements reflect the design policies, especially emphasizing proliferation resistance, safety assurance, and metal fuel performance, and form the basis for developing the detailed system design requirements for key NSSS concepts.

2.2. Core Design. A conceptual core design for demonstrating TRU burning has been developed. The main objectives are to test and demonstrate the TRU fuel, operate a large sized (1,500 MWth) SFR, and show the TRU burning capability of a commercial burner reactor [4]. It is scheduled to use uranium fuel for the initial core due to the uncertainty of the demonstration of TRU fuel. The LTRU core fuel from a light water reactor (LWR) spent fuel and MTRU core fuel, which consists of LMR spent fuel and self-recycled fuel, will be used progressively, and thus three cores a uranium core, LTRU core, and MTRU core were designed.

TABLE 1: Summary of top-tier design requirements of an SFR demonstration plant.

General design
Reactor type: pool type
Plant size: 600 MWe
Plant design lifetime: 60 years
Design basis earthquakes (SSE: 0.3 g)
Initial core: U-Zr metal fuel
Reloading core: U-TRU-Zr metal fuel
Safety and investment protection
Design simplification
Negative power reactivity coefficient
$CDF < 10^{-6}/\text{reactor} \cdot \text{yr}$
No fuel-cladding liquid phase propagation during DBEs
Diversified core shutdown mechanism
Reliable and diversified decay heat removal
Accommodating unprotected ATWS events without any operator's action
Large radioactivity release $< 10^{-7}/\text{reactor} \cdot \text{yr}$
3 days grace time w/o any operator's action for design basis events
Performance and economy
Plant thermal efficiency: net > 38%
Plant availability $\geq 70\%$
Refueling interval: U-Zr initial core ≥ 6 months
TRU burner core ≥ 11 months
Spent fuel storage capacity in RV ≥ 1.5 cycle discharge
100% off-site load rejection w/o a plant trip
Safety grade diesel generator

The core functions are given in Table 2. Every core was designed maintaining the same core dimension of the TRU core. Figure 2 shows the layout of the 600 MWe-rated uranium core. As shown in Figure 2, the core consists of two fuel regions. It consists of 151 fuel assemblies in the inner core and 174 fuel assemblies in the outer core. The fissile enrichments of the inner/outer cores for the radial power control are 15 and 20 wt.%, in which the enrichment of 20 wt.% is the maximum allowable enrichment in the commercial market for the uranium core. The hexagonal fuel assembly consists of 271 rods within a duct wrapper. The outer diameter of a rod is 7.4 mm. The core configuration is a radial homogeneous one that incorporates annular rings with a region-wise enrichment variation. The active core height was adjusted to make the enrichment of the outer core 20 wt.%, and the adjusted height is 85 cm. Table 3 shows a summary of the core performance analysis results, obtained with the equilibrium cycle analysis. The burnup reactivity swing for the uranium core was estimated to be 1,698 pcm.

TABLE 2: Core function for TRU burning.

U core	Trans core	LTRU core	Trans core	MTRU core
Pure U core equilibrium		LWR-TRU equilibrium		
Open fuel cycle	U/LWR-TRU transition	Open fuel cycle	Transition core to final	Pure TRU core
LWR-TRU (LTRU) fuel	cycle	LWR-TRU and self TRU	TRU core	equilibrium (MTRU fuel
demonstration	Open fuel cycle	mixed (MTRU) fuel	Beginning of self cycle	equilibrium)
LTRU test zone install		demonstration		FOAK role start

TABLE 3: Core performance of demonstration cores.

	Uranium core	LTRU core	MTRU core
Charged fuel enrichment (U, TRU) (wt.%) (inner/outer core)	15.0/20.0	19.2/24.8	21.5/28.8
Conversion ratio (Fissile/TRU)	0.54/–	0.80/0.71	0.81/0.64
Burnup reactivity swing (pcm)	1,698	2,588	2,945
Cycle length (EFPD)	255	365	365
Avg. discharged burnup (MWD/kg) (inner/outer core)	51.8/42.4	83.4/76.0	79.5/81.9
Peak fast neutron fluence ($\times 10^{23}$ n/cm ²)	1.92	3.23	3.12
Power peaking factor (BOEC/EOEC)	1.480/1.53	1.50/1.44	1.57/1.50
TRU consumption rate (kg/cycle)	—	145.9	184.9
Sodium void worth (EOEC, \$)	–0.22	7.70	7.03
Doppler coefficient (pcm/°C)	$-1172.7T^{-1.122}$	$-1012.4T^{-1.152}$	$-911.0T^{-1.149}$
Axial expansion coefficient (pcm/°C)	–0.140	–0.217	–0.268
Radial expansion coefficient (pcm/°C)	–0.780	–0.949	–0.980
Sodium density coefficient (pcm/°C)	0.076	0.717	0.627

The LTRU core was designed next to the uranium core. The core uses the spent fuel from the LWR and adapts a once-through cycle option. Radial and axial power distributions were flattened through searching enrichment ratios between the inner and outer cores to minimize power peaking. When the TRU enrichments of the inner and outer core regions reach 19.2 wt.% and 24.8 wt.%, the power peaking factors were estimated to be 1.50 at the beginning of equilibrium cycle (BOEC) and 1.44 at the end of the equilibrium cycle (EOEC), and these values are well below 1.60 of the predetermined design limit.

The MTRU core uses a mixed TRU fuel with LWR spent fuel and self-recycled fuel. In the MTRU core design, reflector assemblies were introduced in the central region of the core to reduce the increased sodium void worth. The TRU consumption rate was estimated to be 185 kg/cycle, and the burnup reactivity swing, 2,945 pcm.

A demonstration core was selected after a series of core designs, ranging from the U core to the LTRU and MTRU cores. A special effort to increase the discharge burnup was made due to a relatively low discharge burnup in the U core. As shown in Table 4, five candidates including the first one as the reference were applied and analyzed. Candidates 1, 2, and 3 had the same core height but a different fuel loading cycle to simplify core modification from U core to TRU core without any geometry changes in structure. However, these modifications could not improve the discharge burnup effectively in the U core as well as sodium void reactivity also increased in the LTRU and MTRU cores, as shown in Figure 3.

TABLE 4: Parameters of candidate cores.

Core candidate	1	2	3	4	5
U core height (cm)	85	85	85	95	106
TRU core height (cm)	85	85	85	85	85
Fuel diameter (mm)	7.4	7.4	7.4	7.4	7.0
Fuel batch cycle (inner/outer core)	3/3	4/3	4/4	3/3	3/3

As alternate approaches, candidates 4 and 5 were suggested by changing the active core heights and a fuel pin diameter for the same purpose, even though modification of the core structure was required. Candidates 4 and 5 had an improved discharge burnup compared with candidates 1, 2, and 3 in the U core, but higher sodium void reactivity relatively than that of candidates 1 and 2 in the MTRU core. Therefore, candidate 1 was selected even though this core showed a lower discharge burnup than that of candidates 4 and 5 because it revealed the best performance from U core to TRU core in the safety aspect between candidates 1, 2, and 3 and could keep the same core dimension.

2.3. Fuel Design. The probability of cladding failure or damage during the steady state and transient conditions must be evaluated by appropriate predictive codes. To prevent a metallic fuel rod failure in a fast reactor, it is required to evaluate the design limits such as (1) cladding strain and cumulative damage fraction (CDF), (2) fuel melting, and (3) eutectic melting.

FIGURE 2: Layout of uranium core (600 MWe).

FIGURE 3: Core performance comparison.

The design requirement for cladding is assumed to be 1% of the thermal creep strain and 0.05 of CDF. The cladding strain limit and CDF limit for metal fuel were evaluated by the MACSIS code. These limits depend on the plenum-to-fuel ratio, cladding thickness/temperature, and burnup.

If the cladding temperature becomes higher than 625°C, it was estimated that the HT9 cladding was not conservative enough to satisfy the CDF limit because the creep rupture strength was too low at a higher temperature. If the cladding temperature becomes higher than 645°C, it was estimated that the HT9M cladding satisfied the CDF limit by establishing the optimum design parameters. Therefore, 625 and

645°C are conservatively selected as the peak clad temperature for the HT9 and HT9M, respectively.

Figure 4 shows the calculation results of the CDF limits according to the plenum-to-fuel ratio, cladding temperature, and burnup. If the plenum-to-fuel ratio was enlarged, it was estimated that the HT9M cladding satisfied the CDF limit at the discharge burnup goal.

Radiation damage to the cladding by fast neutrons can result in swelling and a ductility reduction of the cladding. HT9 and HT9M cladding are very tolerant to fast neutron irradiation owing to their lattice structure of body centered cubic (BCC). HT9 cladding is known to show very low

swelling and maintain its mechanical integrity up to 4.0×10^{23} n/cm². The peak fast neutron fluence of the cladding shown in Table 3 is below 4.0×10^{23} n/cm².

The fuel melting temperature limits of 955 and 1200°C are used for U-TRU-Zr and U-Zr fuel, respectively. It was estimated that the metallic fuel had a sufficient margin to the slug melting temperature. However, the fuel surface temperature to avoid eutectic melting is limited to 650 and 720°C for U-TRU-Zr and U-Zr, respectively. It was calculated that there is a sufficient margin for U-Zr. However, the power-to-eutectic limit was decreased to about 350 W/cm for U-TRU-Zr. This result showed that the concept of a barrier cladding may be necessary for preventing the eutectic melting, in the case of U-TRU-Zr.

2.4. Fluid System Design. The fluid system has been designed to ensure the safety goal of the Gen IV reactor system and enhance the economics through a tradeoff study between various proposed design candidates based on proven technologies [5]. The fluid transport system is composed of a heat transport system and safety system.

The heat transport system consists of a primary heat transport system (PHTS), intermediate heat transport system (IHTS), and power conversion system (PCS). The Decay Heat Removal System (DHRS) is employed as one of the safety design features to remove the decay heat of the reactor core after the reactor shutdown when the normal heat transport path is unavailable.

The PHTS is a pool type in which all the primary components and primary sodium are within a reactor vessel to prevent primary sodium from leaking outside of the containment, as shown in Figure 5. Two PHTS pumps and four intermediate heat exchangers (IHXs) are immersed in the sodium pool inside a reactor vessel, and their arrangement is presented in Figure 6. The PHTS pump is a centrifugal type mechanical pump with a capacity of 290.3 m³/min. The IHX is a counter flow shell and tube types (TEMA type S) with a vertical orientation inside the reactor vessel where PHTS sodium flows through the shell side and IHTS sodium flows through the tube side. The schematic design concepts of the PHTS pump and IHX are shown in Figure 7. The core inlet and outlet temperatures are 365°C and 510°C, respectively.

The IHTS is two loops, and two IHXs are connected to one steam generator and one IHTS pump in each loop as shown in Figure 6. An IHTS pump is a centrifugal type with a capacity of 209.8 m³/min and is located in each cold leg. A steam generator is a helical tube type with a thermal capacity of 776.7 MWt, and its schematic design concepts are shown in Figure 7. The IHTS sodium flows downward through the shell side while the water/steam goes up through the tube side. Steam temperature and pressure at a 100% normal operating condition are 471.2°C and 17.8 MPa, respectively. The cold leg of the IHTS piping is a bottom up U-shape with sufficient height to prevent sodium-water reaction products from reaching the IHX in case of a steam generator tube failure. Also, the IHTS piping is arranged to enhance the natural circulation capability in IHTS pump trip case.

FIGURE 4: CDF limits according to plenum-to-fuel ratio, cladding temperature, and burnup.

The PCS employs a superheated steam Rankine cycle. The normal operating condition at 100% power is shown in Figure 8. It was determined in such a way to minimize the total heat transfer area of IHX and steam generator and maximize the plant efficiency.

The DHRS is composed of two passive decay heat removal circuits (PDRCs) and two active decay heat removal circuits (ADRCs). It was designed to have the sufficient capacity to remove the decay heat in all design bases events by incorporating the principles of redundancy and independency. The heat removal capacity of each loop is 9 MWt. The PDRC is a safety-grade passive system which is comprised of two independent loops with a decay heat exchanger (DHX) immersed in a hot pool region and a natural-draft sodium-to-air heat exchanger (AHX) located in the upper region of the reactor building for each loop. It is operated based on the natural circulation by density and the elevation difference between the DHX and AHX. The ADRC is a safety-grade active system, which is comprised of two independent loops with a DHX, a forced-draft sodium-to-air heat exchanger (FDHX), an electromagnetic pump, and an FDHX blower for each loop. The electromagnetic pump and FDHX blower derive the sodium circulation in the loop and the air flow in the shell side of FDHX, respectively. Because the ADRC can also be operated in natural convection mode against a loss of power supply, the heat transferred to the DHRS can be finally dissipated to the atmosphere through AHXs and FDHXs by the natural convection mechanism of sodium and air. Figure 9 shows the design concepts of heat exchangers.

2.5. Mechanical Structure Design. The reactor enclosure system is composed of double vessels (reactor vessel and guard vessel) and a thick flat plate of the reactor head. Figure 10 shows the configuration of the conceptually designed reactor steam supply system. In this design, the reactor vessel size is 12 m in diameter and 16.5 m in height. IHTS main piping is 144 m long per loop system. Figure 10 shows the top view of

FIGURE 5: Configuration of the heat transport system.

the arrangements of the main components and IHTS piping including the decay heat removal system.

The reactor system is supported by a skirt type support structure which joints the reactor head and the reactor vessel by bolts. This will provide access holes for in-service inspection devices to inspect the reactor and guard vessels. The core support structure is a detached skirt type structure which has no welds between the core support structure and reactor vessel bottom head. This is just put on the flange forged with a reactor vessel bottom head to allow a free thermal expansion.

2.6. Safety Analysis. The TOP, LOF, LOHS, primary pipe break, and reactor vessel leak event are analyzed using the MARS-LMR code. The ANS-79 model is used for a core decay power after a reactor scram. AHX dampers are assumed to open at 5 seconds after a reactor trip. The isolation time of the SG feed water line is assumed to be the same as the pump trip time. Two independent PDRCs and one ADRC are assumed to be available by applying a single failure criterion.

The TOP accident was assumed to be initiated due to a control rod withdrawal by the drive motor failure. The TOP accident is initiated at 10 seconds, and a positive reactivity is inserted by the amount of 30 ¢ during 15 seconds. The reactor trip occurred at 22.73 seconds by a high power/flow trip. The power peaks after the initiation of rod withdrawal, it decreases drastically due to the reactor trip, and the cladding temperature in the reactor core shows the highest value. The peak cladding temperature was calculated at 580.93°C which was lower than the limit value. Figures 11(a) and 11(b) show behavior of the core inlet temperature and total heat balance in the plant, respectively. The AHX heat removal exceeds the core power after 4400 seconds, and the core outlet temperature decreases continuously. In conclusion, PHTS and the fuel temperature meet all safety criteria for the TOP accident.

The LOF means the loss of core cooling capability due to a pumping failure of the primary pumps. The imbalance between the reactor power and primary flow rate is a main

FIGURE 6: The arrangement of PHTS and IHTS.

safety concern in the LOF event. To prevent the occurrence of the severe imbalance between power and flow, the DFR is designed so as far the reactor to be tripped by a high power/flow trip. Figure 12 shows the coolant temperature behaviors during the LOF accident. In this simulation, all primary pumps are tripped at 10 seconds. The reactor scram occurs at 16.9 seconds, and the reactor power and flow rate decrease. The power decreases drastically due to the reactor trip, and the cladding temperature in the reactor core then shows the highest value. The peak cladding temperature was calculated at 624.27°C . The temperature is evaluated to meet the safety criteria.

The LOHS accident was assumed to occur from an initiated steam generator feedwater isolation. IHTS pumps and PHTS pumps are also stopped with the assumption that the loss of offsite power occurred at 5 seconds after the reactor trip. Therefore, the residual heat removal is achieved only by the evaporation of water in SG tubes and by the SHRS after the accident. In this simulation, a loss of feedwater to SG is assumed to occur at 10 seconds. The reactor was tripped at 58.77 seconds by an abnormal rise of the IHX inlet temperature after the accident. The reactor trip occurs late unlike other accidents. Figure 13 shows the coolant temperature behaviors during the LOHS accident. After the pump trip, the coolant temperatures go up rapidly and the maximum coolant temperature is calculated as around approximately 513.56°C . The temperature meets the safety criteria.

Coolant flows into the inlet plenum from four pipes which are connected with two PHTS pumps. The primary pipe break accident is occurred by a pipe break for one of the pipes. The flow through the broken pipe is discharged into the cold pool, and some of the low temperature sodium flowing

through an intact pipe into the inlet plenum is released into the pool. Essentially, this event is similar to an LOF accident. The accident occurs at 10 seconds as shown in Figure 14. The initial temperature increases due to the decrease of sodium flux into the reactor core. In this simulation, the peak coolant temperature was calculated at 579.23°C . The temperature is lower than the limit value.

A reactor-vessel-leak accident is a typical accident of a sodium leak at the PHTS boundary. It mainly affects the level of sodium in the PHTS. To analyze the damage of the reactor vessel leak accidents, the leak was assumed to occur at the bottom of the reactor vessel, conservatively, and the leak size was assumed to be 10 cm^2 in size.

Figure 15 shows the coolant temperature behaviors during the reactor vessel leak. The accident occurred at 10 seconds. The reactor trip occurred at 884.47 seconds. It is detected by the low liquid level from the reactor vessel leak. After the reactor trip, the flow behavior is similar to the loss of flow. The highest cladding temperature was calculated at 609°C . The peak cladding temperature satisfies the safety criteria.

The consequence of the blockage formation in a drive fuel assembly was deliberately analyzed with a subchannel analysis code, MATRA-LMR/FB, for the demonstration reactor. It was applied to the analysis of flow blockage accidents postulated in a conceptual design of the demonstration reactor with 3 types of core designs, that is, uranium, L-TRU, and M-TRU cores. The analysis was performed for a hot fuel subassembly. The blockage size and radial channel blockage position in the subassembly were the main parameters taken into account in the analysis. The three radial positions examined in the analysis were the center, the middle between the subassembly center and the duct wall, and the edge of the subassembly.

Figure 16 summarizes the analysis results. The design basis event, that is, 6 subchannel blockage was ensured to satisfy the safety limits. The cases for the 24 and 54 subchannel blockages, however, could not meet the peak cladding temperature limit. Although a sufficient margin of more than approximately 150°C might be obtained against sodium boiling, fuel melting was threatened for the 54 subchannel blockage.

3. R&D Activities

3.1. Large-Scale Sodium Thermal-Hydraulic Test Facilities. According to the long-term SFR development plan approved by the Korean government, a specific design approval of the prototype SFR will be obtained by 2020, and its construction is scheduled to be completed by 2028. To support this program plan, a large-scale sodium thermal-hydraulic test program called STELLA (sodium test loop for safety simulation and assessment) is recently being progressed by KAERI.

The reference design of the program is the Korean prototype SFR which employs highly reliable safety-grade decay heat removal systems. Since a reliable decay heat removal is one of the most important issues of nuclear safety, the performance of a decay heat removal system should be verified using a large-scale test facility. To this end, the first test facility

FIGURE 9: Heat exchanger design concepts of DHRS.

FIGURE 10: Reactor structure, system, and components.

reflecting the prototype design concept will be performed on the basis of design requirements subject to the prototype reactor. The facility is scheduled to be installed by the end of 2016. The main experiments including the start-up tests will commence in 2017. The STELLA program finally aims the integral effect test to support a specific design approval for a Korean prototype SFR.

STELLA-1 consists of a main test loop, a sodium purification system, and a gas supply and related auxiliary systems. The main components of this facility are a sodium-to-sodium heat exchanger, sodium-to-air heat exchanger, mechanical sodium pump, loop heaters, cold trap, plugging meter, electromagnetic pumps, flow meters, and a sodium storage tank. The general arrangement of the STELLA-1 facility is shown in Figure 17.

The designed maximum temperature of the facility is 600°C, and the designed power capacity of the main heat exchangers, such as sodium-to-sodium and sodium-to-air heat exchanger, are 1 MWt. The maximum electric power into the facility is around 2.5 MWt, and the nominal liquid sodium flow rate supplying the test heat exchangers is designed to be less than 10 kg/sec. During the mechanical pump test, more than 120 kg/sec of liquid sodium circulates along 10-inch diameter pipes.

At the first step of the demonstration of the design characteristics and system performance, separate effect tests for assessing the performance of heat exchangers and the mechanical sodium pump have been planned. The sodium-to-sodium heat exchanger tests are performed to investigate the rate of heat transfer through the tube wall by hot and cold sodium loop operation. In the sodium-to-air heat exchanger tests, the heat transfer performance from liquid sodium flow inside the tubes to the air flow is investigated by cooling the external tube surface with ambient air. To evaluate the heat removal capability in passive mode, a natural circulation flow inside sodium loop piping is also investigated using a bypass of the electromagnetic pump. The PHTS pump test loop consists of a reservoir, pipes, valves, and a vertical pump unit. This loop is equipped with various sensors for measuring the flow rate, temperature, liquid sodium level, and so forth. The main test loop is designed to simulate the transient operation mode using a flywheel as well as normal operation mode.

3.2. S-CO₂ Brayton Cycle System. The S-CO₂ Brayton cycle energy conversion option has many advantages such as excellent thermal efficiency and compactness of its equipment, for example, small turbo machinery and heat exchangers. Furthermore, by coupling the system to the SFR, the safety of the SFR could be enhanced by an elimination of the sodium-water reaction. To adopt the S-CO₂ Brayton cycle to the SFR, several R&D activities were done, such as the system design, operational strategy, Na-CO₂ reaction, and heat exchanger development.

In the system design, a design concept of an S-CO₂ Brayton cycle coupled with a KALIMER-600 was developed, and the system operational strategy was developed to evaluate the operating conditions at various power levels. When changing the system flow rate to vary the system power level, a pressure imbalance occurs from the difference of turbine and compressors design characteristics. To resolve the pressure imbalance, a clutch and throttle valve design concept was introduced and a system transient analysis was done by the use of the MMS-LMR commercial code.

For the enhancement of system performance by decreasing the pressure loss in a high and low temperature recuperator, a new design concept of heat exchanger was proposed by the application of an airfoil type fin to S-CO₂ flow path. For the new model, three-dimensional numerical analysis was performed to investigate the heat transfer and pressure drop characteristics of supercritical CO₂ flow using the commercial CFD code, Fluent 6.3. From the simulation results, the total heat transfer rate per unit volume was almost the same with a zigzag channel PCHE and the pressure drop was

FIGURE 11: Predicted transient behaviors for TOP event.

FIGURE 12: Coolant temperature behaviors for LOF event.

FIGURE 13: Coolant temperature behaviors for LOHS event.

reduced to one-twentieth of that in the zigzag channel PCHE by suppressing the generation of a separated flow owing to the streamlined shape of the airfoil fins [6].

To test the performances of the new design, a model heat exchanger was fabricated as shown in Figure 18 and installed in the test facility in Figure 19. The test facility is composed of a storage tank, an electromagnetic pump, an electromagnetic flow meter, an expansion tank, a heat exchanger test section, a liquid sodium line, and a cover gas line used for the charging and returning the sodium. There are two emersion heaters inside of an expansion tank as a 4 kW heat source, 2 kW each. The storage tank has a capacity of 10 liters with a cylindrical shape. An EM pump is installed vertically upright to prevent trapping the cover gas inside the pump. The material of every

component and piping is stainless steel 316 L, and only the cover gas lines are stainless steel 304. The total charged amount of sodium in the storage tank is 8 liters, and 5~6 liters of sodium were used for the experiment.

From the test results, the pressure loss was one-fifth of that of a zigzag channel which comes from the fact that the streamlined shape of airfoil fins also suppresses the generation of a separated flow. Thus, the airfoil shape fin model resulted in a much smaller pressure drop than observed in the zigzag PCHE. However, the smaller pressure loss in the experimental results than the numerical results seems to come from the uncertainties of manufacturing and fabrication but the heat transfer rate is almost similar to the numerical simulation results [7].

FIGURE 14: Coolant temperature behaviors for pipe break event.

FIGURE 15: Coolant temperature behaviors for vessel leak event.

Even though the S-CO₂ Brayton cycle has many advantages, there still exists a possibility of CO₂ leakage into liquid sodium from a pressure boundary failure. The pressure boundary failure can raise technical issues such as structural integrity from the blow down of high pressure CO₂ gas into a liquid sodium space with a significant chemical reaction and the introduction of solid reaction products into the primary coolant system, which could result in the plugging of narrow flow channels.

To quantify the reaction rate for various sodium temperatures and determine the detailed kinetic parameters coupled with a mass diffusion process, a two zone model was proposed and experimental work on a surface reaction test was carried to decide the value of the E_a/R values of Figure 20. From

FIGURE 16: Outlet temperatures for subassembly with flow blockage.

the test results, it was found that the reaction kinetics over a sodium temperature range of 300°C to 500°C depends heavily on temperature but is not sensitive to a mass transfer effect, and it was also found that the two zone model with a 460°C threshold temperature is valid for the temperature range of a sodium fast reactor [8]. Furthermore, we need to investigate the ingress of a CO/CO₂ mixture gas into the primary coolant path and the resulting induction of a potential CO₂ void transport to the reactor core as a critical issue. The potential introduction of solid particles into the primary coolant system would also lead to a risk of plugging in narrow in-core fuel assembly channels, very narrow sections of PCHes, and so forth. Particle formation makes adequate purification systems necessary; such systems should be equipped with high-performance filters to eliminate particles and control the quantity of solid reaction products. This design feature only needs to be considered for a supercritical CO₂ system. Therefore, highly reliable detection systems are required for mitigating the CO₂ ingress event and should include several complementary devices accommodating various local conditions to cope with a sodium-CO₂ interaction [9].

3.3. Under-Sodium Viewing Technique. The ultrasonic waveguide sensor modules have been developed for potential application to under-sodium viewing of in-vessel structures in opaque liquid metal sodium. The sensor modules have a slender structure so that they can be inserted into the ISI access ports in the rotating plug. Two prototype ultrasonic waveguide sensor modules were designed and fabricated for the basic performance tests in water. One is single waveguide sensor module and the other is the dual waveguide sensor module.

The single waveguide sensor module was designed and developed for the under-sodium viewing and ranging using one channel waveguide sensor. The dual waveguide sensor module was designed and developed for the detection and

FIGURE 17: Current images of general arrangement for STELLA-1 facility.

FIGURE 18: Shape of heat exchanger plates.

identification of the loose parts by the double rotation scanning of two channel waveguide sensors.

The 13 m long H-beam frame structure was designed and constructed to install the 10 m long prototype ultrasonic waveguide sensor modules in a vertical state. The prototype ultrasonic waveguide sensor modules are comprised of an ultrasonic waveguide sensor, the multistage cylindrical guide tubes, and an upper head unit. In the upper head unit, the stepping motors are installed for the rotation and vertical movement of the ultrasonic waveguide sensor.

The experimental facility is composed of a 13 m long H-beam frame, an XYZ scanner, a scanner driving module, and an ultrasonic C-scan system. Also the under-sodium inspection software program (US-MultiView) has been developed for control of the ultrasonic waveguide sensor modules and

the C-scan imaging visualization using a LabVIEW graphical programming language. The visualization imaging resolution using the 10 m long single waveguide sensor module was evaluated by a C-scan test of various targets in water. The test targets are a reactor core mockup, loose part pins, and surface slit flaws on the block. The reactor core mockup and loose part pins were clearly identified and resolved in the image, as shown in Figure 21. It was shown that a spatial resolution of the C-scan image for the detection of surface slits is about 0.8 mm.

The novel under-sodium ultrasonic waveguide sensor module has been developed for actual application in sodium. The under-sodium ultrasonic waveguide sensor where a beryllium (Be) and a nickel (Ni) are coated on the SS304 waveguide plate is suggested for the effective generation of a leaky wave in liquid sodium. The inside surface of the radiating end section of the 1.5 mm thick waveguide plate was coated with 0.25 mm thick beryllium to decrease the angle of the radiation beam and make a well-developed beam profile in sodium. The outer surface of the radiating end section was coated with 0.1 mm thick nickel and micropolished to obtain a surface roughness within $0.02 \mu\text{m}$ such that the sodium wetting was greatly enhanced.

The sodium experimental facility has been designed and constructed to demonstrate the performance of an under-sodium ultrasonic waveguide sensor module in a sodium environment condition. The sodium test facility consists of an open-type sodium test tank, a sodium storage tank, a glove box system with an antichamber, an electrical heater and control unit, and an argon circulation and cooling system. The sensitivity of the under-sodium waveguide sensor module is

FIGURE 19: Test facility of sodium-CO₂ heat exchanger.

evaluated by a measurement of the received ultrasonic signal from a flat reflector in sodium.

Figure 22(a) shows the under-sodium C-scan test, and Figure 22(b) shows the typical ultrasonic pulse-echo signal which has the initial pulse, the reflection signal from the end section of an under-sodium waveguide sensor, and the reflection signal from the test target in sodium (250°C). The signal-to-noise (S/N) ratio of the reflection echo signal from the test target in sodium was measured as the level of 10 dB. The visualization performance tests of the 10 m long under-sodium waveguide sensor module have been carried out by a C-scan test in sodium. The test target is the SS304 block in which SFR character with 2 mm slits was engraved. As shown in Figure 22(c), the “SFR” character was clearly identified and resolved in the C-scan image.

3.4. Metal Fuel Development. Metallic fuels, such as the U-Pu-Zr alloys, have been considered as a nuclear fuel for a sodium-cooled fast reactor (SFR) related to the closed fuel cycle for managing minor actinides and reducing the amount of highly radioactive spent nuclear fuels since the 1980s. Metallic fuels fit well with such a concept owing to their high thermal conductivity, high thermal expansion, compatibility with a pyrometallurgical reprocessing scheme, and their demonstrated fabrication at the engineering scale in a remote hot cell environment [10]. A previous attempt at casting metallic fuels with americium using an injection casting furnace that had fabricated hundreds of U-Pu-Zr fuels for EBR-II resulted in a significant volatile loss of elemental

FIGURE 20: Na-CO₂ chemical interaction model.

TABLE 5: Material balance after casting of U-10 wt.% Zr-5 wt.% RE fuel slugs.

	Melting/casting part	Weight (g)	Fraction (%)
Before casting	Crucible	2,461.3	100
After casting	Crucible	122.4	5.0
After casting	Mold	2,331.7	92.7
Fuel loss		7.2	0.3

americium during the process [11]. The reference fuel for the Korean sodium-cooled fast reactor (SFR) being developed by the Korean Atomic Energy Research Institute (KAERI) is a metallic alloy. To increase the productivity and efficiency of the fuel fabrication process, waste streams must be minimized and fuel losses quantified and reduced to lower levels.

U-Zr alloy system fuel slugs were fabricated by a gravity casting method, as shown in Figure 23 [12]. After casting a considerable number of fuel slugs in the casting furnaces, the fuel loss in the melting chamber, the crucible, and the molds have been evaluated quantitatively. The elemental lumps of depleted uranium (DU), zirconium, and RE (Nd 53%, Ce 25%, Pr 16%, La 6%) were used to fabricate U-10 wt.% Zr-5 wt.% RE alloy fuel slugs. The material balance in the crucible assembly and the mold assembly after melting and casting of fuel slugs are shown in Table 5. A considerable amount of dross and melt residue remained in the crucible after melting and casting; however, most charged materials were recovered after melting and casting of the fuel slugs. The mass fraction of fuel loss relative to the charge amount after the fabrication of U-10 wt.% Zr-5 wt.% RE fuel slugs was low, about 0.3%. Based on these results, there is a high level of confidence that RE losses will be effectively controlled.

FIGURE 21: C-scan performance test results of reactor core mockup, loose parts and slits by the 10 m long ultrasonic waveguide sensor module.

FIGURE 22: Basic performance tests of 10 m long under-sodium ultrasonic waveguide sensor in sodium.

HT9 cladding tubes were preliminary fabricated in cooperation with the steel tube making companies. The HT9 cladding tubes were examined by optical microscopy and TEM (transmission electron microscopy). The microstructure of the cladding tube was martensite + delta ferrite. Tensile tests were carried out at room temperature to 700°C. The HT9 cladding tube had yield and tensile strengths similar to the data in the literature. A burst test was performed by

pumping gas up to a burst of a 200 mm long section tube. The pressurization speed was 14 MPa/min. Burst tests were performed at room temperature to 688°C. The ultimate hoop stresses of the HT9 cladding tubes were 1135 MPa and 487 MPa at room temperature and 688°C, respectively. Tube creep tests were also carried out at 650°C. The HT9 cladding tube had creep rupture strength similar to the data in the literature (Figure 24). The tube fabrication process also is being

FIGURE 23: Low pressure gravity casting system.

FIGURE 24: HT9 cladding tubes.

developed to improve the characteristics of the cladding tube. The HT9 cladding tube will be fabricated with an optimized fabrication process in 2013.

One of the factors that may limit burnup in metal alloy fuel is cladding wastage due to the reaction of fuel constituents and fission products with the cladding (FCCI—fuel cladding chemical interaction). To resolve this issue, diffusion couple tests were carried out by inserting barrier materials such as Zr, Nb, Ti, Mo, Ta, V, and Cr between the fuel slug and cladding. Among these barriers, V and Cr exhibited the most promising performance (Figure 25). After scoping various coating methods, Cr electroplating has been selected as one of the probable candidates because it is cost effective and easily applicable to a smaller tube geometry. To demonstrate barrier tube technology, 20 μm of Cr has uniformly plated at the inner surface of the 9Cr-2W FMS tube having 4.6 mm inner diameter and 170 mm length (Figure 25). However, it was revealed that when plating conventional condition, numerous cracks generated during the plating which acts as the diffusion path for the fuel component during the diffusion couple test. Research has focused to reduce such crack to

FIGURE 25: Diffusion couple test and Cr-plated barrier cladding prototype.

enhance the Cr barrier performance. A diffusion couple test showed excellent results when compared to conventional Cr plating.

The irradiation test of U-Zr-(Ce) metal fuel in HANARO was done from 2010 to 2012. HANARO is an experimental thermal reactor using water coolant. Therefore, the temperature and fission density of a fast reactor fuel were simulated, while the fast neutron flux of HANARO is much lower than fast reactor. The composition of the fuel slug is U-10% Zr-(0, 6 Ce). Its objective is to irradiate U-Zr-Ce fuel up to 3 at.%. It is also intended to identify the characteristics of the Cr barrier which is being developed to suppress a eutectic reaction between the metal fuel and cladding. The composition of the fuel slug is U-10% Zr-(0, 6 Ce). Figure 26 shows the irradiation capsule schematic diagram and coolant channel cross-section. Figure 26 also shows the irradiation history of HANARO metal fuel. The average burnup of metal fuel was about 3 at.%. The as-run linear heat rate was 240 W/cm at BOC, and decreased to 220 W/cm at EOC. The as-run analysis shows that the experiment reached an average 2.73 at.% burnup at the completion of the irradiation test. It was estimated that the maximum burnup goal was satisfied.

Postirradiation examination of the irradiated capsule and fuels is being carried out in a hot cell from 2012. Representative destructive tests are to measure or observe the fuel burnup, microstructure, fission gas release, and constituent redistribution. Nondestructive test such as gamma scans was carried out for the five rodlets. A destructive test such as the measurement of the fission gas release and a microstructure analysis is being carried out.

3.5. Development of Codes and Validations

3.5.1. Reactor Physics Experiment for TRU Burner. KAERI has been collaborating with IPPE for validating the reactor core design code system (TRANSX/TWODANT/REBUS-3), in which the self-shielded fine-group (150 groups) cross-sections are generated by TRANSX [13], and the region-wise spectrums from TWODANT [14] are subsequently used to collapse the cross sections in TRANSX. The resulting few-group (25 groups) cross sections are used for the whole core depletion calculation by REBUS-3 [15].

FIGURE 26: Irradiation test in HANARO.

Four critical assemblies had been constructed in BFS-1 or BFS-2 facilities, called BFS-73-1, -75-1, -76-1A, and -109-2A. The first two critical assemblies represent the early phase of the KALIMER-150 core design in the late 1990's which is a metal uranium fuel (U-10Zr) loaded sodium cooled fast reactor. The BFS-76-1A stands for the recent TRU burner core which is characterized by a core without a blanket, a low conversion ratio core, a high burnup reactivity swing, and the consequent deep insertion of a primary control rod at BOEC. Also, the BFS-109-2A demonstrates the initial uranium core, in which the metal uranium fuel is loaded without radial and axial blankets. The recent experimental work of BFS-109-2A will be finished at the end of this year (2013), and the analysis of BFS-109-2A will be finalized at 2014.

3.5.2. System Transients Analysis Code. For a successful design and analysis of a sodium-cooled fast reactor (SFR), it is necessary to have a reliable and well-proven system analysis code. To achieve this purpose, KAERI has been enhancing the modeling capability of the MARS code by adding the SFR-specific thermal-hydraulic models and reactivity feedback models. This effort resulted in the development of the MARS-LMR code. Before using the MARS-LMR code in wide applications, it is necessary to verify and validate the code models through analyses for appropriate experimental data

or analytical results. The reference design of an SFR, which is being developed in Korea, is a pool-type design. In a pool-type SFR, all the main components of the primary heat transport system are arranged in two big sodium volumes: a hot pool and a cold pool. During the transients in a pool-type SFR, the thermal-hydraulic phenomena in the pools become highly complex due to the formation of mixing, stratification and existence of buoyancy force. Therefore, it is necessary to have flexible modeling including a multidimensional approach to enhance the accuracy of a safety evaluation.

Recently, KAERI evaluated the capability of multidimensional modeling for large pools using available test data. One of the important data sets suitable for this evaluation was provided from phenix end-of-life (EOL) natural circulation tests. In the MARS-LMR modeling, the hot pool region from the core outlet to the inlet of IHXs has been divided into 8 axial nodes, 4 radial nodes, and 6 azimuthal nodes, as shown in Figure 27. Further, the cold pool region has been modeled with 12 axial nodes, 1 radial node, and 9 azimuthal nodes. The remarkable results of this multidimensional pool modeling are compared with one-dimensional modeling in Figure 28. It was found that the overpredicted core outlet temperature with a one-dimensional approach is diminished in the multidimensional calculation. This result indicated that the multidimensional effect in the pool behaviors is important in a pool-type SFR.

FIGURE 27: Nodalization of phenix for MARS-LMR simulation.

FIGURE 28: Predicted core outlet temperature.

4. Summary

In Korea, most energy resource supplies depend on imports because the available energy resources are extremely limited. Therefore, the portion of nuclear power in electricity generation is expected to be continuously increased in the years to come in achieving energy self-reliance. A fast reactor is the most promising future nuclear power plant because of efficient usage of uranium and reduction of radioactive waste. In particular, a sodium cooled Fast reactor (SFR) has been focused as a new generation of nuclear power plants in Korea.

Since 1977, the basic key technology development for an SFR has been continued, and the design concepts of KALIMER-150 and KARIMER-600 have been successfully achieved. In 2008, KAEC approved a long-term advanced SFR R&D plan which aims at the construction of an Advanced SFR prototype plant by 2028 in association with the pyroprocess technology development. To support this R&D plan, KAERI has been focusing on the development of an advanced design concept of a burner reactor, which satisfies the future goals of safety, economics, sustainability, and proliferation resistance. In addition, R&D activities have been worked to achieve a safe and reliable advanced SFR design, such as large scale sodium thermal-hydraulic test facilities, a supercritical CO₂ Brayton cycle system, an under-sodium viewing technique, metal fuel, and a safety analysis code.

Acknowledgment

This work was supported by Nuclear Research & Development Program of the National Research Foundation Grant funded by the Ministry of Education, Science and Technology in Korea.

References

- [1] "The 5th basic plan for long term," Electricity Supply and Demand (2010–2024) 2010-490, Ministry of Knowledge Economy, 2010.
- [2] D. H. Hahn, "Status of the fast reactor technology development program in Korea," in *Proceedings of the 40th Technical Working Group on Fast Reactors Meeting (TWG-FR '07)*, Tsuruga, Japan, May 2007.
- [3] D. Hahn et al., "KALIMER-600 conceptual design report," KAERI/TR-3381/2007, Korea Atomic Energy Research Institute, Daejeon, Korea, 2007.
- [4] H. Song, S. J. Kim, H. Y. Jeong, and Y. I. Kim, "Design studies on a large-scale sodium-cooled burner," in *Proceedings of the International Conference on Advances in Nuclear Power Plants (ICAPP '08)*, pp. 450–457, June 2008.
- [5] IAEA, Fast Reactor Database 2006 Update, IAEA-TECDOC-1531, 2006.
- [6] D. E. Kim, M. H. Kim, J. E. Cha, and S. O. Kim, "Numerical investigation on thermal-hydraulic performance of new printed circuit heat exchanger model," *Nuclear Engineering and Design*, vol. 238, no. 12, pp. 3269–3276, 2008.
- [7] J. E. Cha et al., "Report on the design concepts of Na/CO₂ heat exchangers," Gen IV International Forum Report SFR/CDBOP/2011/015, 2011.
- [8] J.-H. Eoh, H. C. No, Y. H. Yoo, and S. O. Kim, "Sodium-CO₂ interaction in a supercritical CO₂ power conversion system coupled with a sodium fast reactor," *Nuclear Technology*, vol. 173, no. 2, pp. 99–114, 2011.
- [9] J. H. Eoh, H. C. No, Y. H. Yoo, J. Y. Jeong, J. M. Kim, and S. O. Kim, "Wastage and self-plugging by a potential CO₂ ingress in a supercritical CO₂ power conversion system of an SFR," *Journal of Nuclear Science and Technology*, vol. 47, no. 11, pp. 1023–1036, 2010.
- [10] Y. I. Chang, "Technical rationale for metal fuel in fast reactor," *Nuclear Engineering and Technology*, vol. 39, no. 3, 2007.
- [11] C. L. Trybus, "Injection casting of U-Zr-Mn, surrogate alloy for U-Pu-Zr-Am-Np," *Journal of Nuclear Materials*, vol. 224, p. 305, 1995.
- [12] C. T. Lee et al., "Casting technology development for SFR metallic fuel," in *Proceedings of Global-2009*, Paris, France, September 2009.
- [13] R. E. Macfarlane, "TRANSX 2: a code for interfacing MATXS cross-section libraries to nuclear transport codes," LA-12312-MS, 1992.
- [14] R. E. Alcouffe et al., "DANTSYS: a diffusion accelerated neutral particle transport code system," LA-12969-M, 1995.
- [15] B. J. Toppel, "A user's guide for the REBUS-3 fuel cycle analysis capability," ANL-83-2, 1983.