ADVANCED SFR DESIGN CONCEPTS AND R&D ACTIVITIES

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In order to meet the increasing demand for electricity, Korea has to rely on nuclear energy due to its poor natural resources. In order for nuclear energy to be expanded in its utilization, issues with uranium supply and waste management issues have to be addressed. Fast reactor system is one of the most promising options for electricity generation with its efficient utilization of uranium resources and reduction of radioactive waste, thus contributing to sustainable development. The Korea Atomic Energy Research Institute (KAERI) has been performing R&Ds on Sodium-cooled Fast Reactors (SFRs) under the national nuclear R&D program. Based on the experiences gained from the development of KALIMER conceptual designs of a pool-type U-TRU-10%Zr metal fuel loaded reactor, KAERI is currently developing Advanced SFR design concepts that can better meet the Generation IV technology goals. This also includes developing, Advanced SFR technologies necessary for its commercialization and basic key technologies, aiming at the conceptual design of an Advanced SFR by 2011. KAERI is making R&D efforts to develop advanced design concepts including a passive decay heat removal system and a supercritical CO₂ Brayton cycle energy conversion system, as well as developing design methodologies, computational tools, and sodium technology. The long-term Advanced SFR development plan will be carried out toward the construction of an Advanced SFR demonstration plant by 2028.

KEYWORDS : Sodium, Fast Reactor, Burner, Metal Fuel, Passive, Na-CO₂

1. INTRODUCTION

It has been recognized that one of the most promising nuclear options for electricity generation is a fast reactor system which efficiently utilizes uranium resources and reduces radioactive wastes from nuclear power plants, thus contributing to sustainable development. In response to this recognition, the widespread concern about the management of spent fuel that mainly contributes to nuclear waste has caused us to develop Sodium-cooled Fast Reactors (SFRs) as one of the most promising future types of reactors in Korea.

The PWR-SFR transition scenario study showed that SFRs can substantiate domestic waste management claims in Korea [1]. A timely deployment of SFRs with different conversion ratios, breakeven reactors and burners, and a recycling of transuranics (TRUs) by reusing PWR spent fuel in SFRs can alleviate the potential limits to a domestic nuclear energy growth. This development is associated with the availability of uranium and the need to reduce nuclear waste disposal mainly constituting of PWR spent fuel. The SFR mix ratio in the nuclear fleet near the year

2100 is estimated to be approximately 35-40%. PWRs will remain as the main power reactor type until 2100, and SFRs will support the waste minimization and fuel utilization efforts.

The Generation IV (Gen IV) SFR is expected to be available commercially by around 2030 in advance of other Gen IV reactor systems. The Korea Atomic Energy Research Institute (KAERI) is now developing a conceptual design of an Advanced SFR under the third national mid- and long-term nuclear R&D program, newly launched as a 10-year program in 2007.

The KALIMER-600 design [2] that evolved on the basis of the KALIMER-150 design, will serve as a starting point for developing the new advanced design. This new design will adopt advanced design concepts and features with the potential to better meet the Gen IV technology goals of sustainability, safety and reliability, economics, proliferation resistance, and physical protection. A conceptual design of the Advanced SFR will be developed by 2011, consistent with the Gen IV SFR System Research Plan.

R&D efforts will be made for the conceptual design

of the Advanced SFR with emphasis on the core and reactor systems, and on the development of the Advanced SFR technologies necessary for its early commercialization. R&D activities are being performed to improve the economics, safety assurance, and metal fuel performance, and these activities include the development of a Passive Decay Heat Removal Circuit (PDRC) system, a supercritical CO₂ Brayton cycle energy conversion system, design methodologies, computational tools, and sodium technology.

The long-term Advanced SFR R&D plan authorized by the Korean Atomic Energy Commission in December 2008 will be carried out toward the construction of an Advanced SFR demonstration plant by 2028 in association with the pyroprocess technology development in three phases: (1) First Phase (2007-2011)- Development of an Advanced SFR design concept, (2) Second Phase (2012-2017)- Standard design of an Advanced SFR demonstration plant, and (3) Third phase (2018-2028)- Construction of an Advanced SFR demonstration plant. This paper will present the Advanced SFR design concepts developed so far and the R&D activities being performed for the conceptual design of an Advanced SFR.

2. ADVANCED SFR DESIGN CONCEPTS

2.1 Plant Design

Various advanced design concepts have been proposed

and evaluated against the design requirements which were established to satisfy the Gen IV technology goals. In order to improve the economics, the rated power was increased from 600 MWe of KALIMER-600 to 1,200 MWe [3]. Breakeven cores loaded with metallic fuels did not have blankets in order to strengthen the proliferation resistance and employed safety grade residual heat removal system, called PDRC, which operates passively by natural circulation. Table 1 shows the key design parameters of Advanced SFR being developed at KAERI.

2.2 Reactor Core Design

Two types of conceptual core designs, Breakeven and TRU burner cores, were developed. Breakeven core is a reference concept for the Advanced SFR of 1,200 MWe. According to the current study [4], TRU burning rate increases linearly with the rated core powers from 600 MWe to 1,200 MWe. Considering 1) the realistic size of the SFR demonstration reactor planned to be constructed by 2028 according to the long-term R&D plan, and 2) the availability of a KALIMER-600 reactor system design, a TRU burner of 600 MWe was selected.

2.2.1 Breakeven Core

In order to develop a conceptual core design for a 1,200 MWe SFR, which enhances the economic potential and safety of KALIMER-600, the proposed core design concept has the following features: 1) a zoning of the core

Table 1. Key Design Parameters of Advanced SFR

Overall		PHTS	
Net Plant Power (MWe)	1,200.0	Reactor Core I/O Temp. (°C)	390/545
Core Power (MWt)	3,046.4	Total PHTS Flow Rate (kg/s)	15,455.4
Gross Plant Efficiency (%)	41.9	Primary Pump Type	Centrifugal
Net Plant Efficiency (%)	39.4	Number of Primary Pumps	4
Reactor	Pool Type	IHTS	
Number of IHTS Loops	2	IHX I/O Temp. (°C)	325/528
Safety Decay Heat Removal	PDRC	IHTS Total Flow Rate (kg/s)	11,777.7
Seismic Design	Seismic Isolation Bearing	IHTS Pump Type	Centrifugal
CORE		Total Number of IHXs	4
Metal Alloy Fuel Form	U-TRU-10%Zr	SGS	
Conversion ratio	1.0	Steam Flow Rate (kg/s)	1326.6
		Steam Temperature (°C)	503.0
		Steam Pressure (MPa)	16.5
		Number of SGs	2

into two sub-regions having different TRU enrichments, and 2) a reduced core height, in order to decrease the sodium void reactivity. As a fuel zoning concept, the enrichment-split core concept was preferable to the single enrichment core concept [5,6] from the view point of its neutron economy. The zoning with two sub-regions in a core, which makes a fuel fabrication simpler compared with a three sub-region core[2], was also proposed. A large core with higher rated power improves the fuel economy, however, it increases positively sodium void reactivity due to the reduced neutron leakage effect when a

Fig. 1. Layout of 1,200 MWe Breakeven Core

Table 2. Design and Performance Parameters of Advanced SFR Breakeven Core

Parameter	Breakeven Core
Electrical Power (MWe)	1,200
Core Height (cm)	80
Cycle Length (EFPD)	540
Fuel Rod Outer Diameter (mm)	8.7
Charged TRU Enrichment (w/o)	
- Inner Core	13.16
- Outer Core	16.79
Conversion Ratio	0.993
Fissile Pu Inventory (ton/GWe, BOEC)	5.06
TRU Inventory (ton/GWe, BOEC)	7.98
Burnup Reactivity Swing (\$)	0.63
Average Discharge Burnup (MWD/kg)	100.1
Peak Fast Neutron Fluence (n/cm ²)	4.72×10^{23}
Peak Linear Heat Generation Rate (W/cm)	320
Sodium Void Worth (\$)	7.25
Effective Delayed Neutron Fraction	0.00353

sodium void occurs. In order to reduce the sodium void reactivity, the reduction of core height was proposed instead of using an axial moderator layer below the active core used in KALIMER-600, which improves the fuel economy of the core by decreasing the required TRU inventory.

The configuration of 1,200 MWe breakeven core is shown in Figure 1 and its major design parameters and neutronic characteristics are summarized in Table 2. As shown in Figure 1, the total number of fuel assemblies was increased from 333 of KALIMER-600 to 624 in order to maintain the linear power density within the limit. The core height was reduced to 80 cm in order to decrease the sodium void reactivity by enhancing the neutron leakage for the sodium void case. The fuel outer diameter is 0.87 cm. The required TRU enrichments of the fuels are 13.2 wt% for the inner core and 16.8 wt% for the outer core. The average TRU enrichment of the fresh fuel assembly is about 14.95 wt%. The core has a conversion ratio close to 1.0. The required fissile plutonium inventory is 5.07 ton/GWe, which saves 20% of fissile plutonium loading compared to the KALIMER-600. The burnup reactivity swing is 0.69 \$, and the sodium void reactivity worth is 7.25 \$, which is less positive than that of KALIMER-600.

TRU inventory is about 7.98 ton/GWe, which is equivalent to the spent fuel discharged from one year operation of 28 PWRs with 1 GWe rated power. Once the TRU fuel is loaded into the core, there is no additional supply of external TRU. All of the TRU is supplied by recycling its spent fuel. The peak fast neutron fluence and the peak linear heat generation rate are within the design limits. The design limit of 5.0×10^{23} n/cm² for the peak fast neutron fluence is based on new clad material which will be developed in the future for the commercial breakeven core. Comparing the fissile plutonium inventory and discharge burnup, the advanced concept core reveals more attractive aspects of its economic potential. Even though the advanced conceptual core has a larger linear heat generation rate, it is observed that the clad inner-wall temperature is lower than 650°C under a normal operation during the cycle, which ensures that an eutectic melting is not expected during a normal operation.

The core neutronics design, with feedback from the fuel behavior, thermal, and safety analyses will be performed for the proposed candidate core concepts in order to finalize the core design concept for a 1,200 MWe SFR.

2.2.2 TRU Burner Core

A TRU burner core has been developed with a target TRU enrichment of 30 w/o which is the maximum enrichment in the currently established ternary metallic fuel database. In order to have a high TRU consumption rate, it is desirable to have a low conversion ratio. However, a low conversion ratio requires a high TRU enrichment which means that a burner with a TRU enrichment of 30 w/o has the maximum capacity of TRU burning within

the currently established metal fuel technology field.

For this study, a single enrichment fuel concept was adopted. The benefit of using single enrichment in a TRU burner core is that it helps to maintain the maximum charged TRU enrichment lower than that for enrichment zoning. Therefore, a TRU consumption can be maximized within the given TRU enrichment limit. The radial power distribution control was achieved by varying the region-wise cladding thickness, accompanied with a concurrent adjustment of the fuel slug diameter in order to maintain the smear density as constant, while using single enrichment for the whole core region. A clad outer diameter of 7.0 mm was adopted and the number of assemblies in each region was determined to minimize the peak to average power ratio in the core. The active core height was adjusted to make the sodium void worth close to 7.50 \$. The fuel pitch was allowed to vary in order to maintain a bundle pressure drop below 0.15 MPa. The designs maintained an average linear power of 180 W/cm.

The TRU burner core layout for 600 MWe is depicted in Figure 2. The transmutation characteristics and performance parameters are listed in Table 3. The TRU conversion ratio is 0.57 and the core is expected to consume 201 kg of TRU per cycle. The low inventory of

heavy metal in the TRU burner core results in assembly average discharge burnup of 128 MWd/kgHM.

The global reactivity feedbacks resulting from fuel temperature change, thermal expansion, and sodium density change in the equilibrium core are given in Table 4. The effective delayed neutron fractions (β_{eff}) at BOEC and EOEC are 0.0032 and 0.0032, respectively, and the variation with burnup is negligible. The fuel temperature (Doppler) coefficients which reflect the change in reactivity as a function of the fuel temperature were evaluated for the sodium-flooded cases. From the regression analysis, the Doppler coefficients were estimated in the Doppler temperature relationship of $1/T^{1.11}$. The sodium void worth and sodium density coefficient are positive because the positive spectral hardening effect outweighed the negative

Table 3. Design and Performance Parameters of TRU Burner Core

Parameter	Breakeven Core
Electrical Power (MWe)	600
Core Height (cm)	89
Cycle Length (EFPD)	332
Fuel Rod Outer Diameter (mm)	7.0
Charged TRU Enrichment (w/o)	30.00
Conversion Ratio (fissile/TRU)	0.74/0.57
Fissile Pu Inventory (ton/GWe, BOEC)	4.07
TRU Consumption Rate (kg/cycle)	201
Burnup Reactivity Swing (\$)	10.93
Average Discharge Burnup (MWD/kg)	127.9
Peak Fast Neutron Fluence (n/cm ²)	4.25×10^{23}
Average Linear Heat Generation Rate (W/cm)	179
Max. Pressure Drop (MPa)	0.15
Max. Cladding Inner Wall Temp. (°C)	597

Fig. 2. Layout of 600 MWe TRU Burner Core

Table 4. Reactivity Coefficients of TRU Burner Core

Parameter	BOEC	EOEC
Doppler Coefficient (pcm/°C)	$-821.9T^{-1.113}$	$-819.6T^{-1.109}$
Axial Expansion Coefficient (pcm/°C)	-0.171	-0.181
Radial Expansion Coefficient (pcm/°C)	-0.708	-0.743
Sodium Density Coefficient (pcm/°C)	0.706	0.763
Control Rod Worth (pcm/control rod)	-475	-496
Sodium Void Worth (\$)	6.90	7.50
Effective Delayed Neutron Fraction	0.0032	0.0032

leakage effect. The sodium void worth was evaluated to be 6.90 \$ at BOEC and 7.50 \$ at EOEC, respectively.

The uniform radial core expansion due to the coolant temperature rise is one of the major negative reactivity insertion mechanisms in a metallic fuel reactor. The radial expansion coefficient defines the relationship between the reactivity and changes in the effective radius of the core region. In these calculations, the changes in the total radius of the core (outermost radial shield boundary) were considered instead of the effective radius. The uniform radial expansion coefficient is defined to be dependent upon the change in dimensions of the lower core support structure which in turn depends on the inlet coolant temperature. The radial expansion coefficients at BOEC and EOEC were estimated to be $-0.708 \text{ pcm}/^{\circ}\text{C}$ and $-0.743 \text{ pcm}/\%$, respectively and thereby insensitive to the burnup. The individual control rod worth at BOEC and EOEC was estimated to be 475 pcm and 496 pcm, respectively. The variation in the control rod worth with burnup was small because the power distribution was almost constant throughout the cycle owing to the use of a single enrichment concept.

2.3 Heat Transport System Design

The heat transport system of Advanced SFR was designed to be a pool type to enhance system safety through slow system transients, where primary sodium is contained

in a reactor vessel. The heat transport system is composed of Primary Heat Transport System (PHTS), Intermediate Heat Transport System (IHTS), Steam Generating System (SGS) and Residual Heat Removal System (RHRS). The heat transport system was established through trade studies in order to enhance the safety and to improve the economics and performance of the KALIMER-600 design. Trade studies were performed for the number of IHTS loops, the number of PHTS pumps, Steam Generator (SG) design concepts, energy conversion system concepts, cover gas operation methods, and an improved concept of safety-graded passive decay heat removal system. From the study, the heat transport system of Advanced SFR has design features such as two IHTS loops, a Rankine cycle energy conversion system, two double-wall straight tube type SGs, and a passive decay heat removal system as shown in Figure 3.

During the normal operation mode, reactor core heat is transferred to the SG through PHTS and IHTS via Intermediate Heat Exchangers (IHXs). The transported heat is used to convert water to superheated steam which is then supplied to a turbine generator for electricity generation. Operating parameters for 100% power condition were established, as shown in Figure 4, where the net plant efficiency is 39.4%.

The PHTS consists of PHTS pumps, IHXs and reactor internal structures. The PHTS pump is a centrifugal type

Fig. 3. Configuration of the Heat Transport System

Fig. 4. Plant Operating Parameters at 100% Power

Fig. 5. PDRC Configuration and Decay Heat Removal Process

mechanical pump with a capacity of 540 m³/min. The number of pumps was determined to be four by considering manufacturability, shaft length, and capacity. There are four cylindrically shaped IHXs which are counter flow shell and tube types (TEMA type S) with a vertical orientation in the reactor vessel where PHTS and IHTS sodium flow through the shell side and the tube side, respectively.

The IHTS consists of pipings, IHTS pumps, and SGs. The arrangement and relative elevation of the IHTS piping and components were designed to promote the natural circulation for decay heat removal right after the reactor shutdown. The initial natural circulation rate after shutdown from normal full power operating condition is 10 % of nominal flow. The cold leg of the IHTS piping is a bottom up U-shape with sufficient height to prevent sodium-water interface from reaching the IHX in case of SG tube failure. IHTS pump is a centrifugal type with a capacity of 404.2 m³/min and is located in each cold leg of the two IHTS loops. SG is a double-wall straight tube type with a thermal capacity of 1528.7 MWt. One SG is installed in each IHTS loop, and the inner diameter and the length of a heat transfer tube are 13.8 mm and 28.3 m, respectively. The IHTS sodium flows downward through the shell side while the water/steam goes up through the tube side. Steam from the SG at 100 % normal operating condition is at a superheated condition of 503°C and 16.5 MPa. The feedwater system consists of main pumps, auxiliary pumps, flow control valves and heaters. Each SG has its own main and auxiliary flow control valves, but the two SGs share the same feedwater pump.

The RHRS of Advanced SFR is a 3-layer system which consists of PDRC, an Intermediate Reactor Auxiliary Cooling System (IRACS), and a steam/feedwater system. During the normal plant operation, in order to cool down from power operation to refueling mode, the reactor core heat is removed through heat transport path of PHTS, IHTS, and main condenser cooling of SG systems. However, if the normal heat transport path is not available, IRACS and PDRC are to be used to remove the decay heat of the reactor core. The IRACS is an active system and it consists of components operating on off-site power, and the PDRC system operates by the natural circulation of sodium and external air without any active components or operator's actions.

The IRACS is comprised of two independent loops connecting each IHTS loop. Each loop consists of the tube-side IHX, the Forced-Driven sodium-air Heat eXchanger (FDHX), sodium piping connecting the IHX with the FDHX via a part of the IHTS loop, a motor-driven air blower and isolation dampers installed at the upper part of the air stack, a single electro-magnetic pump, and two isolation valves located at the IRACS cold leg and the sodium expansion tank.

The PDRC consists of four independent loops, and each loop is equipped with a single sodium-to-sodium

Decay Heat Exchanger (DHX), single sodium-to-Air Heat Exchanger (AHX), and the piping connecting the DHX with the AHX. During normal plant operation, the DHX is partially dipped into the cold pool sodium in order to prevent unexpected freezing of the PDRC loop sodium. Under accident conditions such as a total loss of normal heat sink, the level of cold pool is raised to that of the hot pool because of the loss of head difference between hot and cold pool, similar to the PHTS pump trip following the reactor shutdown as depicted in Figure 5. After the reactor shutdown, the level of sodium increases from the expansion of primary sodium due to accumulation of reactor core decay heat.

If the sodium level increases higher than slots of DHX, the hot pool sodium overflows into the shell-side of DHX. As the sodium flow rate through the shell-side of DHX increases, the heat transfer rate through the DHX increases due to the enhancement of convection heat transfer. The heat transferred to PDRC is finally dissipated into the atmosphere through AHX by the natural circulation in the PDRC loop.

2.4 Mechanical Structure Design

In order to secure the economic competitiveness of a SFR compared to the pressurized water reactor, several concepts were implemented in the mechanical structural design without losing the reactor safety level. Figure 6 shows the reactor internals and components arrangement in the reactor vessel. The material of reactor vessel and

Fig. 6. Preliminary Component Arrangement

Fig. 7. Preliminary NSSS Arrangements of Two Loop System

Fig. 8. IHTS and IRACS Arrangement of Two Loop System

internal structure is a Type 316 stainless steel. The outer diameter of the reactor vessel is 14.5m, which is a very compact size compared to the other designs.

The primary system consists of 4 sets of primary pump, IHX, and DHX in the reactor vessel. Each intermediate heat transport loop has a mechanical type pump and a SG connected by large diameter pipes as shown in Figure 7. The piping material is a Mod.9Cr-1Mo, which can shorten the piping length to about 60m, compared to stainless steel, because of the higher mechanical strength and a low thermal expansion. The piping diameters for the hot and cold legs are 80 cm and 110 cm, respectively.

As shown in Figure 8, the piping length of each loop is relatively long (about 180m) because of the inverse U shape piping layout which prevents the pressure propagation to the reactor vessel when a sodium-water reaction accident occurs in SG upper tube. Since this long piping layout increases the maintenance and construction cost, the shortening of the piping total length is necessary by properly adjusting over-excessive safety design level.

As for mechanical design, two new design strategies were adopted for the economic improvement of the NSSS. One was an integrated component of a refueling machine and an In-Service Inspection (ISI) tool with a long life waveguide sensor for reactor internals. This integrated component can reduce ISI time for reactor internals and the overall refueling period can be reduced by about 2 days. The other was a Leak Before Break (LBB) technology application for IHTS piping and RV, which can reduce the construction cost by removing the unnecessary large scale facilities to protect against large sodium leak accidents.

3. R&D ACTIVITIES

Various R&D activities have been performed in order to support the development of advanced design concepts and to update computational tools. These activities include validating neutronics analysis codes, PDRC experiment, the conceptual design of supercritical carbon dioxide (S-CO₂) Brayton cycle system, Na-CO₂ interaction test, under-sodium viewing technique, computerization of structural integrity evaluation, sodium technologies, and metal fuel.

3.1 Validation of Neutronics Analysis Codes

The validation of the reactor core analysis code system called K-CORE was carried out with the measured data from the critical experiment. The code system consists of the multi-group cross section library derived from the evaluated nuclear data file, effective cross section generation code, static core analysis code with depletion module, and the perturbation code for reactivity coefficient calculation as shown in Figure 9. The major part of the code system, such as the neutron flux solver, effective cross section generation code and depletion and fuel cycle analysis code will be replaced by new codes which are now under development at KAERI.

The validation effort has been concentrated on

Fig. 9. Code System Diagram for the SFR Core Analysis

evaluating the impact of the evaluated nuclear data file on the integral parameter since the uncertainty of the cross section data plays a dominant role in the uncertainty of the integral parameter in SFR core design. Table 5 shows the current accuracy of the core analysis code validated against three critical assemblies, BFS-73-1, BFS-75-1, and BFS-55-1.

In order to evaluate the impact of the evaluated nuclear data file on the integral parameter, BFS critical assemblies were analyzed by using up-to-date nuclear data files, ENDF/B-VII.0, JEFF-3.1, JENDL-3.3, and JENDL-AC2008. The dependency of the evaluated nuclear data file was observed for the U-238 inelastic scattering, Zr absorption, and Na scattering cross sections on the predicted effective multiplication factor and sodium void reactivity worth [7-9]. Differences in the U-238 inelastic scattering cross sections yielded the difference in the effective multiplication factor by around $1\% \Delta k/k$. The updated Na scattering cross sections in JEFF-3.1 tended to improve the accuracy in the predicted sodium void reactivity, which is more significant in the U fueled critical assemblies. These dependencies will be reflected into the selection procedure of the appropriate cross section data in the near future.

The validation of the cross section data aims to help in the development of the adjusted multi-group cross section library. For this target, KAERI is now developing the sensitivity and uncertainty evaluation code for SFR based on the standard and generalized perturbation theories.

3.2 PDRC Experiment

A large scale sodium thermal-hydraulic experimental facility was designed for verifying the advanced design concept of the PDRC in KALIMER-600, which is focused on assessing its cooling capability during the long- and short-term periods after the 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 begin in 2013 after the startup test in 2012.

The main test section of the experimental facility is composed of a PHTS and a PDRC which are scaled-down from the KALIMER-600 design. The test section includes all major components in the PHTS reflecting the real configuration. The preliminary concept of the main test section is shown in Figure 10.

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 was designed to comply with proper scaling criteria for geometric, hydrodynamic and thermal similarities. The key design parameters of the experimental facility from integral scaling consideration are summarized in Table 6. The reactor vessel height and diameter are about 3.7 m and 2.3 m, respectively. The reactor core is simulated by electrical heaters of 1.9 MW capacity which corresponds

Table 5. Accuracies of the Predicted Core Parameter for BFS Critical Assemblies

Parameter	Accuracy
Reactivity	$< 0.6\% \Delta k/k$
Control rod worth	$< 5\%$
Sodium void reactivity	$< 18\%$ for BFS-75-1 $< 4\%$ for BFS-55-1
Fission rate distribution	$< 3\%$ for U-235 and Pu-239 fission
Effective delayed neutron fraction	$< 4\%$
Spectral indices or reaction rate ratio	
F49/F25 ^{a)}	$< 1\%$
F28/F25	$< 8\%$
C28/F49	$< 6\%$
F37/F49	$< 4\%$
C37/F49	$< 2\%$
F51/F49	$< 6\%$
F53/F49	$< 14\%$
F48/F49	$< 16\%$
F40/F49	$< 10\%$
F64/F49	$< 1\%$

^{a)} XYZ/XYZ : X=reaction type (F=fission, C=capture),
Y=last digit of atomic number,
Z=last digit of mass number

Fig. 10. Preliminary Concept of PDRC Experimental Facility

Table 6. Key Design Parameters of PDRC Experimental Facility

Parameter	Symbol	Scaling Ratio (Model/Prototype)	
		Expression	Design Data
Length Ratio	l_R	l_R	1/5
Area Ratio	a_R	a_R	1/25
Volume Ratio	V_R	$l_R \times a_R$	1/125
Temperature Rise/Drop Ratio	ΔT_R	-	1
Velocity Ratio	u_R	$\sqrt{l_R}$	1/2.24
Time Ratio	t_R	$\sqrt{l_R}$	1/2.24
Gravity Acceleration Ratio	g_R	-	1
Core Power Density Ratio	$\tilde{q}_R$	$1/\sqrt{l_R}$	2.24
Power Ratio	q_R	$a_R \sqrt{l_R}$	1/55.9
Flow rate Ratio	$\dot{m}_R$	$a_R \sqrt{l_R}$	1/55.9
Pressure Drop Ratio	ΔP_R	l_R	1/5
Aspect Ratio	AR_R	$l_R/\sqrt{a_R}$	1

Fig. 11. S-CO₂ Brayton Cycle for Advanced SFR

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 ton. Operating temperatures of the system are preserved in the experiment. In the test, the natural circulation cool-down capability of the PDRC in conjunction with the system will be investigated for various design basis events.

3.3 Conceptual Design of S-CO₂ Brayton Cycle System

A feasibility study for the S-CO₂ Brayton cycle as a power conversion system was conducted at KAERI [10]. A thermal balance of the heat transport system shown in Figure 11 was established for 100 % power operating condition of Advanced SFR. The S-CO₂ Brayton cycle adopts two recuperators to increase its cycle efficiency, and two compressors are also utilized to avoid a sharp change of the physical properties near its critical point. The flow-split ratio to the downstream of Low Temperature Recuperator (LTR) becomes an important parameter. From the considerations of the cycle efficiency and the size of heat transfer area of LTR, the optimum value of the flow-split to cooler was determined to be 71%.

In order to produce the preliminary design parameters of turbomachinery in the cycle, design codes were developed for an axial turbine and a radial type compressor. The S-CO₂ turbine is a four stage axial type and the S-CO₂ compressors are two and three stage centrifugal types, respectively. Based on the design codes, design parameters were prepared to configure KALIMER-600 S-CO₂ turbomachinery models. Three dimensional shapes were configured with the ANSYS-BladeGen and a series

Fig. 12. S-CO₂ Turbine Performance Curves (Design Point, Pressure Ratio=2.75)

of flow analysis was conducted with commercial CFX code. Using the models, off-design characteristics and the sensitivities of the performance of the S-CO₂ turbomachinery were investigated and typical turbine performance curves were generated as shown in Figure 12.

To enhance the safety and economics, various kinds of heat exchangers were investigated and a Printed Circuit Heat Exchanger (PCHE) was selected for S-CO₂ Brayton cycle energy conversion system. A one-dimensional analysis computer code was developed to evaluate the performance of the PCHE and its design data was produced. In parallel with the typical PCHE design, an airfoil shape fin heat exchanger was newly designed to improve its performance. The new design concept was evaluated by

three-dimensional CFD analyses, which showed that the new concept can conserve a total heat transfer rate and reduce a pressure drop to 1/14 compared to a typical PCHE design as shown in Figure 13.

The MMS-LMR code was developed to analyze the transient phenomena in a SFR with a S-CO₂ Brayton cycle. A simple power/load reduction and recovery event was selected for the transient calculation.[11] Then, the transient behavior was analyzed for the KALIMER-600

S-CO₂ Brayton cycle. Figure 14 shows the transient analysis results.

3.4 Na-CO₂ Interaction Test

Even though the S-CO₂ Brayton cycle energy conversion system has many advantages, there still exists a possibility for a potential tube rupture accident due to a high-pressure difference between the tube-side CO₂ gas and shell-side liquid sodium. In general, the basic mechanism of a Na-CO₂ interaction can be divided into two distinct regimes such as a rate-controlled regime and a diffusion-controlled regime, and the temperature of the reaction zone is the key parameter to distinguish the reaction mechanism.

The chemical reaction regime is shown in Figure 15, and proper reaction models for each reaction regime have been proposed by the Arrhenius formula. The model parameters employed in each model are to be experimentally determined by using the results of Na-CO₂ interaction test.

For the verification of the proposed chemical interaction model, two kinds of test section will be used as shown in Figure 16. The first one will be a surface reaction test, and the other one will be the capsule test section for a gas injection test.

The purpose of the surface reaction test is to quantify the reaction rate for various sodium temperatures and to determine detailed kinetic parameters coupled with a mass diffusion process. The test apparatus consists of a main test body for the reaction channel test and associated

Fig. 13. Comparison of Heat Transfer and Pressure Drop between the Typical PCHE and Airfoil Fin Heat Exchanger

Fig. 14. Flow Rate and Temperature Change in the PHTS/IHTS and the Na-CO₂ Inlet/Outlet in the S-CO₂ Side for the Transient Operation

Fig. 15. Na-CO₂ Chemical Reaction Regime Fig. 17. Images for a Sodium Ignition ($T_{Na} \sim 600^{\circ}\text{C}$) Fig. 16. Experiment Rigs for the Na-CO₂ Interaction Test

components including a sodium storage tank, vapor trap, gas supply system, vent line, instrumentation and gas sampling system. High purity CO₂ gas is supplied and flows in a rectangular duct channel formed above the free surface of the liquid sodium filled in the sodium tray. Quartz glass windows (40mm × 20mm) are installed at both side-walls of the main body, and reaction behaviors are clearly observed and recorded as digital images through these windows.

For the gas injection test, a direct CO₂ gas injection

will be made to simulate more practical accident conditions. In this test, the CO₂ gas injection nozzle was designed to be located at the bottom of the reaction capsule, and thus the CO₂ gas bubbles moving upward direction react with the liquid sodium continuously. The test apparatus consists of a main test body for the capsule test and relevant components like a gas supply system, vapor trap, instrumentation and gas sampling system.

The initial temperature of the liquid sodium and supplied CO₂ gas are used as key parameters for the test conditions, and they are carefully determined by reflecting system operation parameters of the S-CO₂ Brayton cycle coupled with the SFR system. The sodium temperature is set from 200°C to 650°C, and the preparative heating of the injected CO₂ gas is set from 200°C to 600°C. Hence isothermal conditions between liquid sodium and serviced CO₂ gas were made (within ±20°C) to simulate actual design conditions during the CO₂ ingress accident. The flow rate of the serviced CO₂ gas is used as a controllable parameter ranging from 250 scc/min to 10 Standard Liter Per Minute (SLPM) for corresponding test conditions.

Based on the preliminary test results, a strong temperature dependency on the reaction mechanism was tentatively estimated, and vigorous ignition phenomena of the liquid sodium were observed near the sodium temperature of 600°C as shown in Figure 17. The various test conditions are currently being investigated and the detailed model verification to quantify the rate of Na-CO₂ interaction will be completed by the end of 2009.

3.5 Under-sodium Viewing Technique

Reactor core and in-vessel structures of SFR should be inspected by visual examination according to the ISI guideline of ASME Section XI Division 3 [12]. Since sodium is opaque to light, conventional visual examination

Fig. 18. Ultrasonic Waveguide Sensor Module

Fig. 19. Visualization Images Using the 10m Long Waveguide Sensor

is not appropriate for in-vessel structures under sodium. An ultrasonic wave should be applied for an under-sodium visual examination of the in-vessel structures. The ultrasonic waveguide sensor has been developed for the under-sodium viewing of in-vessel structures in opaque liquid metal sodium. Relevant theoretical and experimental research has been performed for the feasibility of studying the ultrasonic waveguide sensor technologies [13].

An experimental facility was constructed for feasibility tests in water. It consists of a scanner, a high power ultrasonic system, a computer system, an oscilloscope, and waveguide sensor modules. A 10 m long waveguide sensor module was designed and fabricated. The waveguide sensor module has a slender cylindrical tube structure, and it is vertically supported at the reactor head. The waveguide sensor module consists of a commercial Piezoelectric Translator (PZT) ultrasonic transducer, a liquid wedge, a waveguide strip plate, and an acoustical shielding tube as shown in Figure 18. The ultrasonic transducer is installed over the reactor head.

A special liquid wedge is attached to produce an A0 Lamb wave in a low frequency range below 2 MHz. The waveguide strip plate guides the A0 Lamb wave. A visualization image can be achieved by scanning the waveguide sensor module above the core top plane and rotating the double rotating plug on the reactor head. The C-scan visualization experiment for a specimen with slits was performed in order to evaluate the sensitivity and

Fig. 20. Overall Contents of SIE ASME-NH Computer Code

resolution of the ultrasonic waveguide sensor.

The visualization image was clearly identified with a 2mm accuracy as shown in Figure 19. The feasibility of the ultrasonic waveguide sensor was demonstrated by C-scanning experiments in water. In the future, an ultrasonic visualization system will be set up and performance testing of the ultrasonic waveguide sensor technologies in both water and sodium environments will be conducted.

3.6 Computerization of Structural Integrity Evaluation

Carrying out an elevated temperature design of Gen IV reactor structures and components that complies with the ASME subsection NH rules takes a lot of effort due to the complicated application procedures of the ASME-NH rules. A computer program was developed and named as Structure Integrity Evaluation by using the ASME-NH rules (SIE ASME-NH). The detailed application procedures of the ASME-NH rules are programmed with the Fortran language step by step. The program can be used as a postprocessor of the ANSYS or ABAQUS finite element analysis software in order to evaluate the design integrity quickly and accurately, which saves the great engineering costs by above 70%. All the ASME-NH material data required for the evaluations can be generated automatically with proper interpolation or extrapolation methodology, which greatly reduces the designer's labor effort.

The SIE ASME-NH program consists of several modules of a primary stress limits evaluation, inelastic strain limits evaluation, creep-fatigue evaluation by an elastic and an inelastic analyses, and also material database. Figure 20 shows the overall contents of the SIE ASME-NH program. The input data file can be easily prepared with command based control cards, which are independently used for any required evaluation item.

3.7 Development of SFR System Transients Analysis Code

Advanced SFR design concepts employ enhanced safety features such as PDRC and IRACS for decay heat removal in accident situations. The large sodium pool for PHTS and the use of metallic fuel are also helpful to mitigate the severity of accidents. Care should be taken to evaluate the safety features and there is a need to develop an analysis tool with flexible modeling capabilities and an enhanced accuracy. Multi-dimensional Analysis of Reactor Safety-Liquid Metal Reactor (MARS-LMR) code [14] is being developed at KAERI as a simulation tool for the system transients safety analysis of advanced concepts of Liquid Metal Reactor (LMR).

In the MARS-LMR code, a table for liquid metal coolant thermal properties was implemented using the Equation Of State (EOS) based on the semi-empirical soft-sphere model [15]. For a description of the pressure drop in a wire-wrapped rod bundle, the correlations by

S.C. Cheng and N. E. Todreas [16] were implemented. The heat transfer by a liquid metal flow in nuclear fuel bundles is described by a modified Schadt's correlation [17]. For the liquid metal heat transfer in a heat exchanger tube, the correlation by Aoki [18] was selected and it has been found that the Graber-Rieger model [19] is the best for a shell side heat transfer.

The MARS-LMR code was used for the preliminary performance analysis of the PDRC for KALIMER-600 design to evaluate whether the formation of natural circulation is enough. Figure 21 shows transient temperature changes at different PDRC locations, and the trend of temperatures are consistent with an increase in the power transported to the PDRC loop. From these results, it can be said that the MARS-LMR code has been successfully developed for application to a natural circulation in a liquid metal-cooled reactor and a complete passive operation of a natural circulation heat removal system

Fig. 21. Transient Temperature Change for PDRC Performance Test

Fig. 22. The Comparison of Outlet Temperatures for SHRT 17, 39 and 45

can be modeled and analyzed successfully with the code.

In order to validate using experimental data, Shutdown Heat Removal Tests (SHRT) that were conducted in the Experimental Breeder Reactor-II (EBR-II) were also analyzed by the MARS-LMR code. The SHRT tests are mainly aimed at demonstrating passive removal of decay heat following a reactor shutdown from a natural circulation of the primary coolant. These tests also demonstrate a passive reactor shutdown following a loss of forced circulation. Three tests, SHRT-17, 39, and 45, which include a loss of electrical power to all primary pumps, were selected to validate the MARS-LMR code. These tests have different periods of primary pump coastdown from 100 % flow to a natural circulation flow of about 3 %. There was a reactor scram for SHRT-17 and no scrams for the other two tests [20].

Figure 22 shows the simulation results of the outlet temperature behaviors in the instrumented subassembly. During the early stage of transients, the calculation results of MARS-LMR are similar to the experimental data. It needs to be noted that the MARS-LMR analysis results agree well with the measured data for very low flow of natural circulation. It has been shown that the MARS-LMR code results are in good agreement with the SHRT test results and thus the code can be utilized to simulate various system transients accurately.

3.8 Sodium Technologies

3.8.1 Experimental Study on the Friction Loss Coefficient in a Micro Gap of Double Wall SG Tube

Even though there are reliable design solutions, there still exist concerns on potential sodium-water reactions when there is a leak in SG tubes. In order to avoid sodium-water reactions, a double-wall tube SG is being developed at KAERI. The objective of these experiments is to study the possibility of a leak detection method by measuring

the pressure in the gap between the inner tube and outer tube of a double wall tube. The experiments were done to measure the amount of leakage of Helium gas by pressing the test section using the hydraulic press in order for the pressure to be the same amount as the residual stress. The initial pressure that mimics the drawing pressure at the time of a manufacturing and the contact pressure that mimics the residual stress in the gap was established. The leakage in the gap was measured by controlling the pressure of Helium gas. An experimental apparatus has been set up to measure the amount of leakage through the gap by using the drawing pressure, the residual stress in the gap, and the surface roughness as experimental parameters.

Figure 23 shows the leakage characteristics in terms of the Reynolds number of the Helium gas. The results of the preliminary experiment show that the friction coefficient in the gap is inversely proportional to the Reynolds number (Re) of the Helium gas in the log-log plot. Thus, the frictional coefficient (f) in the gap can be expressed in terms of the Reynolds number as A/Re^B and it is approximately proportional to the $1/4$ power of the contact pressure. Here, A and B are the positive constants and d is the unit parameter for the surface roughness.

It is anticipated that the detection of the leakage can be made possible by measuring the pressure changes in a gap of a double wall tube with actual manufacture conditions.

3.8.2 Acoustic Leak Detection Test

The successful prevention of a water/steam leakage into SG sodium at an early phase depends on the fast response and sensitivity of a leak detection system. This study aimed at testing the acoustic leak detection tool by using KAERI sodium-water reaction signals for the leak detection of small water/steam-into-sodium leaks in the design of a SFR SG.

Fig. 23. Effect of Reynolds Number on the Friction Coefficient

Fig. 24. Performance Test Results Using KAERI Acoustic Leak Detection Methodology

Acoustic noise measurements were carried out in the sodium test facility of KAERI at a sodium temperature of 300~350 °C with flow rates of 0.004~0.54 g/sec. A detection tool performance of the developed acoustic leak detection methodologies [21] was tested by using the sodium-water reaction noises and the background noise of the PFR super-heater according to the learning conditions of the neural network as shown in Figure 24. It was possible to detect a leak up to S/N: -20dB at leak rate of 0.1 g/sec and to detect a leak up to S/N: -27dB at leak rate of 0.27 to 0.54 g/sec.

The current results obtained for water/steam-into-sodium leak detection can be utilized to develop a SFR SG acoustic leak detection system with several different kinds of background noises.

3.8.3 Wastage Characteristics of a Modified 9Cr-1Mo Steel Tube Material for a SFR SG

When a leak occurs, important phenomena such as “wastage”, may result in damage to or a failure of the adjacent tubes. It is very important to predict these phenomena quantitatively from the view-point of designing a SG and its leak detection systems. The objective of this study was to investigate the basic sodium-water reaction phenomena caused by small water/steam leaks. For this, wastage tests for modified 9Cr-1Mo steel were prepared.

The tube material wastage tests at KAERI were conducted in a small leak sodium-water reaction test facility-2. It mainly consists of a reaction vessel, sodium and steam supply system, and a drain system. The reaction vessel is a 13.8-in.-diameter by 25.6-in.-long stainless steel vessel. Circular type defects were used in these tests whose diameter ranged from 200~400 micrometers, and the targets of an actual tube shape and size were used. Steam was injected to the target from a steam supply

system through this assembly at a 150kg/cm² and 350°C. Modified 9Cr-1Mo steel was chosen for the test specimen material because this material was specified for the heat transfer tube for a KALIMER-600 SG.

Tests were made to determine the effect of the water leak rate, leak nozzle to target distance, and sodium temperature on modified 9Cr-1Mo steel wastage rate. The result for wastage rate versus leak nozzle to target distance for these tests in 400°C stagnant sodium is shown in Figure 25. It showed that the tube spacing had a significant effect on the tube material wastage rate. It also showed that the wastage rate increased as the leak rate increased.

A series of tests were conducted to clarify the wastage behavior of modified 9Cr-1Mo steel as a SG tube material for KALIMER-600. The data obtained from this study will be used to prepare the design criteria and design analysis procedures for these SGs from the viewpoint of sodium-water reactions.

3.9 Metal Fuel Technology

Metallic fuel was selected as a SFR fuel material to meet the four major requirements of the Gen IV SFR such as its economics, safety, sustainability and proliferation resistance. For sustainability and proliferation resistance, long-lived minor actinides such as Np, Am and Cm are transmuted in the reactor. To enhance the economics and safety, the fuel will be irradiated up to a higher burnup while maintaining its integrity.

For the fuel fabrication technology, practicality is a primary factor. Since fuel materials with minor actinides are radioactive, the fuel should be fabricated under a radiation-shielded environment such as a glove box or hot cell. Therefore, fuel fabrication technology with high reliability, simplicity and easy maintenance is being developed. An advanced fuel casting system with an induction furnace in conjunction with gravity casting, which can control transport of volatile elements during a melting process of fuel slug with minor actinides, was designed and installed. U-Zr fuel slugs satisfying design requirements in dimensions and microstructure were fabricated as shown in Figure 26. The microstructure examination and thermo-mechanical property tests such as heat capacity, thermal conductivity and thermal expansion were performed. The process optimization is currently being investigated to get the uniform microstructure of fuel slug containing volatile elements.

Fuel cladding is the most important fuel component since it maintains the safety of both the fuel and the reactor by preventing a release of radioactive fission products out of a fuel rod. It should withstand high temperature and high neutron fluence conditions. A cladding with a good creep resistance at high temperature, and with both the low swelling and the high ductility up to high neutron fluence, is desirable for the Gen IV SFR fuel. A batch of the Ferritic/Martensitic Stainless (FMS) cladding alloys were

Fig. 25. Relation Between Wastage Rate and Leak Nozzle to Target Distance

Fig. 26. Advanced Fuel Casting System and U-10Zr Fuel Slugs (6mm Dia. $\times$ 300mm L)

Fig. 27. Creep Strain Rate of New Cladding Alloys

designed and manufactured to evaluate the effects of minor alloying elements such as B and C on the mechanical properties. Creep tests of cladding alloys were performed at 650°C under constant load condition. Figure 27 shows that B003 alloy with 0.017 wt% boron and B004 alloy with 0.07 wt% carbon showed a lower creep strain rate than the Gr.92 steel, a reference alloy. In the future, optimization of both the alloy compositions using additives such as Nb, Ta and V, and cladding tube fabrication processes such as cold work and heat treatment, will be studied.

Fuel design, fabrication and fuel components should

be verified by performance tests and evaluations. Fuel irradiation behavior under all the postulated irradiation conditions needs to be predicted by modeling of fuel performance. To resolve one of the technical issues in metallic fuels, the possibility of a eutectic melting between a fuel metal and a cladding, a barrier concept between a fuel metal and a cladding was investigated. Interaction tests between a fuel metal and a cladding were performed to evaluate the effects of the various barriers such as Zr, Nb, Ti, Mo, Ta, V and Cr. Among these barriers, V and Cr exhibited the most promising performance. Cr and U-Zr did not react with each other and the elemental Cr remained after a 25 h diffusion couple test at 800°C, when the thickness of Cr coating was 38 μ m. Figure 28 shows that there was no inter-diffusion through Cr barrier between a fuel metal and a cladding. Fuel irradiation tests will be performed in HANARO research reactor under simulated fast reactor conditions such as fission density and temperature except fast neutron flux level.

4. CONCLUSIONS

Based on the experiences gained through the development of KALIMER conceptual designs characterized by a pool-type U-TRU-10%Zr metal fuel loaded reactor, KAERI is developing Advanced SFR design concepts that can better meet the Gen IV technology goals and the technologies necessary for its early commercialization, with the aim of establishing a conceptual

- sensor for under-sodium visualization of sodium fast reactor,” KAERI/TR-3540/2008, Korea Atomic Energy Research Institute, Daejeon, Korea (2008)
- [14] Ha, K. S., “Development of MARS-LMR and Steady-state Calculation for KALIMER-600,” Technical Report No. KAERI/TR-3418/2007, Korea Atomic Energy Research Institute, Daejeon, Korea (2007)
- [15] Young, D. A., “A Soft-sphere Model for Liquid Metal,” Technical Report No. UCRL-52352 (1977)
- [16] Cheng, S. K., and Todreas, N. E., “Hydrodynamic models and Correlations for Bare Rod and Wire-wrapped Hexagonal Rod Bundles - Bundle Friction Factors, Subchannel Friction Factors and Mixing Parameters,” Nucl. Eng. Des., 92, pp. 227-251 (1986)
- [17] Kazimi, M. S., and Carelli, M. D., “Heat Transfer Correlation for Analysis of CRBRP Assemblies,” Westinghouse Report, CRBRP-ARD-0034 (cited from Ref. 7) (1976)
- [18] Aoki, S., “Current Liquid-metal Heat Transfer Research in Japan,” Prog. Heat Mass Transfer, 7, pp. 569-573 (1973)
- [19] Graber, H., and Rieger, M., “Experimental Study of Heat Transfer to Liquid Metals flowing in-line through Tube Bundles,” Prog. Heat Mass Transfer, 7, pp. 151-166 (1973)
- [20] D. Mohr, L.K. Chang, E.E. Feldman, P.R. Betten, and H.P. Planchon, “Loss-of-Primary-Flow-without-Scram Tests: Pretest Predictions and Preliminary Results,” N.E.D. Vol. 101, pp. 45-56 (1987)
- [21] T.J. Kim, J.Y. Jeong, and S. Hur, “Performance Test for developing the Acoustic Leak Detection System of the LMR Steam Generator”, Transactions of the Korean Nuclear Society Autumn Meeting, Busan, Korea (2005)