Preliminary Study for Conceptual Design of Advanced Long Life Small Modular Fast Reactor

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1. Introduction

Under the International Nuclear Energy Research Initiative (I-NERI) Project between Ulsan National Institute of Science and Technology (UNIST) and Argonne National Laboratory (ANL), an advanced long life small modular fast reactor is under development.

As one of the non-water coolant Small-Modular Reactor (SMR) core concepts for use in the mid- to long-term, ANL has proposed a 100 MWe Advanced sodium-cooled Fast Reactor core concept (AFR-100) targeting a small grid, transportable from pre-licensed factories to the remote plant site for affordable supply [1]. Various breed-and-burn core concepts have been proposed to extend the reactor cycle length, which includes CANDLE [2] with a cigar-type depletion strategy, TerraPower reactors [3] with fuel shuffling for effective breeding, et al. UNIST has also proposed an ultra-long cycle fast reactor (UCFR) core concept [4, 5] having the power rating of 1000 MWe. By adopting the breed-and-burn strategies, the UCFR core can maintain criticality for a targeting reactor lifetime of 60 years without refueling.

The objective of this project is to develop an advanced long-life SMR core concept by adopting both the small modular design features of the AFR-100 and the long-life breed-and-burn concept of the UCFR. In this study, the feasibility of the long-life breed-and-burn core concept were assessed and the preliminary selections on the reactor design requirement such as fuel form, coolant material, primary cooling system, and steam cycle system was peformed.

2. Feasibility Study of Core Concepts

In this section, it has been assessed the feasibility of long life breed-and-burn core concepts using both deterministic and Monte Carlo code systems.

2.1 Long Life Breed-and-burn Core Concepts

Fast reactor core concepts such as UCFR-1000, AFR-100, CANDLE, and TP-1 (TWR), adopted breed-and-burn mode operation to extend the fuel cycle length. CANDLE has demonstrated the strengths of breed-and-burn strategy with its cigar-type geometry that is the simplified conceptual schematic of the geometry of 8m-tall CANDLE reactor concept as shown in Fig. 1. CANDLE reactor concept has an enriched uranium region that plays a role of an igniter and a blanket region along the axial direction as a breeder that utilizes natural uranium or depleted uranium for its material.

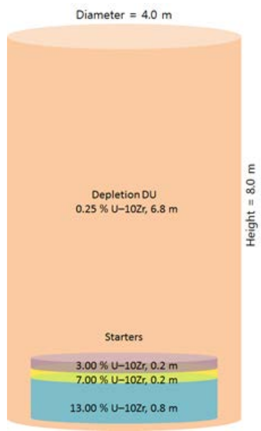

Fig. 1. Simplified conceptual schematic of CANDLE core

Figure 2 shows the result of the core depletion calculation. It is noticed that there is a saturation region in the graph for the multiplication factors, which means that a steady breeding is being proceeded along the axial direction. It is also presented that the movement of maximum power position whose trend line tells that the propagation speed is 3.4 cm per year.

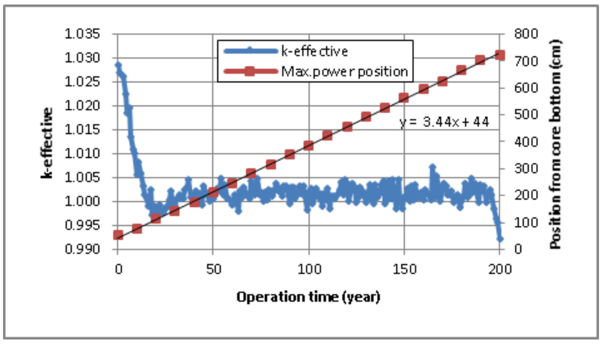

Fig. 2. Evolution of k-eff in CANDLE core

The propagation behavior of the CANDLE core can be also confirmed through the evolution of the axial power profile as shown in Figure 3. Once the core burns in the enriched uranium region at the core bottom where it shows relatively high power, the active core arrives at the depleted uranium region where the power profile and the speed of movement is steady.

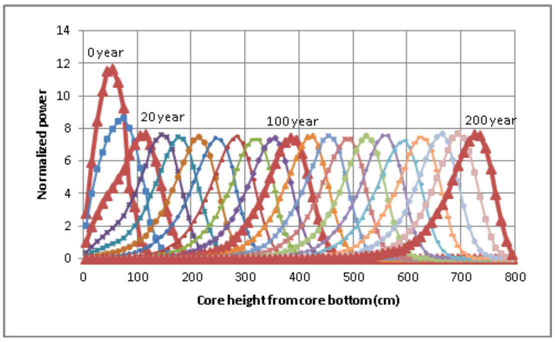

Fig. 3. Propagation of axial power profile in CANDLE

UCFR has been also chosen and analyzed to evaluate the feasible core life by utilizing breed-and-burn strategy. Figure 4 shows the feasibility test of the long life operation with modifying the core height which includes the LEU and blanket region. As the figure shows, only the red one has the feasibility of operating the long life operation of 35 years and it is noticeable that the blanket should be filled with enriched uranium instead of natural uranium because it cannot maintain the criticality by the natural uranium in this small geometry. It is expected that the power density can be increased by shortening the pin length when the blanket is enriched.

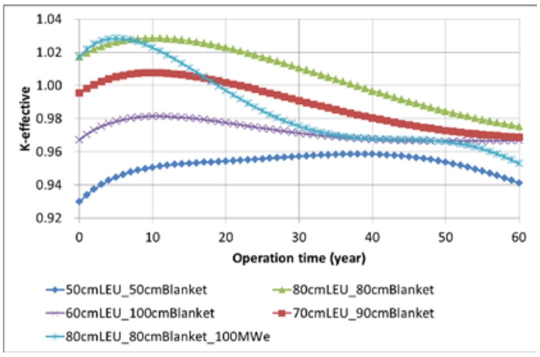

Fig. 4. Feasibility test of long life operation with UCFR-50 model

2.2 Coolant and Fuel Sections for long-life SMR core concepts

2.2.1 Coolant material

Typical fast reactor coolant is sodium, lead and LBE. Table I shows the material properties of some liquid metals. Coolant material is basically expected to have low melting point, density, viscosity and absorption cross section and a high boiling point, specific heat, and thermal conductivity. In this respect, sodium is preferable material for a coolant of nuclear power plant system. Lead and LBE have been also used and researched as a coolant material for its chemical stability but its high density and corrosion possibility can be a big issue when they are used in the long life reactor.

Table I. Coolant Material Properties

Properties at RT*	Na	Pb	LBE	He	Ga	GaIn	NaK
Atomic weight	23.0	207.2	208	4	69.72	-	34
Melting point (℃)	97.8	327.4	123.5	-272	29.76	15.7	-11.1
Boiling point (℃)	892	1737	1670	-269	2204	2000	783.8
Density (kg/m^3)	880	10500	10300	0.178	6095	6280	872
Specific heat (J/kg*K)	1300	160	146	5200	381.5	326	1154
Thermal conductivity (W/m*K)	76	16	11	0.152	31	41.8	25.3
Viscosity (c_P)	0.34	2.0-2.5	1.7	0.018	1.810	1.69	0.468
Absorption cross section (Σ_a)	0.01347	0.0056	0.00303	-	0.148	-	-
Coefficient of Volumetric thermal expansion ($10^{-6}K^{-3}$)	200	87	130	-	59.5	-	39.35

Figure 5 to 7 show the coolant performance of each material by comparing cross section and k_{eff} difference. The capture cross section of LBE is greater to that of sodium in fast energy region but the elastic cross section of LBE is greater than that of sodium, too. The k_{eff} of LBE reactor is greater than the k_{eff} of sodium reactor because the LBE prevents the neutron leakage from the core. However, the big problem of LBE coolant is its high corrosiveness. With high-performance reflector, sodium fast reactor is able to perform efficiently as LBE reactor. Because of coolant activation and the potential for sodium/water interactions between high-pressure steam and a low-pressure sodium loop, an intermediate coolant loop is recommended.

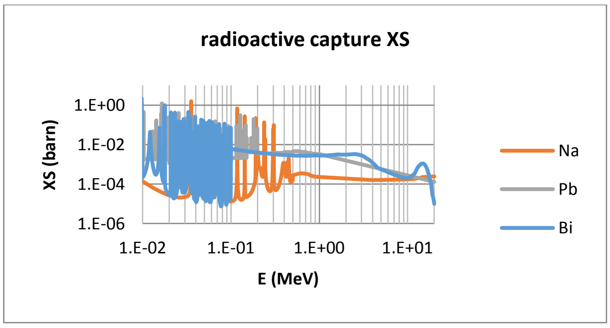

Fig. 5. Radioactive capture cross section by coolant material

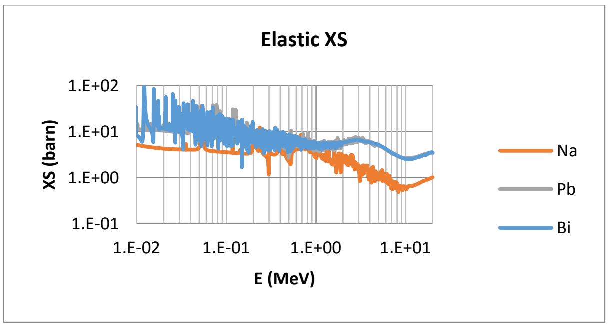

Fig. 6. Elastic cross section by coolant material

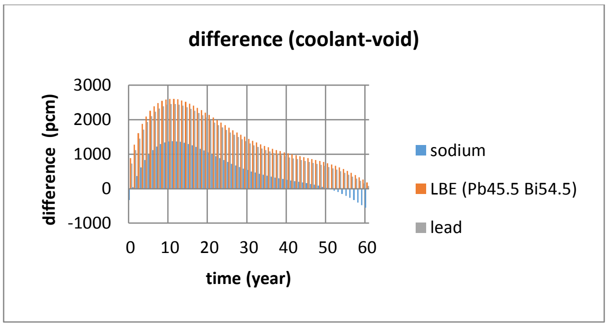

Fig. 7. K-eff Difference [coolant-void]

2.2.2 Fuel type

The neutron reaction characteristics of major elements at typical SFR are provided in Table II. TRU and U are considered as fuel materials. Iron is the primary structural material, and Zr, O, and C are additional elements of the metal, oxide and carbide fuels. The slowing down power of the most of fast reactor materials is less than 1% of hydrogen's in PWR. The slowing-down power of Zr is smaller than O and C, and the metallic fuel density is larger than oxide and carbide fuels. As a result, the metallic fuel has harder spectrum compared to the oxide and carbide fuel, which is favorable to increase the breeding and extend the cycle length.

Table II. Comparison of Neutron Reaction Characteristics of Major Elements

	Scattering XS (barn)	Atomic density (#/barn-cm)	Slowing down powers (cm-1)
TRU	4.0	3.2E-03	1.1E-04
U	5.6	5.6E-03	2.7E-04
Zr	8.1	2.6E-03	4.6E-04
O	3.6	1.4E-02	5.8E-03
C	3.9	1.6E-02	1.0E-02
Fe	3.4	1.9E-02	2.3E-03
Na	3.8	8.2E-03	2.7E-03
H (PWR)	11.9	2.9E-02	3.5E-01

3. General Requirements Study of Developing Reactor System

The requirements for the long life fast reactor core concept have been developed through the assessments and performance studies above. It is possible to have the small modular reactor breed-and-burn strategy by blending the long life fast reactor concept. The specifications have focused on the maximization of the advantages of each item and they have been evaluated for the aspect of both SMR and SFR.

3.1 Reactor Core

There are several design requirements for reactor core such as material type of fuel, coolant, reflector, structure, shield, control rod and some parameters such as power, core size, cycle length, peak temperature, maximum neutron flux, linear pin power density, discharge burnup, and etc. From the result of the study above, zirconium-uranium metal fuel and sodium coolant have been found to be a proper material. For its lots of experimental data and operation experience, HT-9 is the proper for reflector, structure, and shielding material. B_4C is used for control rod because of it has high absorption cross section in respect to fast neutron and its abundant data.

3.2 Pool vs. Loop

The major difference between the two types is whether the heat exchanger is inside the vessel or not, which brings many different characteristics for the two types. Table III summarizes general pros and cons of the pool type and loop type. Relatively large coolant inventory, large vessel size, and large site are no longer disadvantages when it comes to small size reactor. Therefore, the reactor type has been determined to pool type focusing on its advantages. Large thermal capacity is the key nature of pool type, which provides favorable safety characteristics and makes it preferable in core design.

Table III. The pros and cons of reactor type

	Pool type	Loop type
Pros	• Large heat capacity • Less Leakage • Simple design w/o branches • Hot coolant never comes into contact with the vessel wall	• Size is smaller than pool-type • It can be built in a factory and transported to the site • Easier to inspect
Cons	• More coolant inventory • Vessel is so large • Large building site is needed • It is built in on site because of size • Difficulty to inspect of internal structures during operation • Pipe stress	• Pipework is longer and more • Less experience with large-scaled loop type reactors
Example	BN600, PRISM, PHENIX, PFR, SUPER PHENIX I, CDFR I, BN1600	MONJU, JOYO, SNR300, CRBR, SNR 2

3.3 Pump

Table IV shows things to consider when selecting a pump. In the respect of small reactor, the drawbacks of electromagnetics pumps are negligible while the drawbacks of mechanical pumps are not in the respect of long cycle operation.

Table IV. Pros and cons for each kind of pumps

	Electromagnetic pumps	Mechanical pumps
Pros	• No rotating mechanical pieces • Very limited maintenance • Great reliability (for br10 as example, 170.000 hrs (~20 years) of operation without major incident (same for ancillary system of spx) • Very small impact of cavitation	• Large operational feedback from reactors • Good efficiency (70 ~ 80%) • Important inertia when stopped

Cons	◆ Low efficiency (maximum 40%) ◆ Risks of electromagnetic instabilities for large pumps ◆ Important component volume required for very large flow rates (ie. Some tons/s) ◆ Electrical insulation and magnetic materials working at 550℃ ◆ No operational feedback from large pumps immersed in reactor	◆ Several rotating elements ◆ Limited life duration for hydrostatic bearings ◆ Necessity to cool engines, bearings ◆ Necessity of periodical maintenance

3.3 Steam Cycle

Table V summarizes the characteristics of Rankine cycle and Brayton cycle. Rankine cycle is better for easy and short-term deployable application but it is inevitable for steam cycle to have the risk of water-sodium reaction which is a critical issue in SFR. On the other hand, Brayton cycle has not been used widely not only by SFR but also by the other reactors. Brayton cycle, however, is economically superior to Rankine cycle. First, it has a little possibility to allow water-sodium reaction that Rankine cycle should prepare vast budget to prevent. Second, it can realize high thermal efficiency which is the key criterion for thermal cycle. Brayton cycle has been evaluated to have the thermal efficiency of around 45 %, which brings more than 40 million dollar every year comparing Rankine cycle. Lastly, Brayton cycle has relatively small and cheap components. For the future-oriented reactor concept being developed in this study, Brayton cycle is valuable enough to be adopted.

Table V. Rankine cycle vs. Brayton cycle

Cycle	Rankine	Brayton
Characteristics	◆Most widely used steam cycle ◆Use liquid; water in general but ammonia, mercury ◆Closed cycle	◆ Gas turbine ◆ Higher temperature, higher efficiency ◆ Open & closed cycle (NPP uses closed with gas) ◆ small and cheap
For SFR	◆ traditionally adopted ◆ potential for energetic water–sodium reactions is a long-standing issue ◆ To prevent water-sodium reaction, expensive double wall and complex safety system needed	◆ SFR outlet temperature is between 500 ℃ and 550 ℃ due to metal clad peak temperature limit ◆ low efficiency: multiple reheat strategy and intercooling needed

3.4 SMR SFR Design Requirements

The design requirements have been studied above and the requirement parameters have been determined that is summarized in Table VI. Even though they can be changed in the future progress, the parameters are the basic criteria for this SMR SFR design development.

Table VI. Design Requirement Parameters for SMR SFR

Parameters	New model
Core thermal Output (MWth)	108
Core electric power (MWe)	50
Coolant	Sodium
Primary circulation	Pool
System pressure	Non-pressurized
Core inlet/outlet temperature (℃)	355 / 510
Thermodynamic cycle	Brayton cycle (SCO_2)
Fuel form	U-Zr
Design life (year)	30
Fuel cycle (year)	30
Pump	EM pump
Cladding and duct	HT-9
Reactor vessel diameter (cm)	~300
Maximum Neutron flux (#/cm^2s)	~$1x10^{15}$

4. Conclusions

A conceptual design of long life small modular fast reactor is under development by adopting both the small modular design features of the AFR-100 and the long-life breed-and-burn concept of the UCFR. The feasibility of the long-life fast reactor concepts was reviewed to obtain the core design guidelines and the reactor design requirements of long life small modular fast reactor were proposed in this study. Detailed core design parameters will be determined, and the core performance characteristics will be evaluated.

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