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공학박사 학위논문

Power Optimization Method for Land-Transportable Fully Passive Lead-Bismuth Cooled Small Modular Reactor Systems

육상 운송가능 완전피동 납-비스무스 냉각
소형모듈화 원자로 계통의 출력 최적화 방법론

2013년 2월

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에너지시스템공학부

조 재 현

Power Optimization Method for Land-Transportable Fully Passive Lead-Bismuth Cooled Small Modular Reactor Systems

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Abstract

Power Optimization Method for Land-Transportable Fully Passive Lead-Bismuth Cooled Small Modular Reactor Systems

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In order to take a good position in future energy spectrum, nuclear energy should be satisfied with future energy demand by overcoming the weaknesses of current Pressurized Water Reactors (PWRs); nuclear waste, nuclear safety, and nuclear economy. The spent nuclear fuels (SNFs) accumulated by operation of PWRs during last 50 years are over the 200,000 tons worldwide without any solution. From several nuclear accidents including Fukushima accident (March, 2011), nuclear energy is confronted with criticism about safety issues. Also, large amount of initial investment of current PWRs is chronic problem with blocking the activated investment of private companies.

As a future nuclear energy solution, long-burning technology and fully passive cooling Small Modular Reactors (SMRs) are emerging nuclear concepts. In order to overcome SNFs problem, long burning reactors using fast neutron spectrum could transmute high level waste into low or intermediate level waste. Also, utilization of ^{238}U in fast reactor is more efficient than it of thermal reactor. As a coolant for fast reactors, Lead-Bismuth Eutectic (LBE) coolant has many advantages; no production

hydrogen, no reaction with water and air, high capability of natural circulation, and negative void coefficients for small size core.

On the other hand, demand of fully passive cooling SMRs is increasing worldwide because they are simpler, standardized, and safer modular design by being factory built, requiring smaller initial capital investment, and having shorter construction times. They could be small enough to be transportable used in isolated locations without advanced infrastructure and without power grid, or could be clustered in a single site to provide a multi-module, large capacity power plant. Also fully passive cooling without pump even in normal operation enhances the passive safety of nuclear power plants. Thus, the solution integrated by long burning technology with LBE coolant and fully passive SMRs could solve the current weaknesses of PWRs by burning the nuclear waste, enhancing the nuclear safety, and increasing the nuclear economy.

For the reactor power size, power of SMRs should be maximized to maximize the economy of SMRs, having modularization fabricated remotely and transported to the site. Moreover, it is needed to have a specific power level matching the specific demand of towns or sites that are either off-grid or on immature local grids, being right-sized for growing economies and infrastructures of developing nations. Thus, the maximized power level of SMRs should be estimated. However, flexibility of the power is limited by land-transportable shipping size, materials endurance, long burning core neutronics, and accidents conditions. The dissertation is aiming at developing the power maximization method for LBE natural circulation cooled SMRs satisfying the constraints shipping size, materials endurance, neutronics as well as safety under beyond Design Basis Events (DBEs).

To achieve the goal of dissertation, three research questions are coming up: 1) what are limiting factors to design LBE natural circulation cooled SMRs, 2) what are design tools to design LBE natural circulation cooled SMRs and how are they validated, and 3) How to develop a power optimization method.

Design limitations are divided by limitations for steady state and accidents conditions. Steady state limitations including land-transportable shipping size limits, materials limits, neutronics limits are determined. The quasi-static reactivity balance equation is used to obtain the limitations of reactivity in accidents conditions for selected Beyond DBEs; Unprotected Transient OverPower, and Unprotected Loss Of Heat Sink. Void coefficients should be negative because steam could be penetrated into core in unprotected Steam Generator Tube Rupture.

For the design tools and validations, LBE coolant experiments using HELIOS (Heavy Eutectic liquid metal Loop for Integral test of Operability and Safety of PEACER) facility are conducted. In the forced convection test, pressure losses of core, orifice, gate valve, and expansion tank are obtained. In the natural circulation test, temperature distribution and mass flow rate are obtained for specific core heat. Also, long-term stability of LBE natural circulation is confirmed by 600-hours experimental test. Predictions for hydraulic-resistance and natural circulation behavior of experimental results are conducted by MARS-LBE and CFD (Computational Fluid Dynamics). Pressure loss coefficients of measured data are good agreement with CFD results and natural circulation experimental results are good agreement with MARS-LBE predictions when MARS-LBE uses the recommended pressure loss coefficients from CFD simulations. With comparison between measured data and natural circulation governing equations, natural circulation SMRs design equation is derived and validated and could be used for the optimization method.

Based on the previous answers including design limitations and design tools, power optimization method is developed with the flow chart. Using the power optimization method, natural circulation cooling capacity and neutronics maximized power could be calculated. Natural circulation cooling capacity is the capability of core power cooled by only natural circulation with fixed land-transportable shipping size limit and within the corrosion, erosion and DBTT limits. Neutronics maximized power is the capability of core power within materials Displacement Per Atoms limit, reactivity swing and excess reactivity swing limitations. From the comparison between these two power

capacities, smaller power for each core height is the maximized reactor power for each core height. Then, one specific maximized reactor power that the largest value with the core height is determined. Case study for 20 years long-burning small modular reactor with LBE natural circulation using the power optimization method shows the maximized power is 206MWt.

Keywords: long burning, fast reactor, lead-bismuth eutectic, small modular reactors, power maximization, natural circulation
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Chapter 1 Introduction

Twenty-five years have gone after Chernobyl nuclear accident (1986), which called the worst nuclear accident to date. Because worldwide nuclear power plants played an essential role safely in the energy spectrum during the period between Chernobyl and the early last year, it seemed that nuclear renaissance come to energy industry again. Unfortunately, Fukushima nuclear accident occurred in March 2011 by earthquake and tsunami in Japan and many nuclear power plants in worldwide have been opened to criticism.

Nonetheless, currently 64 plants are under construction worldwide [1] that is almost same with number of plants planned before Fukushima accidents. It does not means although Chernobyl and Fukushima accidents hit the nuclear industry, nuclear technology is unrivaled and no need to something to improve. It means that nuclear energy is still considered as the most reliable solutions to climate change and energy insecurity when it is compared with other energy source; coal, gas, and renewable energy. In other words, if renewable energy overcome their weakness such as low efficiency and low power density, many nations will determine the decreasing the nuclear energy portion to their energy spectrum.

On this account, nuclear energy should be improved to the new energy source that are very happy to future energy needs. It is necessary to review the weakness of current nuclear power plants and to suggest new nuclear energy solution for future energy which could overcome all weakness of current nuclear power plants.

This chapter describes significance of the research topic by describing the weakness in current pressurizer water reactor technology and new nuclear energy concepts including long burning and small modular reactor technology.

1.1 Weakness in current PWR technology

The almost type of current nuclear power plants is pressurized water reactor (PWR). There are three weakness in current PWR technology; nuclear waste, nuclear safety, nuclear economy.

As of September 2012 from first stage of nuclear power plant, 437 nuclear power units with an installed electric net capacity of about 372 GW in 30 countries are in operation. The past 50 year operation of nuclear power plants, however, results in a significant accumulation of spent nuclear fuels (SNF) which is currently over 200,000 tons worldwide in 2005 and estimated 700,000 tons up to 2050. Because of the high radioactivity requiring long management periods and the strong opposition of the general public, SNFs are becoming one of the most critical issues that must be resolved for the continued expansion of nuclear energy as an effective and sustainable energy source.

The significance of the SNF issue has been elucidated in over 20 years history of the Korean case. Since Kori-1 nuclear power plant was started in

1978 in the Republic of Korea, another 22 power plants are operating and 5 PWRs are under construction. Now, it is predicted that SNF storage pools of the operating NPPs in Korea will be saturated by 2016. However, even the construction of the central SNF interim storage, not to mention the final disposal site, was delayed due to the difficulties encountered in finding repositories sites, as well as vigorous public opposition. The Korean peninsula is geologically a very old terrain and underground bedrocks are finely divided. This situation represents a major challenge to the development of HLW repositories. In contrast, the permanent disposal site for Low and Intermediate Level Waste (LILW, equivalent to Low and Medium Level Waste) has been enthusiastically by Korean public accepted in 2005.

Today, several countries with a significant amount of accumulated SNFs are seeking for recycling options that may drastically reduce the toxicity and the volume of final waste forms. U.S.A. is also taking a look at the recycling approach.

The most important philosophy of design and operation on nuclear power plants is to minimize the likelihood of accidents, and avoid major human consequences when they occur. However, there have been three major reactor accidents in the 50-year history of civil nuclear power generation; Three Mile Island (TMI), Chernobyl, and Fukushima. Three Mile Island (USA 1979) where the reactor was severely damaged but radiation wan contained and there were no adverse health or environmental consequences. Chernobyl (Ukraine 1986) where the destruction of the reactor by steam explosion and fire killed 31 people and had significant health and environmental consequences. Fukushima (Japan 2011) where three old reactors (together with a fourth) were written off and the effects of loss of cooling due to a huge tsunami were inadequately contained [2]. These major nuclear reactor accidents obvious to give an adverse effect of the public acceptance to nuclear power plant.

The economic cost of nuclear power has been a key barrier to the construction of new reactors around the world. As an influential interdisciplinary study conducted at the Massachusetts Institute of Technology some years back stated, “Today, nuclear power is not an economically competitive choice” [3]. The lack of competitiveness arises mainly from its capital intensity. The ongoing electricity sector restructuring process around the world, leading to a greater emphasis on economic competition, has accentuated this problem. Financial risks that were previously borne by consumers are increasingly seen as the responsibility of investors.

The cost of generating electricity consists of three main components: the capital cost of constructing the generating facility, the annual fueling and operations and maintenance costs, and the waste management expenses.

1.2 New nuclear concepts for future energy

The above three weakness of current PWR technologies should be solved in order to keep the nuclear energy. Integration of long burning technology and small modular reactors with natural circulation could be solution. Each technology is described in following subsections.

1.2.1 Long burning technology

Long burning technology uses fast neutron spectrum which would utilize ^{238}U more efficiently than thermal reactors operating on thermal neutron spectrum. It is undoubtedly the most efficient system for the effective utilization of uranium resources, due to the possibility of using the uranium stored in used nuclear fuel to be recycled while producing energy. It is also possible to adopt the strategy of burning plutonium along with the uranium stored as tailings from enrichment plants. With these unique features, the energy potential of uranium increases significantly compared to light water reactors. In addition, the radioactive wastes containing long lived minor actinides become practically insignificant. Due to long refueling cycle, economics is enhanced by low refueling costs.

The requirement of a fast neutron spectrum implies the usage of coolants with low moderating power, such as sodium and lead-bismuth eutectic. Sodium has superior thermal hydraulic properties. There is a large experience with the operation of sodium-cooled fast reactors. While several power reactors have been shut down, BOR-60, JOYO, Phenix and BN-600 are still operating. Sodium features a reasonable low melting temperature, but also a low boiling point (892^{o}C), which raises safety concerns regarding unprotected transients leading to a coolant heat-up. Sodium exhibits high chemical activity with water and air. A limited sodium leak and fire has stopped the operation of the Japanese MONJU reactor since 1995. The choice of lead-bismuth eutectic coolants is motivated on the one hand by their high boiling points (1670^{o}C), which avoids the risk of coolant boiling. Lead-bismuth eutectic provides a low melting point (123.5^{o}C), limiting problems with freezing in the system and features a low chemical activity with water and air excluding the possibility for fire or explosions. A drawback connected with lead-bismuth eutectic is the accumulated radioactivity (mainly due to the α emitter ^{210}Po, $T^{1/2}$=138days), which could pose difficulties during fuel reloading or repair work on the primary circuit. However, IPPE Obninsk staff has developed methods to cope with the polonium during refueling and maintenance.

The values of specific stored potential energy for different coolants, which could be released in events of sever accidents, are summarized in Table 1.1.

1.2.2 Small modular reactor

Larger nuclear power reactors typically have lower specific costs due to the economy of scale, resulting in nuclear power plants with reactors of 1,000-1,600MWe being most commonly commercialized today. However, there is currently a growing trend in the development and commercialization of small modular reactors (SMRs). The main advantages of SMRs are that they could be suitable for areas with small electrical grids and for remote locations, and that due to the smaller capital investment for a single SMR unit the financial risks associated with their deployment would be significantly smaller than for a large reactor. This offers flexibility for incremental capacity increase which could potentially increase the attractiveness of nuclear power to investors. Also, modular concept that reduces the amount of work on-site, makes it simpler and faster to construct. Design simplicity including integral pool type enhance the economy. Small power opens passive safety features and expanded potential siting options is also advantage. As passive safety features is important in nuclear power plants to enhance safety after Fukushima accidents, many SMRs of light water coolant type and lead-bismuth coolant type adopts natural circulation in normal operation as illustrated in Figure 1.1.

1.2.3 Natural circulation technology

Natural circulation is an important mechanism in several industrial systems and the knowledge of its behavior is of interest to nuclear reactor design, operation and safety. In the nuclear technology, this is especially true for new concepts that largely exploit the gravity forces for the heat removal capability. Natural circulation in a PWR occurs due to the presence of the heat source and the heat sink constituted by the steam generator. In a gravity environment, with core located at a lower elevation than steam generator, driving forces occur that generate flow rate suitable for removing nuclear fission power.

Advantages of natural circulation of reactor coolant are improvement of safety by passive cooling characteristics and simplicity in design and miniaturization. In nuclear industry, TRIGA Mark II light water reactor having 250kW was cooled by natural convection and silent operation of nuclear submarines only was cooled natural convection of the water. Current PWRs are designed to remove decay heat by natural convection in the event of a Loss Of Flow Accident (LOFA).

1.3 Objectives

The lead-bismuth eutectic natural circulation cooled SMRs could be applicable to several needs of future energy; clean distributed energy system (smart grids) for developed country, developing nations, desalination for east Asia and Africa, and nuclear ships. It is assumed to have a small power level matching the smaller demand of towns or sites that are either off-grid or on immature local grids, being right-sized for growing economies and infrastructures of developing nations. However, four limitations to design reactor vessel and guard vessel having dimensions limited by the requirement of transportability by rail. Also, materials endurance including erosion, corrosion, and embrittlement give the reactor design limitations.

In order to maximize the economy of SMRs, power of SMRs should be maximized having modularization fabricated remotely and transported to the site. The purpose of the dissertation is to develop the power maximization method for small modular reactors with LBE natural circulation satisfying the constraints shipping size, materials endurance as well as safety under beyond DBEs.

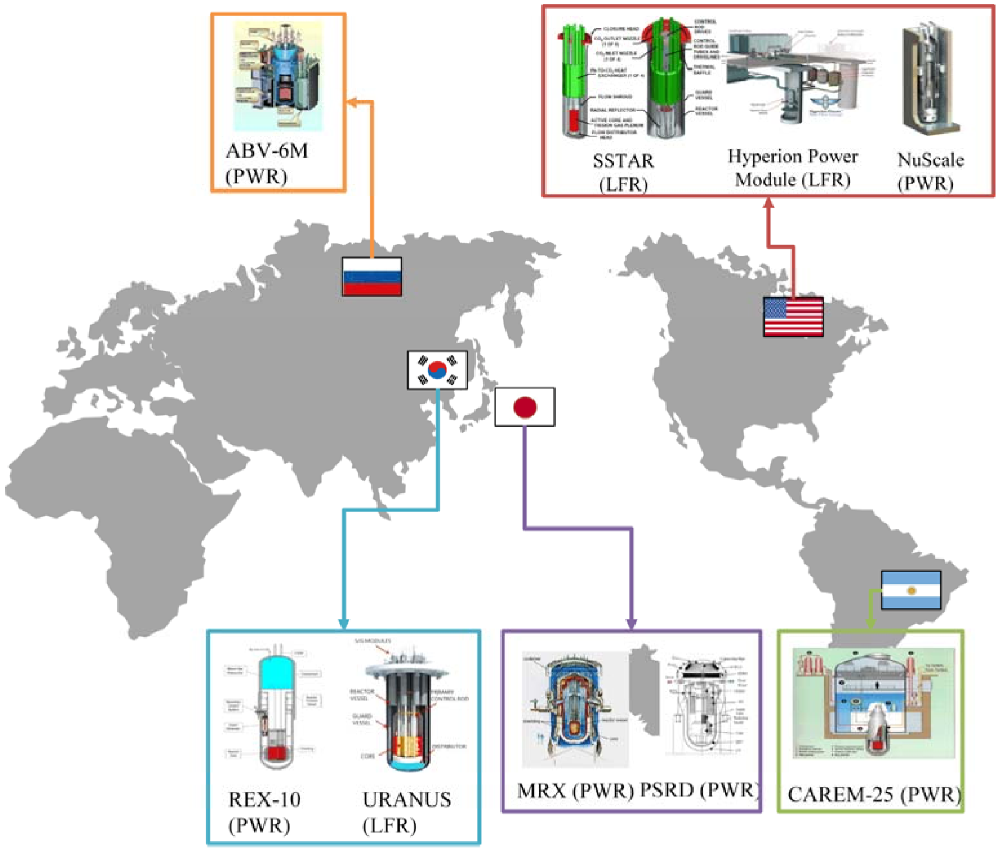


Figure 1.1 Natural circulation cooled small modular reactors

Table 1.1 Comparison of different coolants according to values of stored potential energy [4]

Coolant	Water	Sodium	Lead-Bismuth Eutectic
Parameter	P=16MPa T=300 ^{o}C	T=500^{o}C	T=500^{o}C
Maximum potential energy, GJ/m3, including:	~21.9	~10	~1.09
Thermal energy	~0.9	~0.6	~1.09
Potential chemical energy of interaction	with zirconium ~11.4	with water ~5.1 With air ~9.3	None
Potential chemical energy of interaction of hydrogen released with air	~9.6	~4.3	None
Potential energy of compression and chemical energy	~21	~9.4	None

Chapter 2 Review of the State of the Art

2.1 Fully passive cooled SMRs

Natural circulation is an important mechanism in several industrial systems. In the nuclear energy systems, natural circulation in a PWR occurs due to the buoyancy force induced from temperature gradient along the core and steam generators. Silent operation of nuclear submarines was only cooled by natural convection of the water. Current PWRs are designed to remove decay heat by natural circulation in the event of a LOFA.

There is currently a growing trend in the commercialization of the small modular reactors because SMR have many advantages in field of economy and safety. To enhance the passive safety, several designs adopt fully passive system which means coolant circulated by natural buoyancy without pump. There are two types of coolant in fully passive cooled SMRs: water and LBE. Below subsections describe the water cooled SMRs with natural circulation and LBE cooled SMRs with natural circulation.

2.1.1 Water natural circulation cooled SMRs

The water coolant has many experience in nuclear power system including many nuclear submarine which is one of the type of small modular reactors. With this reason, the realization of commercial power plant for land base SMRs with water cooled by natural circulation is easier than other coolants.

USA-NuScale, RF-ABV-6M, ROK-REX-10, Japan MRX, PSRD, Argentina-CAREM-25

2.1.2 LBE natural circulation cooled SMRs

The LBE coolant has many advantages comparing with water coolant. The specific potential energy released is twenty times less than water coolant as shown in Table 1.1. In LBE coolant system, compact design is available due to high heat transfer rate of liquid metal comparing with water coolant. Because boiling point is about 1670^{o}C of LBE coolant, pressurizer is no needed to reactor systems. There is no departure nucleate boiling (DNB) limit in LBE coolant. When hypothetical code disruption accidents occurs, melted fuel floats on LBE coolant due to high density and there no re-criticality.

SSTAR, HYPERION-USA; KOREAN-PASCAR, URANUS

2.2 LBE natural circulation experiments

There is great interest in natural circulation flow of LBE in various nuclear reactor systems but very few experimental studies have been carried out so far. Takahashi et al. (2005a) have carried out experimental and theoretical studies in LBE-water two-phase loop. The single phase natural circulation studies were carried out before LBE water two-phase boiling experiments (Takahashi et al., 2005b). Another experimental study was carried out by Ma et al. (2006, 2007) in TALL facility with forced flow and natural circulation of LBE for

ADS. The numerical results from TRAC/AAA and RELAP5 analysis were compared with the experimental results. Wu and Sienichi (2003) carried out 1D linear stability analysis for a uniform diameter rectangular natural circulation LBE loop. It was found that single phase LBE could be unstable in a high Reynolds number region and any increase in loop friction makes the forward circulation more stable. Tarantino et al. (2008) carried out steady state pre-test analysis of an LBE loop, NACIE. They performed steady state 1D analytical as well as 3D CFD studies and compared the results. There are also some analytical studies carried out for LBE cooled reactor systems. A 2-dimensional MASKA-LM computer code is developed for numerical calculations of lead coolant flows, temperatures and transport of impurities in integral system of BREST-type reactors (Kumayev et al., 2005). Heat andmass transfers in liquid metal systems aremodelled, for the coupled simulation of thermal hydraulic, physical and chemical processes in the real configuration of the reactor circuit. Abánades and Pena (2009) carried out natural circulation studies in a 2D axisymmetric geometry of an LBE cooled ADS design, using a CFD code. It was found from the analysis that the ADS design based on Lead/Bismuth eutectic natural convection cooling will operate safely, even if the gas injection mechanism to enhance fluid motion fails. Cheng et al. (2004) carried out dynamic behaviour analysis of an accelerator driven test facility using SAS-4A, a code developed by Idaho National Laboratory. Five different systems with different types of fuel and different types of coolant (LBE and sodium) were evaluated. Analysis of various transient scenarios was carried out. Davis (2003) evaluated a pool type LBE cooled reactor design that relies on forced circulation of the primary coolant, a conventional steam power conversion system and a passive decay heat removal system. The ATHENA computer code was used to simulate transients during various postulated accidents. Lee and Suh (2006) carried out natural circulation studies on lead bismuth cooled PEACER-300 and PEACER-550 (Proliferation resistant Environment-friendly Accident-tolerant Continual-energy Economical Reactor). The capabilities of the heat removal by natural circulation of LBE are estimated. A test loop, HANS (Heavy metal Alloy Natural circulation Study loop) is installed in BARC for thermal hydraulic, instrumentation development and material compatibility related studies. An experimental study is carried out on LBE natural circulation in this loop. A 1D computer code LeBENC (Lead Bismuth Eutectic Natural Circulation) is developed and validated using the experimental data.

TALL: Thermal-hydraulic ADS Lead bismuth Loop (Sweden, KTH)
NACIE: NAtural CIrculation Experiment (Italy, ENEA)
HANS: Heavy metal Alloy Natural circulation Study loop (India, BARC)
LCS: Lead Correlation Stand (USA, LANL)
HELIOS: Heavy Eutectic liquid metal Loop for Integral test of Operability and Safety of PEACER (ROK, SNU)

Figure 2.1. LBE test facility in worldwide

Chapter 3 Rationale and Approach

3.1 Problem statement

The small modular reactor with LBE natural circulation is the one of the promising energy solutions for future energy demands solving the nuclear waste, nuclear safety, and nuclear economy. However, there are no design criteria and adequate power level having natural circulation capability for LBE coolant small modular reactors. Because the electrical power level is related with economy of power plants, the finding the maximized power level is very important to obtain the indication for new nuclear power plants.

From the previous literatures, several small modular reactors with LBE natural circulation are developed and LBE loops are tested to demonstrate the natural circulation of LBE coolant in worldwide. Based on many experiments, it is verified that LBE natural circulation is capable to adopt in nuclear reactor systems. Also, there are good agreement with measured results and predictions by modeling.

The motivation of the dissertation comes from the need to obtain the realistic power level of LBE natural circulation cooled small modular reactor.

3.2 Goals

The main goal of the dissertation is to develop the power maximization method for small modular reactors with LBE natural circulation satisfying the constraints shipping size, materials endurance, long-burning criticality as well as safety under beyond DBEs. To achieve the goal of dissertation, three research questions are coming up;

1) What are limiting factors to design LBE cooled SMRs with natural circulation?
2) What are design tools to design LBE cooled SMRs with natural circulation and how are they validated?
3) How could power optimization method be developed based on findings from the previous two questions?

3.3 Approach

From the previous questions, three approaches are developed as follows; 1) design limitations, 2)design tool development and validations, and 3)power optimization method.

Design limitations are divided by limitations for steady state and accidents conditions. Steady state limitations including land-transportable shipping size limits, materials limits, neutronics limits are determined based on “10CFR 50 Appendix A General Design Criteria” of NRC. The quasi-static reactivity balance equation is used to obtain the limitations of reactivity in accidents conditions for selected Beyond Design Basis Events (BDBEs); UTOP, ULOHS. Void coefficients should be negative because steam could be penetrated into core in unprotected STGR.

For the design tool development and validations, LBE coolant experiments using HELIOS facility are conducted. In the forced convection test, pressure losses of core, orifice, gate valve, and expansion tank are obtained. In the natural circulation test, temperature distribution and mass flow rate are obtained for specific core heat. Also, long-term stability of LBE natural circulation is confirmed by 600-hours experimental test. Predictions for hydraulic-resistance and natural circulation behavior of experimental results are conducted by MARS-LBE and CFD. Pressure loss coefficients of measured data are good agreement with CFD results and natural circulation experimental results are good agreement with MARS-LBE predictions when MARS-LBE uses the recommended pressure loss coefficients from CFD simulations.

Power optimization method is derived based on developed limitations. Firstly, natural circulation SMRs design equation is defined. Then, this equation is updated with pressure loss coefficients and neutronics correlations.

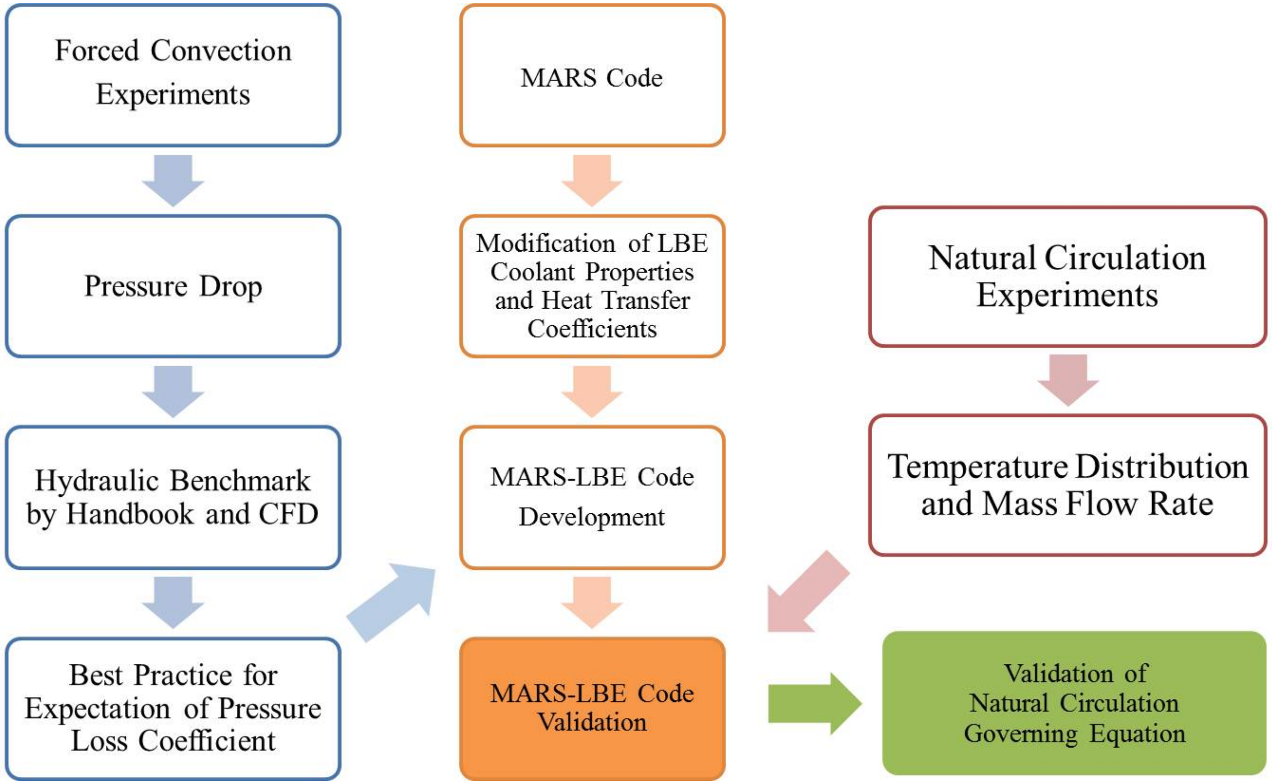


Figure 3.1. Approach for design tool development and validations

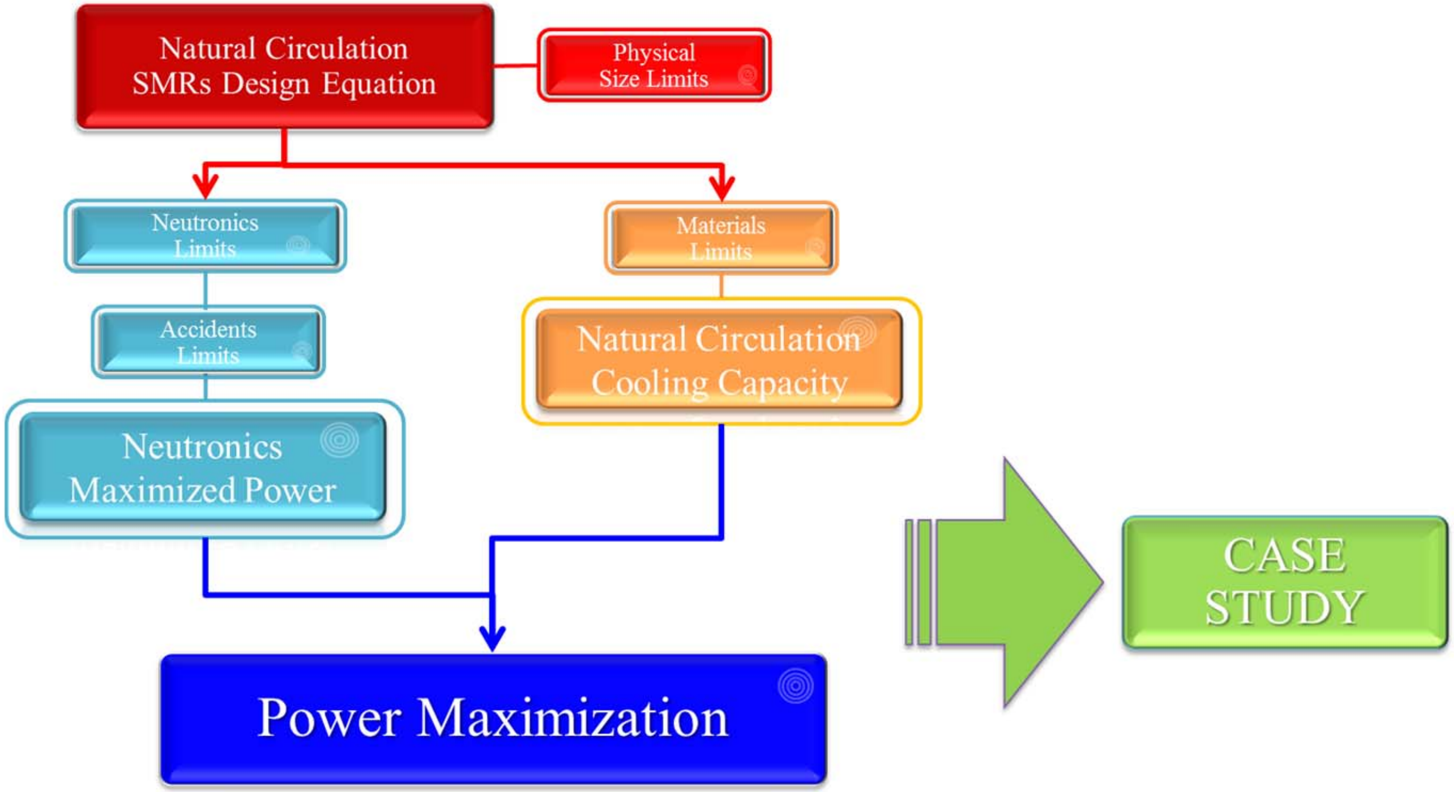


Figure 3.2. Approach for power optimization method

Chapter 4 Design Limitations

Because the main purpose of the dissertation is to develop the power maximization method for LBE cooled SMRs, the reactors derived by using power maximization method should be satisfied to many limitations related with basic concepts and its safety in accident condition as well as normal operation. Several limitations should be used for power maximization method and the others may be used for necessary conditions for designing of reactors. This chapter deals with all limitations including steady state limitations and accidents limitations.

4.1 Steady state limitations

The steady state limitations consist of three categories; shipping size limitations, materials limitations, and neutronics limitations.

Entire reactor modules that fabricated at a factory could be transferred to reactor site by land-transportation vehicle such as freight train, heavy-duty tractor in order to enhance the flexibility of construction anywhere. The limitations is that guard vessel module which is largest component among the all reactor modules is smaller than train transportable dimension limits. Also weight of guard vessel is less than train transportable weigh limit. The core of target SMRs is fabricated as casualization due to non-proliferation, and easy refueling. Thus, dimension of core also limited to Type B fuel casks dimension limits. [Reference]

Materials limitations are related with physical properties, corrosion and erosion behaviors of LBE coolant with structure materials. One of the candidate structural materials is the martensitic steel T91, that is a readily available industrial materials, which was qualified as steam generator material for non-nuclear and nuclear power plants. The modified 9Cr-1Mo steel T91 has higher strength, low thermal stress and lower ductile-brittle transition temperature (DBTT) shift after irradiation. Also the T91 has good swelling behavior for high burn-up condition [5].

Because melting point of LBE coolant is 123.5^{o}C [6], all local coolant temperature is maintained above the melting point. Ductile to brittle transition temperature (DBTT) of materials rise with irradiations. At high burn-up condition, DBTT of irradiated materials is about 200^{o}C. Thus, coolant temperature should be above the point. The corrosion

The lead-alloy coolant velocities are limited by erosion concerns of protective oxide layers to about 2.5m/s [7]. Based on this erosion behavior, local velocity of LBE coolant is limited to the velocity. The melting point of UO2 is usually assumed to decrease with increasing burn-up. The temperature limitations of cladding peak temperature and fuel peak temperature are 1500^{o}C and 2800^{o}C, respectively.

The criticality of reactor core should be maintained during burning cycle. Because the main concept of the SMRs with LBE natural circulation is long-burning cycle to solve the nuclear waste problem, core design with sensitively tuning is needed for long burning criticality. With this criticality, burnup reactivity swing and maximum excess reactivity swing should be limited to 1$ and 2$, respectively. The rationale of the limitations of reactivity swing and excess reactivity are that enhancing passive safety and obtaining guarantee of shutdown capability.

Table 4.1 Steady state limitations

Category	Rationale	Design Variables	Requirement
Shipping Size Limitations	Train transportable dimension limit	Guard Vessel Diameter	<4.6m
	Train transportable dimension limit	Guard Vessel Height	<18.9m
	Train transportable weight limit	Guard Vessel Weight	<143tone
	Type B fuel casks dimension limit	Reactor Core Diameter	<2.0m
	Type B fuel casks dimension limit	Reactor Core Height	<5.6m
Materials Limitations	Coolant melting	Coolant Temperature	$> 123.5^{o}C$
	No ductile to brittle transition	Coolant Temperature	$> 200^{o}C$
	No cladding corrosion	Coolant Temperature	$< 470^{o}C$
	No cladding oxide film erosion	Coolant Velocity	< 2.5m/s
Neutronics Limitations	Criticality for burning cycle	Effective Multiplication Factor	>1
	Enhancing passive safety due to small investment of reactivity in control rods	Burnup Reactivity Swing	<1$
	Guarantee of shutdown capability during whole reactor operation time	Maximum Excess Reactivity	<2$
	No cladding embrittlement	Displacement Per Atom (DPA)	<200dpa

4.2 Accidents limitations

4.2.1 Design Basis Events

Based on NRC's Severe Accident Policy Statement that are applied to liquid metal fast reactor such as PRISM and KALIMER, beyond design basis events are listed; Unprotected Transient OverPower (UTOP), Unprotected Loss Of Flow (ULOF), Unprotected Loss Of Heat Sink (ULOHS), Sub-Assembly Blockage, and Steam Generator Tube Rupture (SGTR).

Because there is no coolant pump normal ULOF accident is impossible. However, change of flow rate induced from local blockage may possible. Sub assembly blockage accidents in PWRs may lead melting of blocked fuel assembly, in contrast of it, target SMRs have monolithic core without assemblies, thus there are cross flow in whole core region. There is no impossible to melt sub assembly down in blockage accidents.

In safety behaviors, three uncontrolled transient scenarios were considered. The first is UTOP caused by the withdrawal of the most efficient control assembly. External reactivity insertion by control rod withdrawal without scram. Inherent reactivity feedback mechanisms may limit power increase. The second is ULOHS initiated by loss of flow at the secondary cooling system. Loss of heat removal by malfunction of the SG feed water pump without scram. Inherent reactivity feedback mechanisms make power decrease to decay power level. The third is SGTR by SG tubes rupture and steams penetrate into low pressure primary side.

In the dissertation, the designed reactor by power maximization method have passive safety shutdown in accidents conditions without safety analysis. The safety criteria is derived from scenario of each accident. A UTOP accident may be caused by a positive reactivity perturbation in the core. In this analysis, the positive reactivity is assumed to be initiated by an inadvertent ejection of the most efficient control assembly without scram during the full power normal operation. In contrast to thermal reactors controlling a group of control rods, fast reactors manipulate a group of control assemblies. Right after the insertion of positive reactivity the core power rapidly increases, leading an immediate temperature rise. Increasing temperature brings a prompt negative reactivity feedback through the Doppler effect and core deformation effect of reactivity. Then, power is decreased to new steady state condition. A ULOHS may be initiated by significant reduction or elimination of heat removal capability in the secondary system. All heat removal capability of steam generators are presumed to disappear by feed-water pump failures. Water flow to the steam generators exponentially decreases for a few seconds after no active pump motion due to inertia momentum. Unlike the UTOP scenario, the ULOHS analysis assumes that RVACS system is automatically activated at the pre-set temperature enough to melt fusible diaphragm blocking air-flow in the normal operation. At the initial stage, both inlet and outlet coolant temperatures rapidly rise, which results in significant negative reactivity feedback, quickly dropping the total reactor power. Increasing coolant temperature gives the negative reactivity coefficients by coolant density effect and core radial expansion effect. In these two accidents, inherent reactivity feedback could be safely shutdown of reactor systems.

In a SGTR, steams penetrate into LBE primary side because secondary side pressure is 80bar and secondary ~1 bar. Large amount of bubbles rise to cover gas of reactor head. Then cover gas pressure is increased by up to the pressure that safety valve operation limit. These bubbles are counter current flow with flow direction of LBE coolant, then, increased resistance to natural circulation flow. Mass flow rate is decreased and coolant temperature is increased and negative coolant density effect of reactivity is inserted. Small amount of bubbles enter the core. Void fraction in the core is increased and void coefficients is inserted in the core. Thus, if void coefficient is negative, reactor could be safely shutdown in STGR accident.

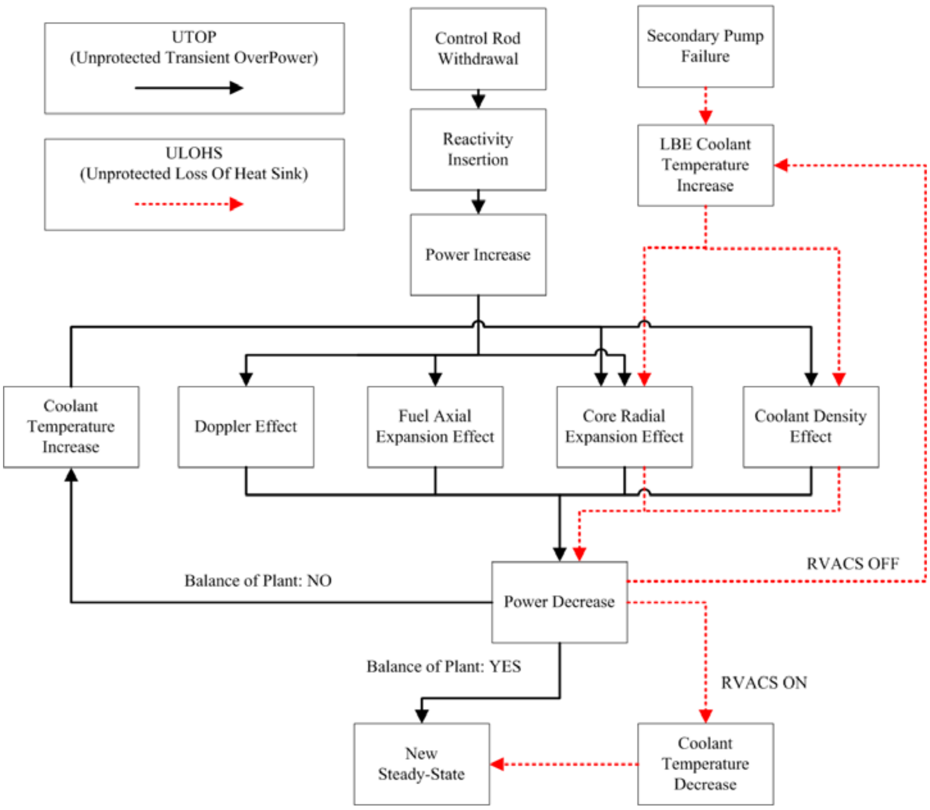


Figure 4.1 Inherent reactivity feedback for Unprotected Transient OverPower (UTOP) and Unprotected Loss Of Heat Sink (ULOHS)

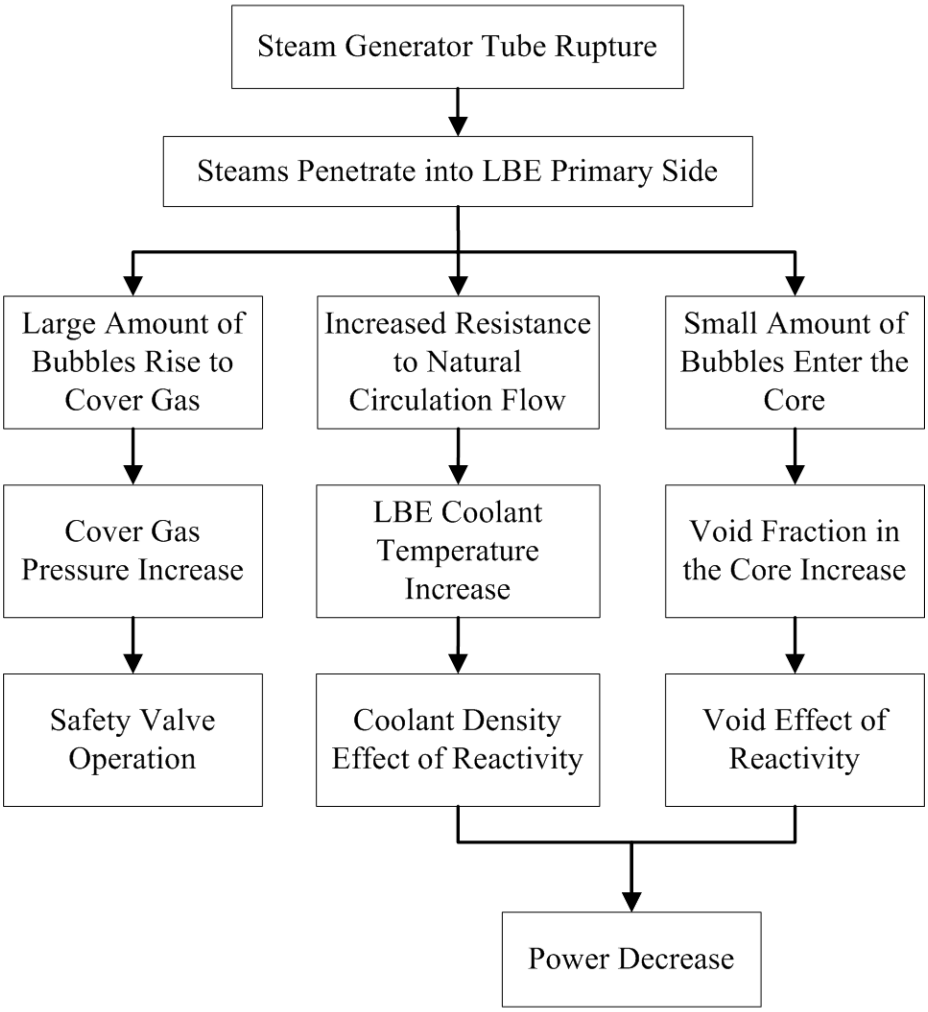


Figure 4.2. Scenario of SGTR (Steam Generator Tube Rupture) for LBE coolant systems

4.2.2 Modified reactivity requirement for safe passive shutdown

An reactor core can be influenced by external events only through changes in the coolant inlet temperature and flow rate or through externally core geometry changes. Of these three communication paths, the BOP can influence the core only through coolant inlet temperature. These three all-encompassing paths by which external changes can influence the reactor, are embodied in the two generic events; ULOHS and UTOP.

Given the limited ways the core can be influenced by external events, it is useful to write a quasi-static reactivity balance as:

$$0 = \Delta\rho = (P-1)A + (P/F-1)B + \delta T_{in}C + \Delta\rho_{ext}$$

Where P and F are normalized power and flow, δT_{in} is the change from normal coolant inlet temperature, and $\Delta\rho_{ext}$ is the externally-imposed reactivity. A is the power coefficient of reactivity that means change of reactivity per 1% increase of normalized power. B is the power divided by flow coefficient of reactivity that means change of reactivity per 1% increase of normalized power divided by normalized flow rate. C is the inlet temperature coefficient of reactivity that means change of reactivity per 1^{o}C increase of inlet coolant temperature.

When the accidents occur in reactors, reactivity is converged to the value that is satisfy the quasi-static state equation. For example, if power is increased, term ‘P-1’ is changed to positive and term ‘(P-1)A’ is changed to negative. In order to compensate this term, change from normal coolant inlet temperature should be negative to get the positive of ‘$\delta T_{in}C$’.

Chapter 5 Design Tool Development and Validations

5.1 LBE coolant T-H test facility: HELIOS

In 2005, LBE integral test loop, named as HELIOS (Heavy Eutectic liquid metal Loop for Integral test of Operability and Safety of PEACER) was constructed in Seoul National University. The main design criterion of the HELIOS is to obtain the similarity of natural circulation capability between HELIOS and its prototype, PEACER-300, which is lead bismuth eutectic cooled transmutation reactor. In order to obtain the similarity, HELIOS was designed to have same height and same total pressure loss coefficient with PEACER.

The height of activated HELIOS loop is 12.0 meters and the difference of each thermal center point is a 7.4 meters. It consists of primary side with LBE and secondary side with single phase oil. The main LBE loop that is benchmarked for comprises a core, an expansion tank, a heat exchanger, a mechanical pump, orifice, gate valve, tee junction, 45 elbow, 90 elbow, and straight two inch pipes.

Four electrically heated rods in the core are the principle heat source, with its 60 kW maximum power. On the other hand, the heat exchanger on the top of the HELIOS is the heat sink region. The LBE flows downward in the shell side of the heat exchanger while the secondary fluid flows through upward on the tube side. The functions of the expansion tank are to accommodate the level change and to control the oxygen activity in LBE. The centrifugal sump type mechanical pump has about 45 ft of LBE head and 20 US gpm of flow rate.

Figure 5.2 shows four sets of differential pressure meters and ten thermo-couples in several regions of HELIOS. Differential pressure meters were installed in core region, orifice region, gate valve region, and combined region in order to measure pressure loss of each region.

There are two types of critical instrumentations for the forced convection test: thermocouples and differential pressure transducers. Thermocouples of Type K with stainless steel 304 used to monitor the fluid and wall temperature, are installed in numerous locations in the HELIOS. All the thermocouples were calibrated to an accuracy of ±0.5K. The pressure loss in the primary side of the HELIOS occurs mainly in the core, the orifice, and gate valves were expected by pre-calculation. The core has the largest pressure loss due to its three spacers with quite small flow areas. The each spacer takes about 74% of the flow area in the core with 49.5 mm inner diameter. An orifice and a gate valve also have comparatively large pressure losses associated sudden expansions and sudden contractions. In order to measure pressure losses in the important components, differential pressure transducers are set up to cover the core region, the gate valve region and the combined region. Differential pressure transducers are manufactured by Rosemount Company with Model Rosemount-3051 CD3A.

Figure 5.1. HELIOS facility (picture)

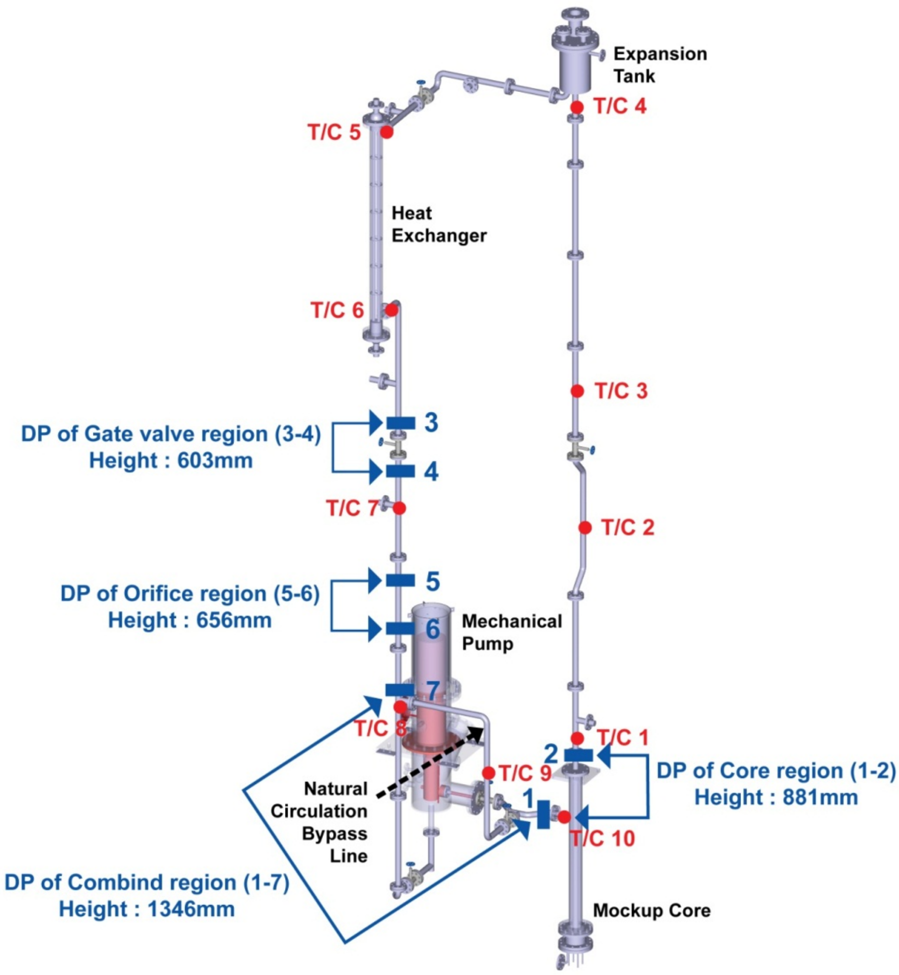


Figure 5.2. HELIOS facility

5.2 Lead-Alloy Cooled Advanced Nuclear Energy Systems (LACANES) benchmarking

5.2.1 Introduction to LACANES benchmarking

In 2007, the OECD Nuclear Energy Agency has published a comprehensive handbook on lead-bismuth eutectic alloy and lead properties, materials compatibility, thermal-hydraulics and technologies [1] to integrate available information on such heavy liquid metals (HLM). Meanwhile, a systematic study on HLM was proposed which covers thermal-hydraulic safety issues of lead alloy-cooled advanced nuclear energy systems (LACANES). This study mainly characterizes thermal-hydraulic behaviors of those LACANES under the steady-state forced and natural convection, which is of critical importance for the system design development effort, while such studies have been extensively carried out for sodium coolants.

By utilizing large-scale lead-alloy coolant loop test facilities, experimental data can be examined and qualified for used in benchmarking of these models. Hence, the reference of benchmark is large-scale lead-bismuth (Pb-Bi) coolant loop test facility HELIOS of the Seoul National University in the Republic of Korea.

According to the HELIOS test results, two phases of approach are suggested:

- Phase I - Isothermal steady-state forced convection case
- Phase II - Non-isothermal natural circulation case

Prior to the Phase I, a comparative study on the pressure loss coefficient of each part of HELIOS under isothermal conditions are performed as well. All thermo-physical properties of LBE coolant are based on the OECD/NEA LBE handbook.

The complete list of participants and codes used are shown in Table 5.1. Figure 5.3 shows schematic diagram of entire benchmark plan. Based on the specification [1], participants performed preliminary analysis using well known correlations and performed the blind computer simulation by thermal hydraulics [8] system codes and three dimensional computational fluid dynamics (3D CFD) codes. The CFD result was used as the reference where experimental data is not available . Then the results are compared components by components. Finally, the optimized correlations and recommendations on the pressure loss prediction method will be suggested as the “best practice guidelines”.

5.2.2 Pressure loss coefficients as the best practice guidelines

The measured pressure losses in the core region, the orifice region, the gate valve region, the expansion tank region are shown in Figure 5.4. In a forced convection test, the pump speed was increased gradually and the pressure losses were measured as a function of mass flow rate at 250^{o}C isothermal conditions. predictions by internal correlations embedded in thermal hydraulic system codes, specific handbook correlations, and CFD simulation were compared with measured data. Based on the comparison, the recommendations for prediction method of pressure loss were obtained. comparison results for three components of a core, an orifice, and a gate valve will be explained. In all the components, prediction by correlations will be compared with measured data. Especially, CFD simulation of the core region will be described. After that, accumulated pressure losses of all participants and calibrated results tuned on measured data will be summarized.

The core was designed based on the scaling analysis parameters determined from the conceptual design for the PEACER-300 core. Therefore, the HELIOS core has a downcomer, a circular flow channel with circular heating rod, and the spacers to simulate the prototype as closely as possible. The outer vessel of the core has been made of a 127 mm diameter pipe, including a 60.5 mm diameter pipe as the downcomer, and the core barrel. Inside the core barrel pipe, four main heating rods have been employed with three spacers. A bottom vessel of the mockup core was designed to provide a drain line from the hotleg side and a maintenance space for instrumentation. Because the main heating rods are fixed at the flange by the commercial 12.7 mm Swagelok fittings welded on the flange bottom, some curved shape was needed to increase the radial space for the welding of fittings. This is the reason why the four main heaters do not have straight shape but some curvatures in the non-heater section. Thus, LBE flows from the horizontal core inlet down to the bottom vessel through the downcomer outer wall, and rises back upward through the downcomer inner wall and goes out to a vertical core outlet, just as it does in the real plant. The core region has the most complicated geometry among all regions [10].

In the predictions by the participants, pressure loss in the spacers is the largest in the core region (50-74 % of the total pressure loss), and it varies from 14 to 38 kPa. On the other hand, friction loss from rod bundles and remainders are quite similar. Therefore, further verification of the prediction method of a pressure loss in the spacers is needed.

Most participants have tried to calculate the pressure loss on spacers by using the Rheme correlation as given Eq. (2) below.

(2)

The drag coefficient (Cv) is a function of the Reynolds number in average bundle [10]. Rheme indicated Cv were scattered based on experimental results, have not specific value. Thus, several correlations of Cv were constructed based on several experimental results. As a result, the results of those participants who used the Rheme correlation have large variations. Used Cv are summarized in Table 2. All results calculated by Rheme correlations underestimated measured data with 25 % - 48 % difference.

One participant tried to predict pressure loss in the spacers using the empirical formula of orifice geometry that is thick-edged orifice in a straight tube. This empirical orifice correlation is defined by Eq. (3), (4), (5), and (6).

(3)

(4)

(5)

(6)

The predicted pressure loss based on the empirical orifice correlation is larger than other predictions obtained using Rheme correlation. It is closer to the measured value compared to the case when the Rheme correlation is used [13-16].

There are two evaluations of the pressure loss in the core region obtained by two participants with the use of the two CFD codes, StarCD and CFX.

Both evaluations used the same k-epsilon model but with different meshes. In the StarCD and the CFX simulations, a polyhedral mesh with 410,000 polyhedral and 2,620,000 tetrahedral meshes were used, respectively. Figure 8 illustrates the computational domain showing mesh in the lower plenum and the lower part of the core rods, including the lower spacer.

Figure 9 shows the CFD results for pressure distribution at a plane cross centerline of the core region. There is a sharp pressure drop across the three spacers. These pressure loss diagrams clearly indicate that the highest portion of a pressure loss in the core region corresponds to the spacer region. The pressure loss calculated by the StarCD code has a slightly higher value than that calculated with the CFX code. The difference in the results obtained by the two codes could be attributed to a difference in the meshes used. Figure 10 shows the y+ values of the CFD simulation. It indicates the StarCD simulation has a higher quality surface mesh than the CFX simulation [17-19].

Figure 11 shows the predicted pressure losses for the core region based on handbook correlations and CFD evaluations, compared to the measured data for a high mass flow rate condition. Among the results, a prediction based on the empirical orifice correlation for spacers by the participant E, and the result obtained using the StarCD code are highly accurate evaluations.

In fact, concerning the two prediction methods based on handbook correlation, the Rheme correlation for spacers produced no good results compared to the measured values, as presents about 40% variation between the predictions from the participants and the measured data. Instead the orifice correlation for spacers has the best agreement with the measured value. It appears that the key physical phenomena responsible for pressure loss in a spacer and in an orifice are those related to sudden expansion and sudden contraction. However, orifice empirical correlations cannot be recommended directly to calculate the pressure loss of spacers in other fluid system since this correlation was constructed for orifice.

Thus, new correlation of Cv was recommended as shown in Eq. (7), which was obtained by experimental measured data of HELIOS.

(7)

All used Cv and new one which was modified by experimental data of HELIOS are shown in Figure 12.

On the other hand, the two CFD results have shown good agreement with the measured values yielding ~1 % and ~11 % variation, respectively. Some parametric studies with different mesh size, turbulence model, wall treatment, etc should be needed to obtain the suitable modeling options for LACANES. It indicates CFD predictions with suitable modeling options provide more reliable results than those based on a handbook correlation [17].

Figure 13 shows the two-dimensional drawing of the HELIOS orifice region. Orifice body has a sudden contraction of a small hole, from 52.9 mm down to 32.46 mm diameter. Then, LBE spreads out along a 45o expansion tube [10].

Figure 14 shows the predicted pressure losses in the orifice region based on handbook correlations and the corresponding measured data for a high mass flow rate condition. The predicted results by participants A, B, C, E, F, and I are close to the measured data. They all used empirical orifice correlations from the Idelchik handbook [16, 20-21]. However, the values have small variations because they employed different correlations of different version of Idelchik handbooks. Also, in the Idelchik handbook, several options to calculate pressure drop of orifice are introduced. Nevertheless, pressure drops calculated by these correlations have shown good agreements with the measured data within 10% difference.

One of the empirical orifice correlations in the Idelchik handbook is expressed by Eq. (8) which is a recommended correlation for the orifice region.

(8)

Figure 15 shows a two-dimensional drawing of the HELIOS gate valve. There are five gate valves along the forced convection test line. They make the flow path for a thermal hydraulic test or a material test. In the gate valve, LBE undergoes a sudden contraction and a sudden expansion on the valve plugging region. Gate valves are designed to be fully opened in the normal operation condition. A typical gate valve has no obstruction in the flow path. There is a sudden contraction and a sudden expansion from 52 mm to 37 mm and from 37 mm to 52 mm diameter, respectively [10].

Figure 16 shows the predicted pressure losses versus the measured data in the gate valve region for a high mass flow rate condition. Two methods to predict the pressure loss in the gate valve region are introduced below.

First of all, the participants, A, B, C, and D used the Borda-Carnot correlation for sudden expansion of cross-sectional area of the gate valve. However, there are variations between the results because the participants used different correlations for the predictions of a sudden contraction. The correlations used by participants to predict the pressure loss of sudden area contraction are summarized in Table 2 and plotted in Figure 17 [22-25]. These coefficients obtained based on experimental results at large Reynolds number ($>10^4$) and fully developed flow condition.

Secondly, the participant E used CFD code to predict a pressure loss in the gate valve.

A comparison between the results obtained using the handbook correlations and the measured data shows that all correlations based on handbooks tend to overestimate the results. On the other hand, the CFD code prediction has shown good agreement with the measured data.

Table 5.1. List of participants and code for the OECD/NEA benchmark on LACANES

Country	Institute	Participant	Code*
Italy	ENEA	Paride MELONI and Francesco Saverio NITTI	RELAP5-Version HLM
Italy	ERSE	Vincenzo CASAMASSIMA	LEGO
Russian Federation	GIDROPRESS	Alexander V. DEDUL	TRIANA
	IAEA	Vladimir V. KUZNETSOV	
Russian Federation	IPPE	Oleg KOMLEV	HYDRA
Germany	KIT/IKET	Abdalla BATTA, Xu CHENG, and Andreas CLASS	HETRAF, STAR-CD®
Germany	KIT/INR	Wadin JEAGER	TRACE
Russian Federation	RRC KI	Alexey SEDOV	
Republic of Korea	Seoul National University	Il Soon HWANG and Jae Hyun CHO	MARS-LBE, CFX®

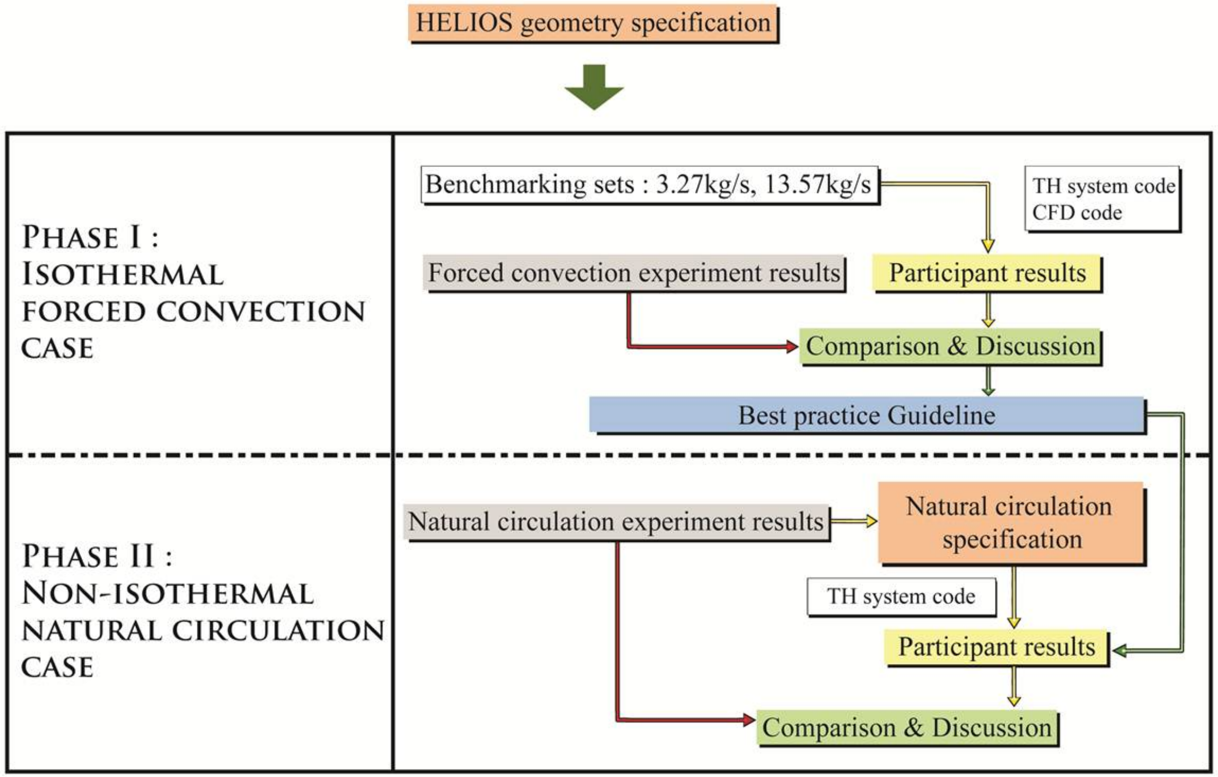


Figure 5.3. Overall procedures of LACANES benchmark

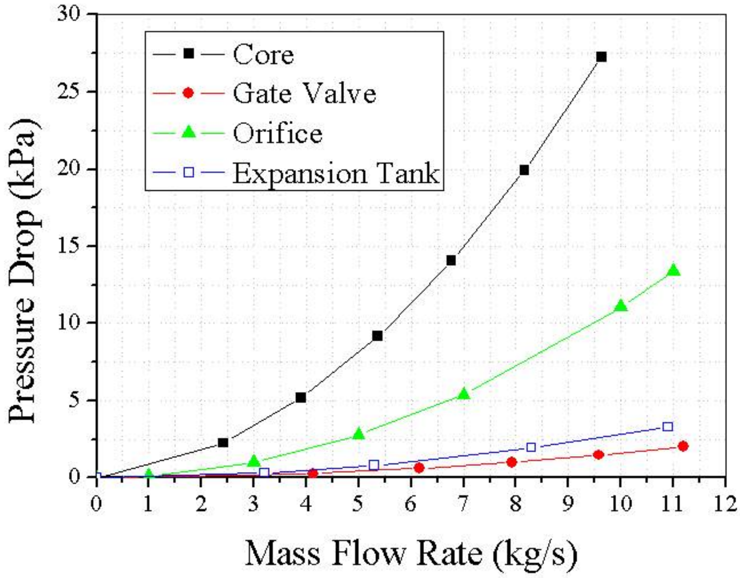


Figure 5.4 Experimental results for pressure drop in the HELIOS

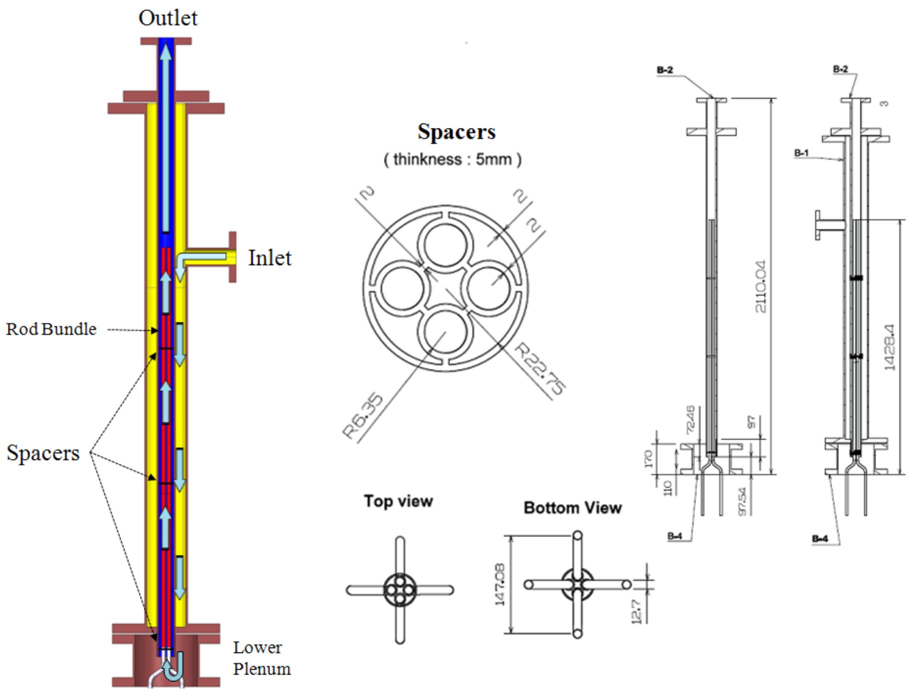
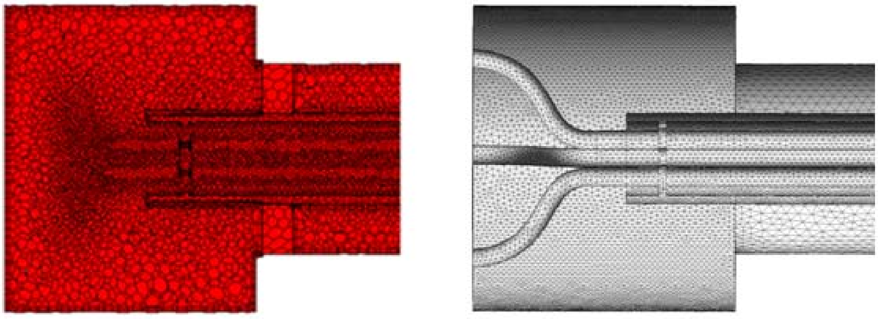


Figure 5.5 Two-dimensional drawings of HELIOS core region



(a) (b)

Figure 5.6 Cross section in the computational domain showing mesh in the lower plenum and the lower part of the core rods including the lower grid, (a) Star-CD, (b) CFX

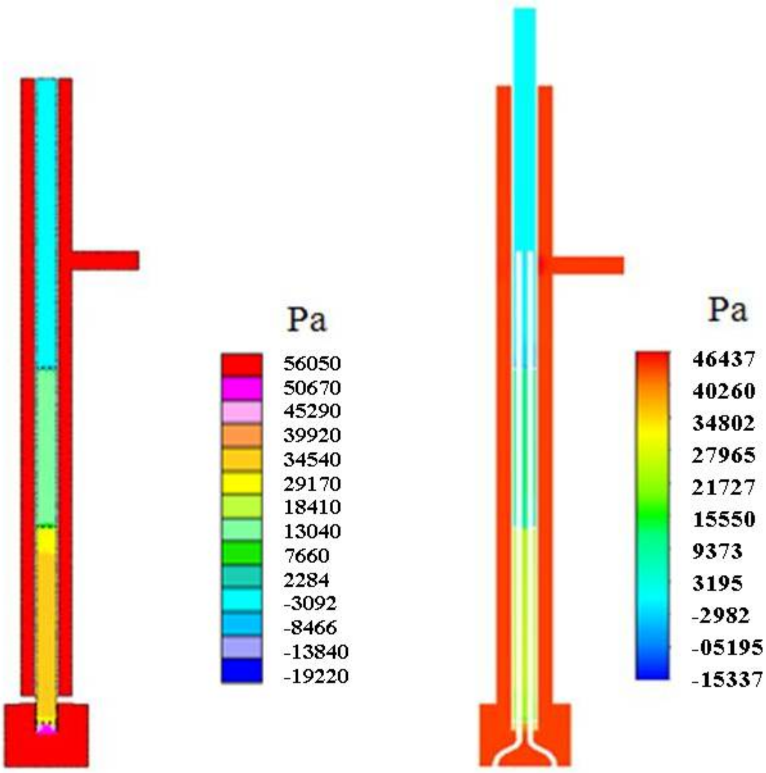


Figure 5.7 Pressure counters results at a plane cross the center line of the core (a)Star-CD (b)CFX

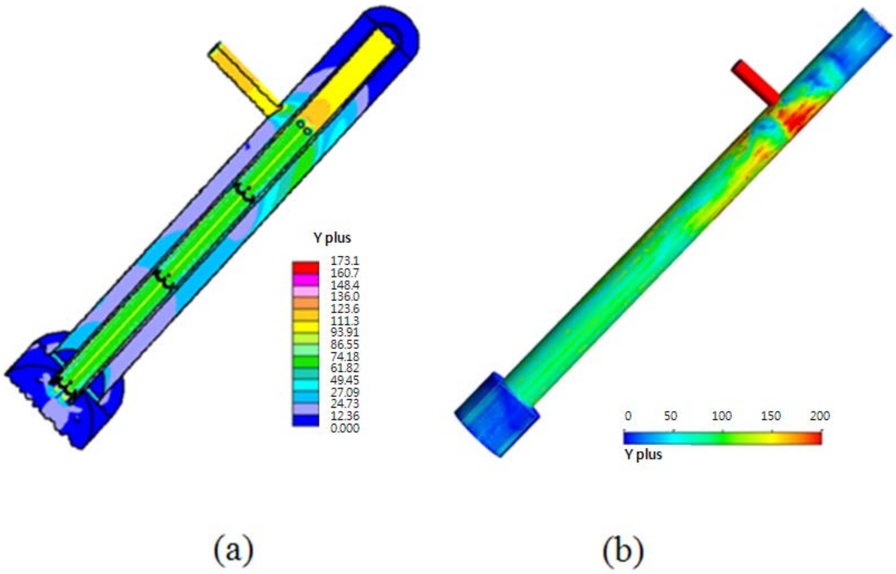


Figure 5.8 Computational domain and resulting y+ values (a) Star-CD, (b) CFX

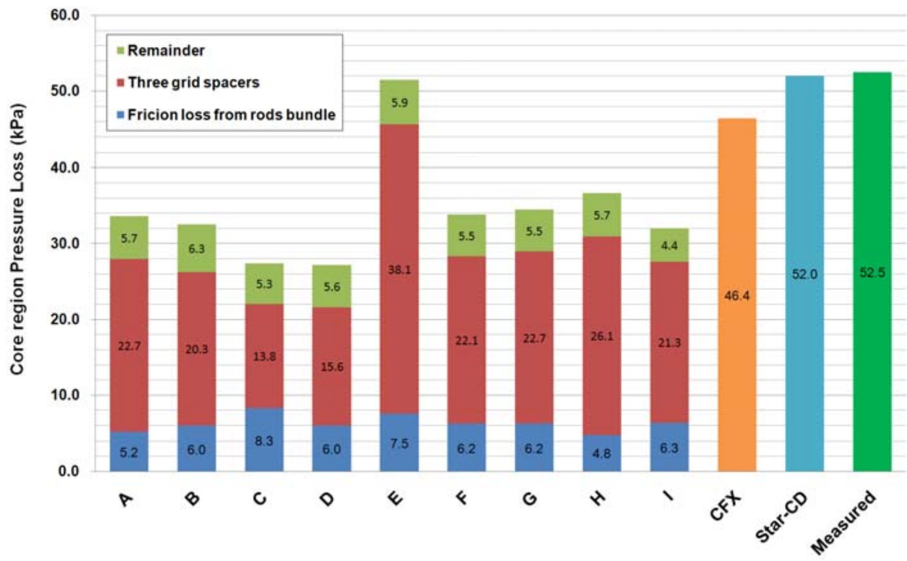


Figure 5.9. Calculated pressure loss from handbook correlation by nine participants, CFD estimation by two participants and measured data of HELIOS core region under high mass flow rate condition (13.57kg/s)

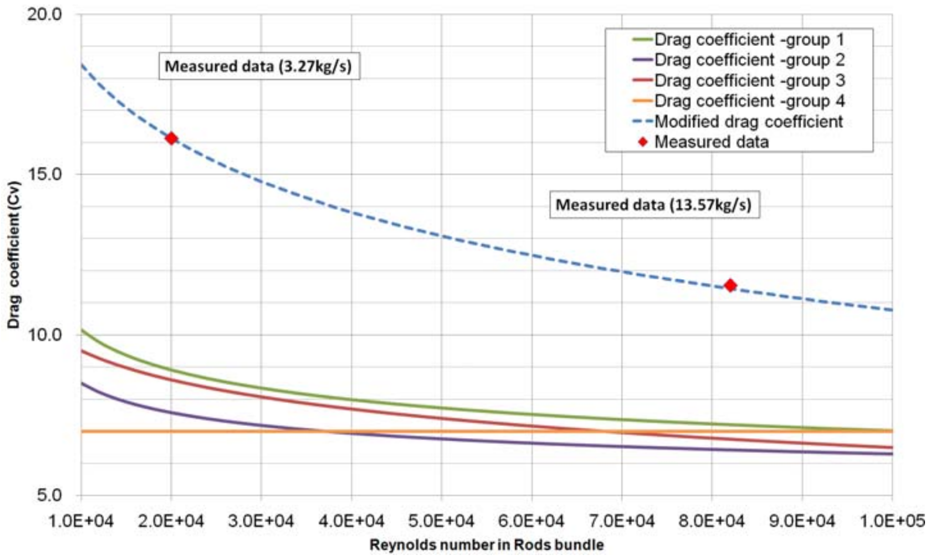


Figure 5.10. Drag coefficient (C_v) of Rheme correlation for predicted pressure loss of grid spacers; modified new one based on measured data, and four set used in benchmarking

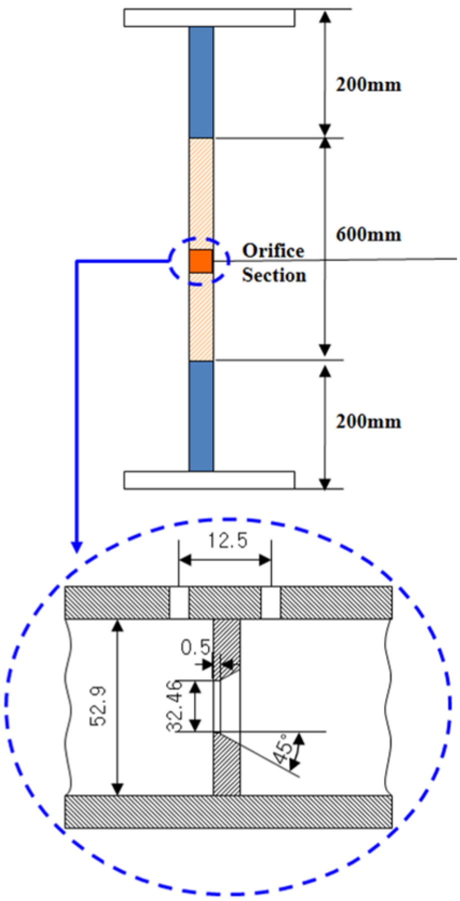


Figure 5.11. Two-dimensional drawing of HELIOS orifice region

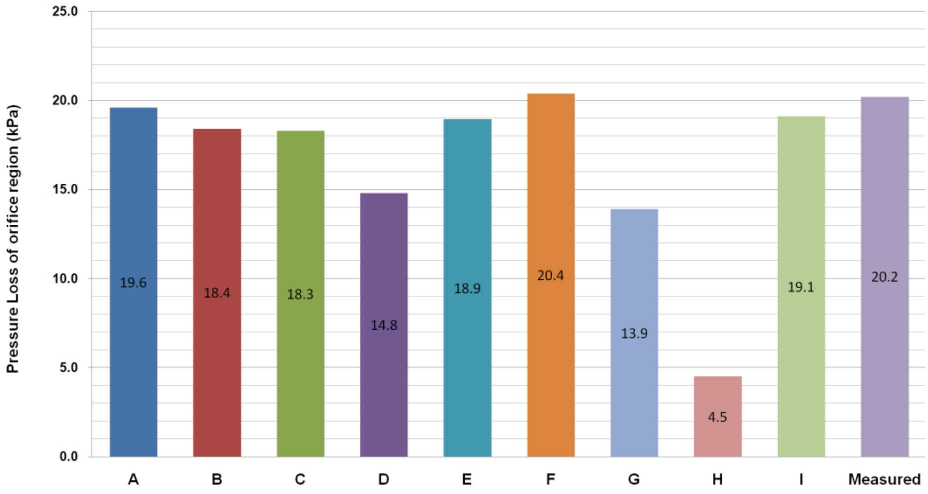


Figure 5.12. Calculated pressure loss of HELIOS orifice from handbook by eight participants and measured data of HELIOS orifice region in high mass flow rate condition (13.57kg/s)

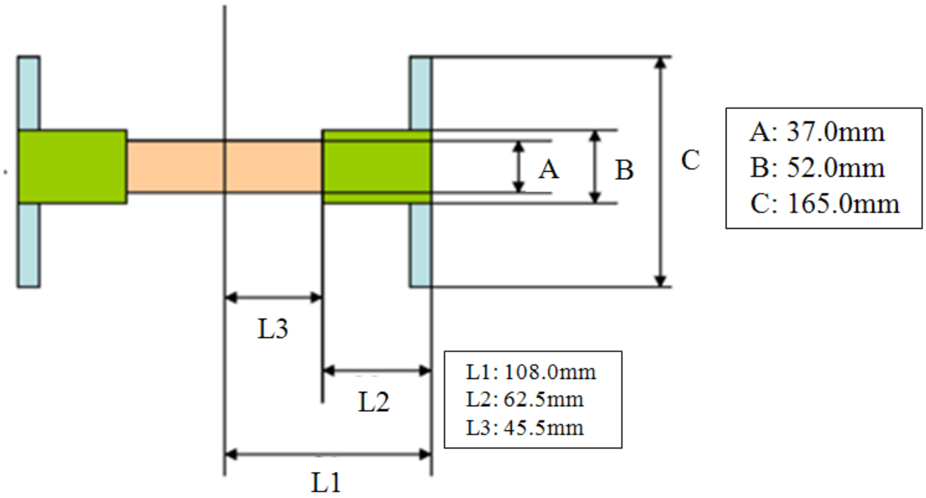


Figure 5.13. Two-dimensional drawing of HELIOS gate valve

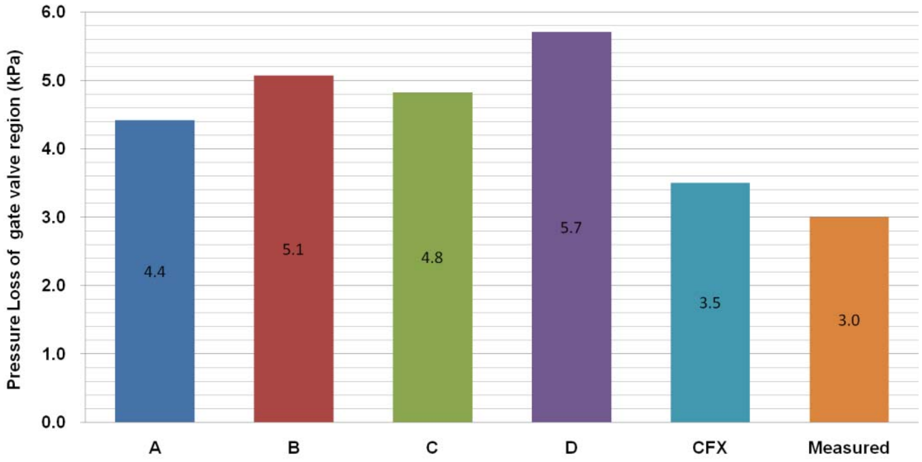


Figure 5.14. Calculated pressure loss of from handbook by eight participants and measured data of HELIOS gate valve (1EA) in high mass flow rate condition (13.57kg/s)

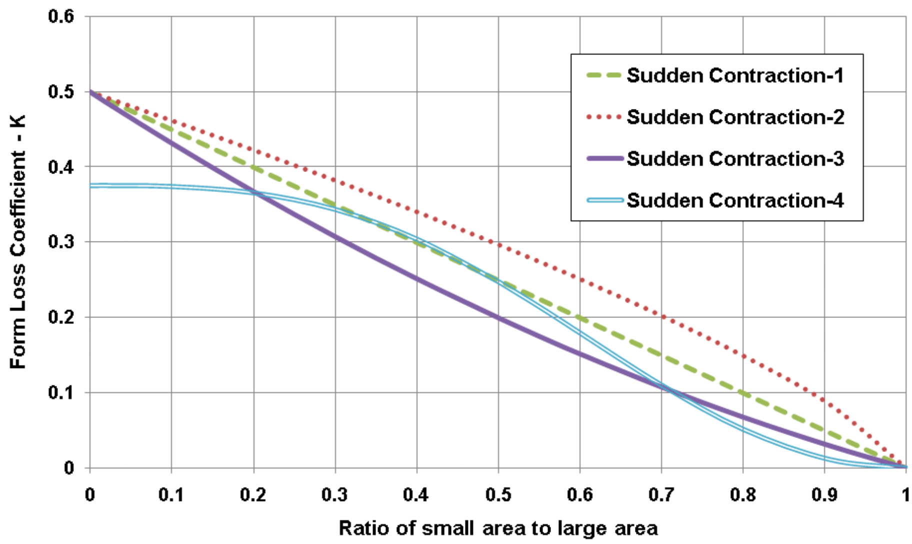


Figure 5.15. Form loss coefficient for predicted pressure loss of sudden area contraction; four set used in benchmarking

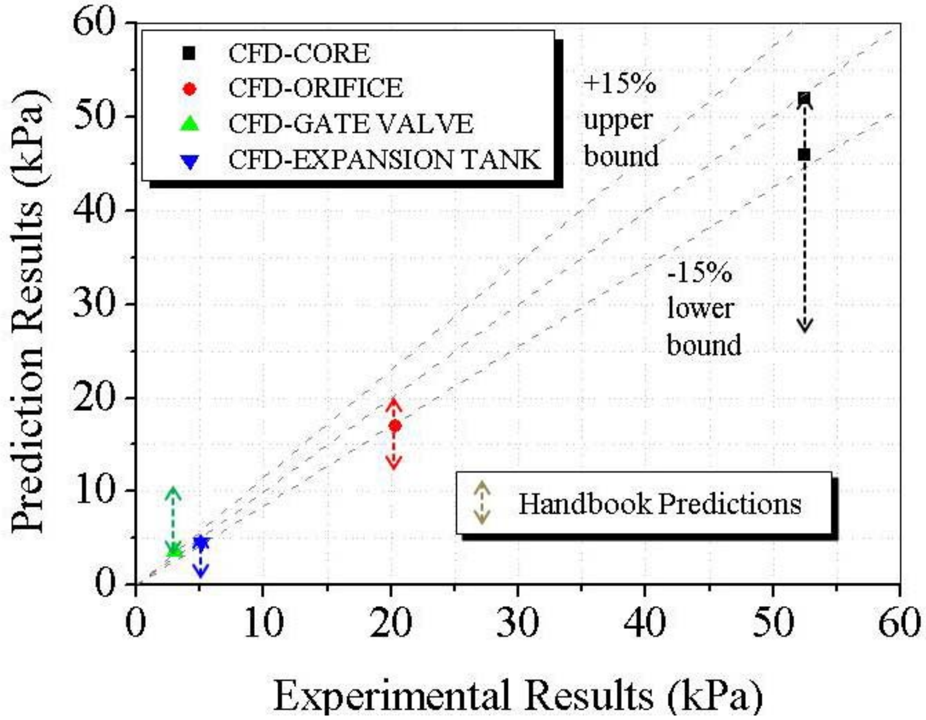


Figure 5.16. Pressure drop measurements with CFD predictions and Handbook predictions

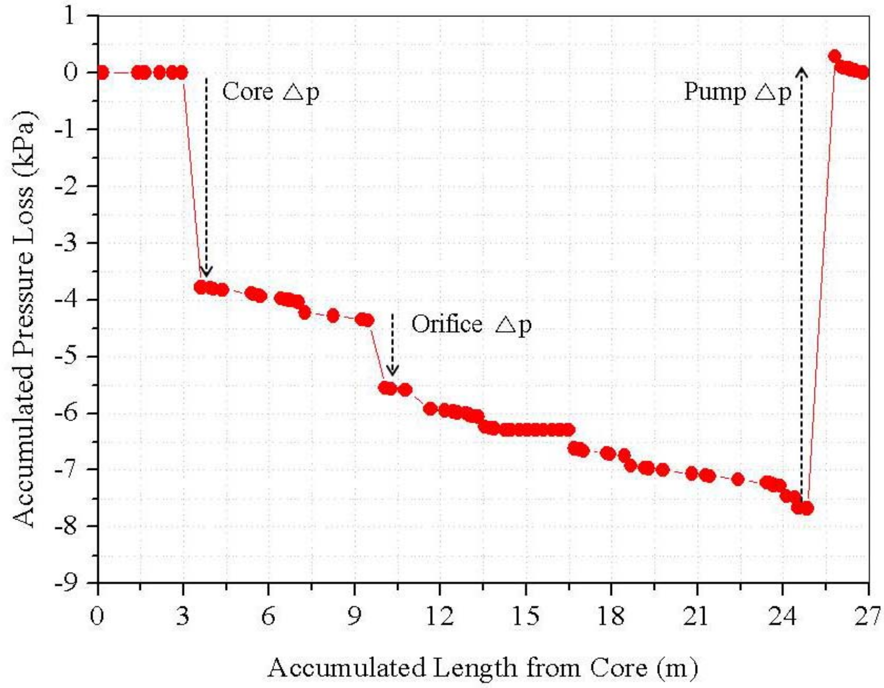


Figure 5.17. Recommended pressure losses in all components of the HELIOS

5.3 Predictions for natural circulation and validations

5.3.1 MARS-LBE simulations for natural circulation

MARS code was developed by KAERI for PWR system T-H analysis. It is integrated code of RELAP5 code and COBRA-TF code that is one dimensional thermal-hydraulic code and 3 dimensional core analysis code, respectively. In order to simulate LBE coolant system, the code was tuned by LBE properties. Density, expansion coefficient, and specific heat are changed in property table of code. Thermal conductivity, viscosity are modified from sourced code. Nusselt number of LBE coolant was selectively inserted in heat transfer module of MARS code. Instead of Dittus-Boelter correlation for PWR, Notter and Sleicher correlation for low Prandtl number coolant is adopted.

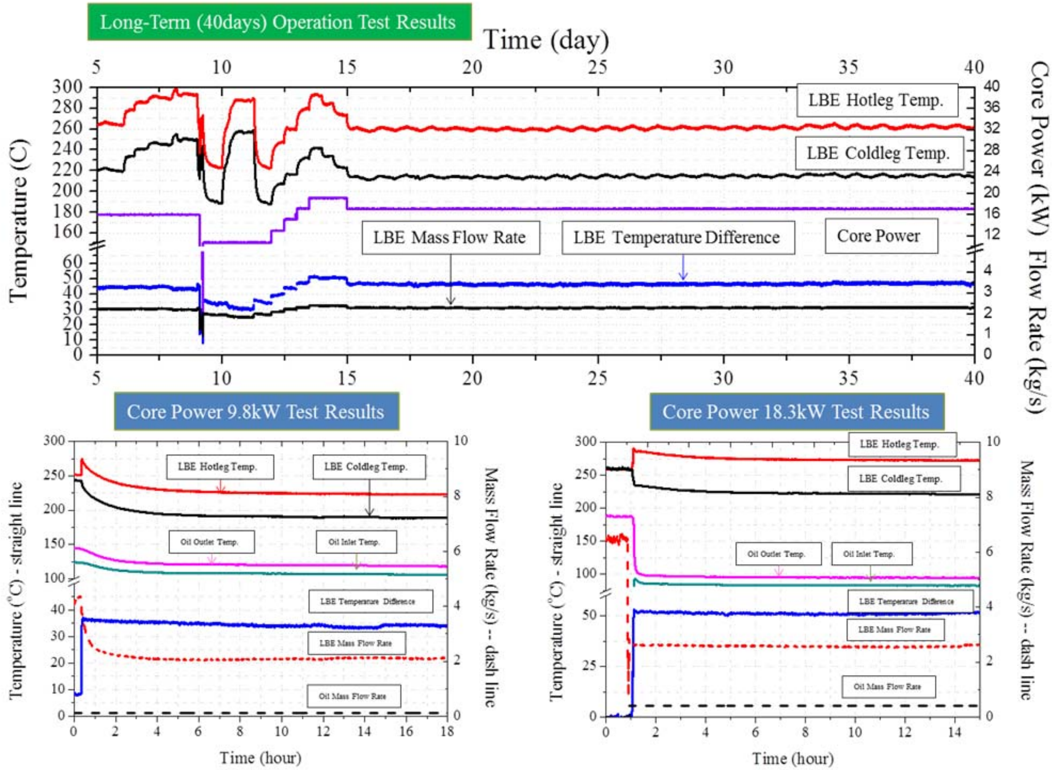


Figure 5.18. Experimental results for natural circulation

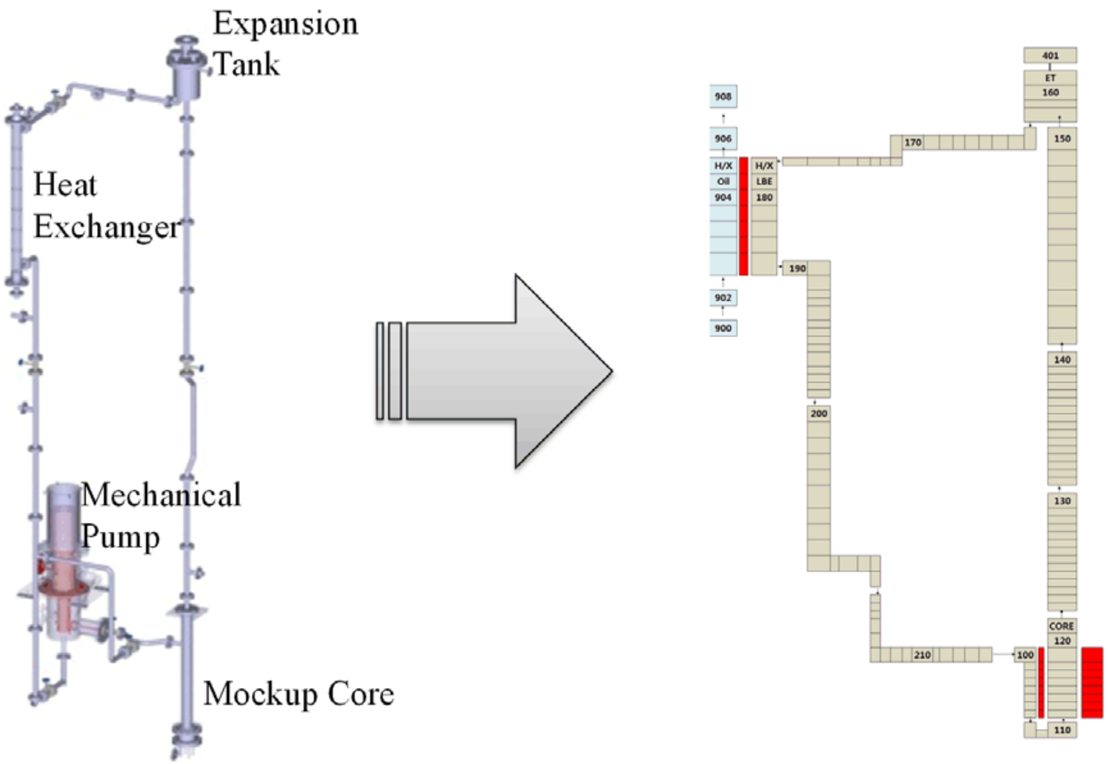


Figure 5.19. Nodal scheme of HELIOS for MARS-LBE code

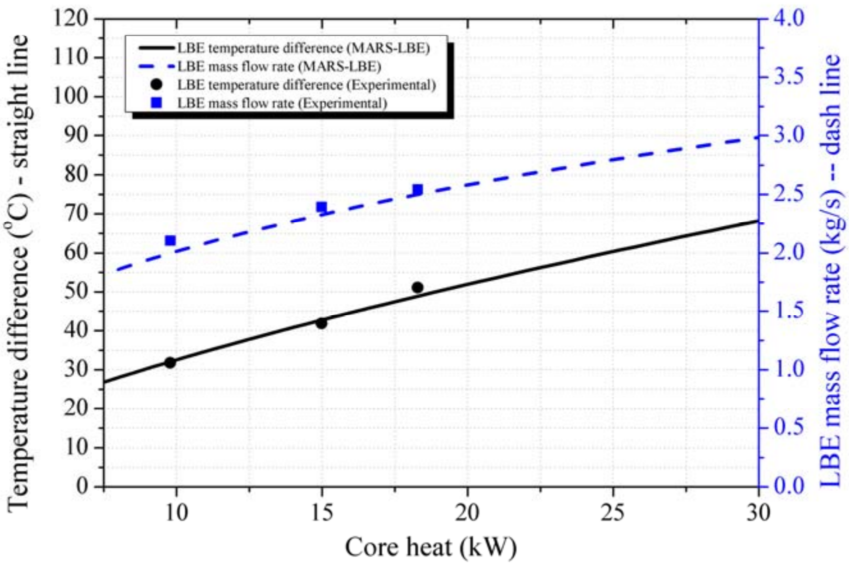


Figure 5.20. Predictions by MARS-LBE with measured data in HELIOS

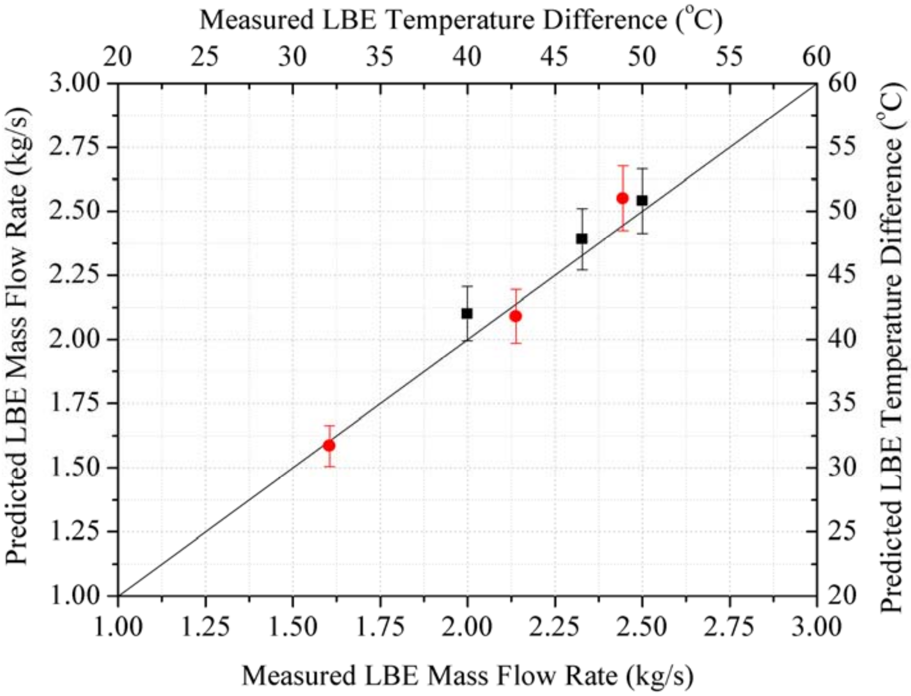


Figure 5.21. Predictions by MARS-LBE with measured data in HELIOS

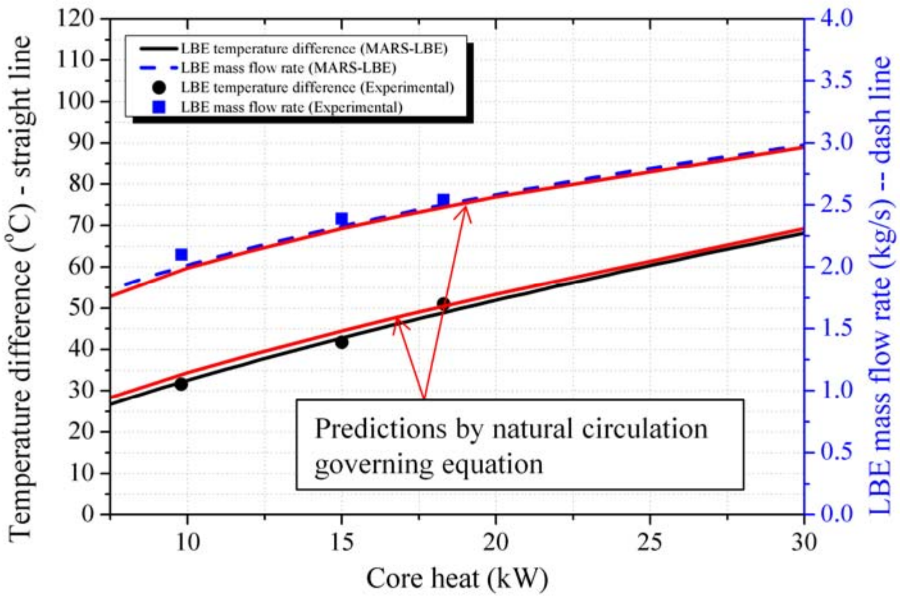


Figure 5.22. Predictions by natural circulation governing euqations with measured data

Chapter 6 Power Optimization Method

In order to develop the power optimization method as pre-conceptual design level, one equation that includes designing of natural circulation of primary coolant is needed. In the previous chapter, the equation of force balance between buoyancy force and resistance force is suggested. This equation can be derived to simple relationship between power and geometry.

$$\rho g \beta H_{cd} \Delta T = \sum \frac{1}{2} \rho V_i^2 \left(k_i + f_i \frac{l_i}{d_i} \right)$$

$$P_{core} = \dot{m} C_P \Delta T = \rho A_i V_i C_P \Delta T$$

$$V_i^2 = \frac{P_{core}{}^2}{\rho^2 A_i^2 C_P{}^2 \Delta T^2}$$

$$2 C_p{}^2 \rho^2 g \beta \frac{\Delta T^3}{P_{core}{}^2} = \sum \frac{1}{\Delta H \cdot A_i^2} \left(f_i \frac{l_i}{d_i} + k_i \right)$$

Based on this equation, If power (P) and geometry are fixed, coolant temperature difference (∆T) is determined by natural circulation. If power (P) and target coolant temperature difference (∆T) are fixed, geometry is determined to allow natural circulation. If geometry is given, sets of power (P) and coolant temperature difference (∆T) are determined to allow natural circulation.

6.1 Natural circulation cooling capacity

Hydraulic resistance is the most important in the predictions of natural circulation system because among the many uncertainties it is most unpredictable term. Conventionally, pressure drop is obtained by experimental test results for water coolant. Using non-dimension analysis with Reynolds number, the many correlations are used without consideration on difference of coolant. However, as mentioned in the chapter 5, many correlations have variations with measured data because pressure drop behavior of each specific geometry could not cover the it of similar geometry. Also, large viscosity of LBE coolant may give the difference with correlations by water base handbooks. In the contrasts of hydraulic-resistance handbooks, CFD simulations is good prediction power because it three dimensionally calculates the pressure drop with considering complexity of geometry and properties.

Friction loss coefficients and form loss coefficients are functions of velocity, flow rate and hydraulic diameter. Thus, equations could be changed to function of hydraulic diameter, flow area, flow length, coolant temperature difference, and reactor power.

$$\text{function}\left(d_i, A_i, l_i, \Delta T, P_{core}\right) = 0$$

6.2 Neutronics maximized power

In the optimization equation, there are free-bound parameters in core region, A_1,l_1,d_1 and p/d. Unlike with other geometric parameters, geometry of core region are inter-correlated with neutronics behavior. Relationship of the parameters of core region is derived by following procedures;

1) Determine the target burning cycle with consideration of economy benefit and nation nuclear law.
2) With limitation of displacement per atom (DPA) of fuel cladding materials, T91, determine the average fuel power density that not exceeds DPA limit during target burning cycle.
3) Obtain the correlations between power and core height for different fuel volume fraction. Power is function of fuel volume fraction and core height.
4) All power has smallest enrichment that have criticality during whole burning cycle. Also, large enrichment gives large excess reactivity. It could indicate the range of reactivity for all power.
5) Selected power region, optimized enrichment is determined to obtain smallest reactivity swing for all power.
6) For target burning cycle and fuel power density, one set of maximum power and core height is determined for fuel volume fraction within 1$ burnup swing and 2$ excess reactivity.

After above the procedure is conducted, power, core flow area, active core height, and core hydraulic diameter are correlated with each other by neutronics characteristics.

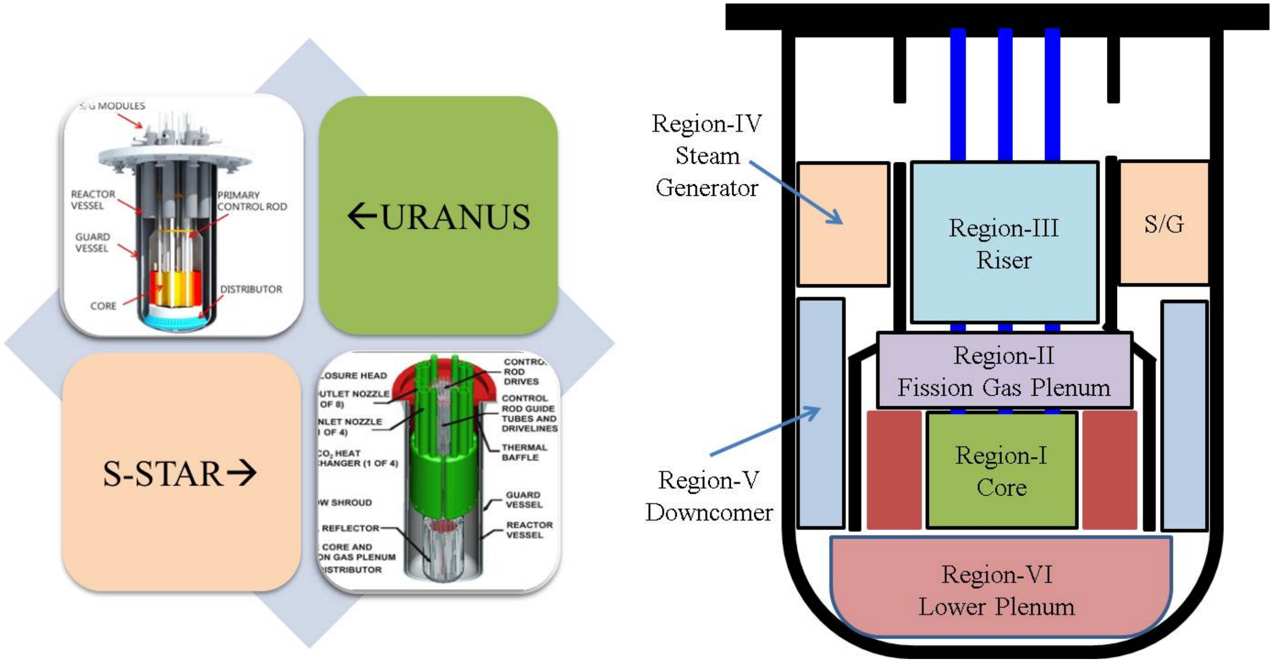


Figure 6.1. Reactor schematic Diagram

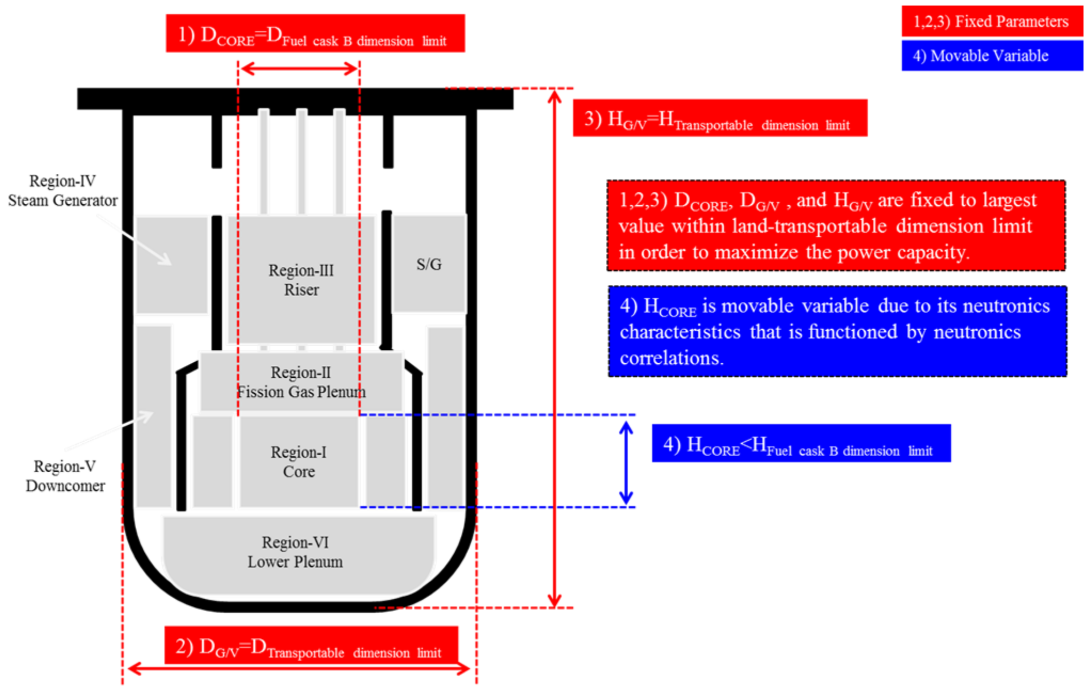


Figure 6.2. Land transportable shipping size limit

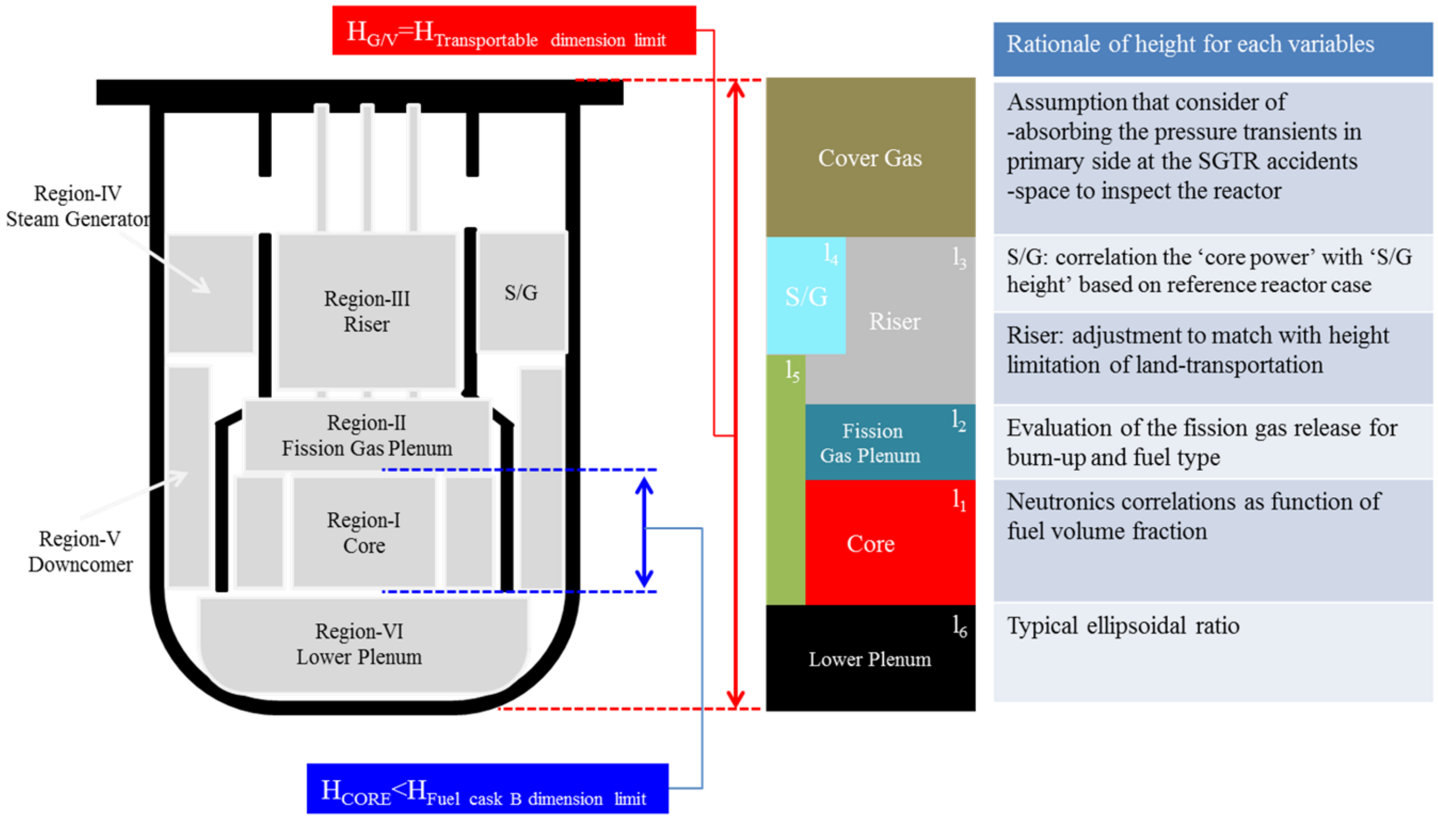


Figure 6.3. Land transportable shipping size limit

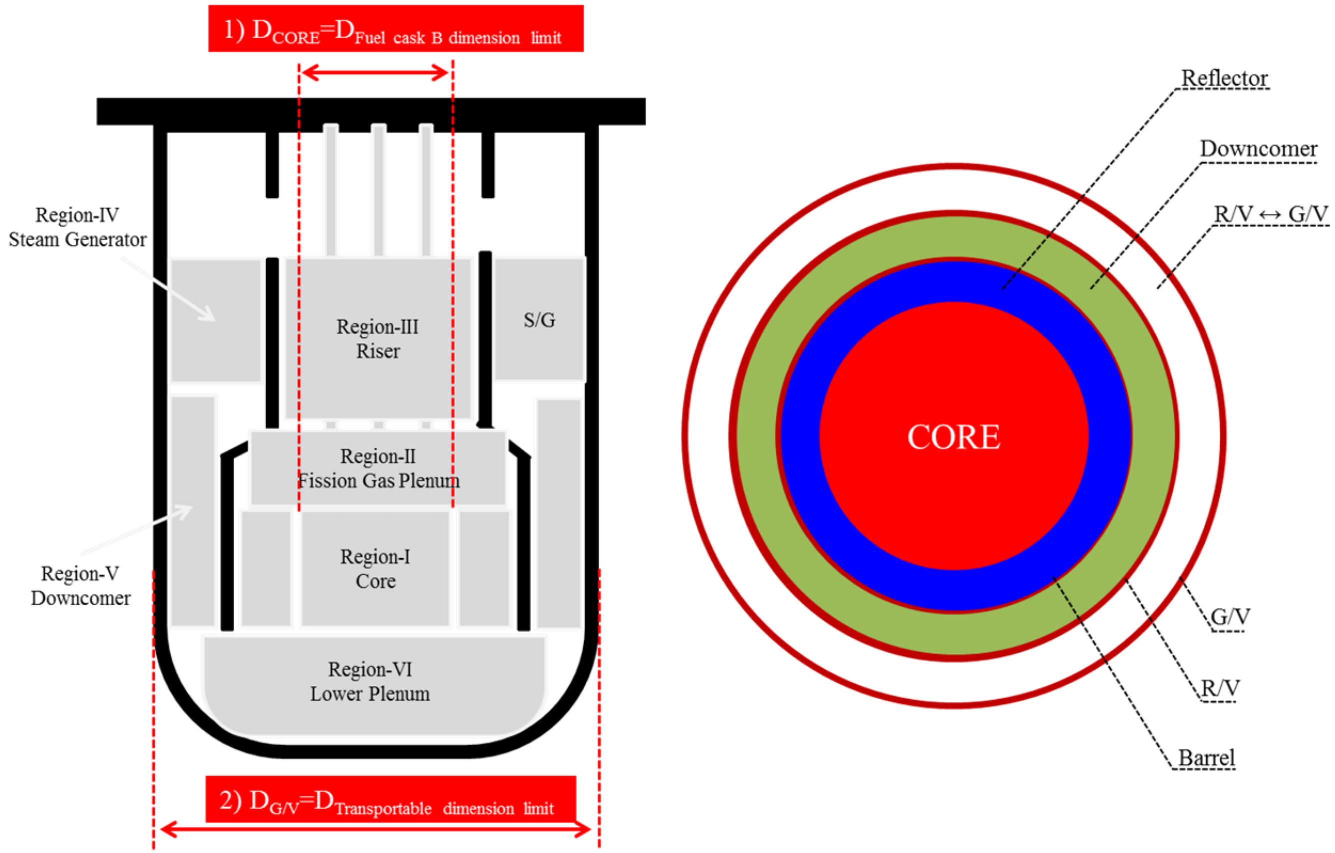


Figure 6.4. Land transportable shipping size limit

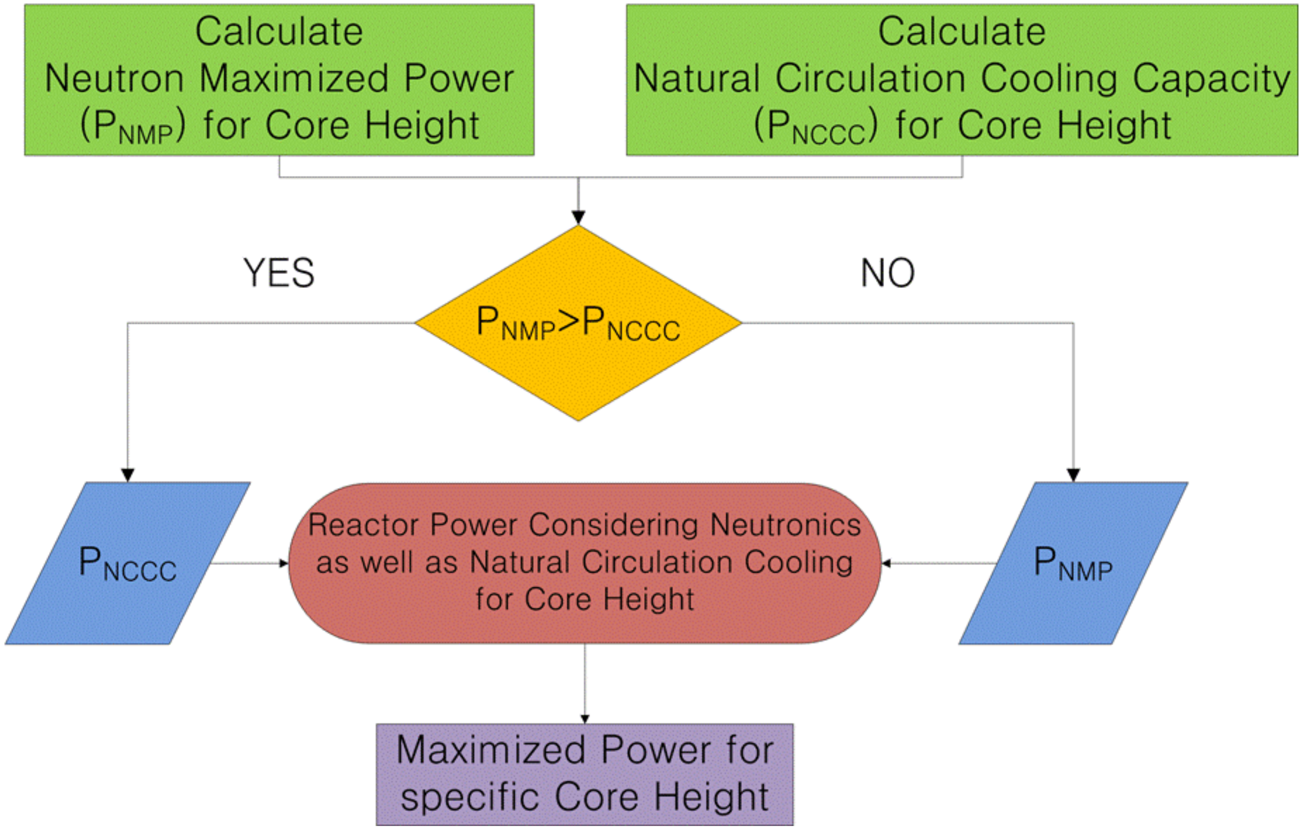


Figure 6.5. Method for calculation of maximized power for specific core height

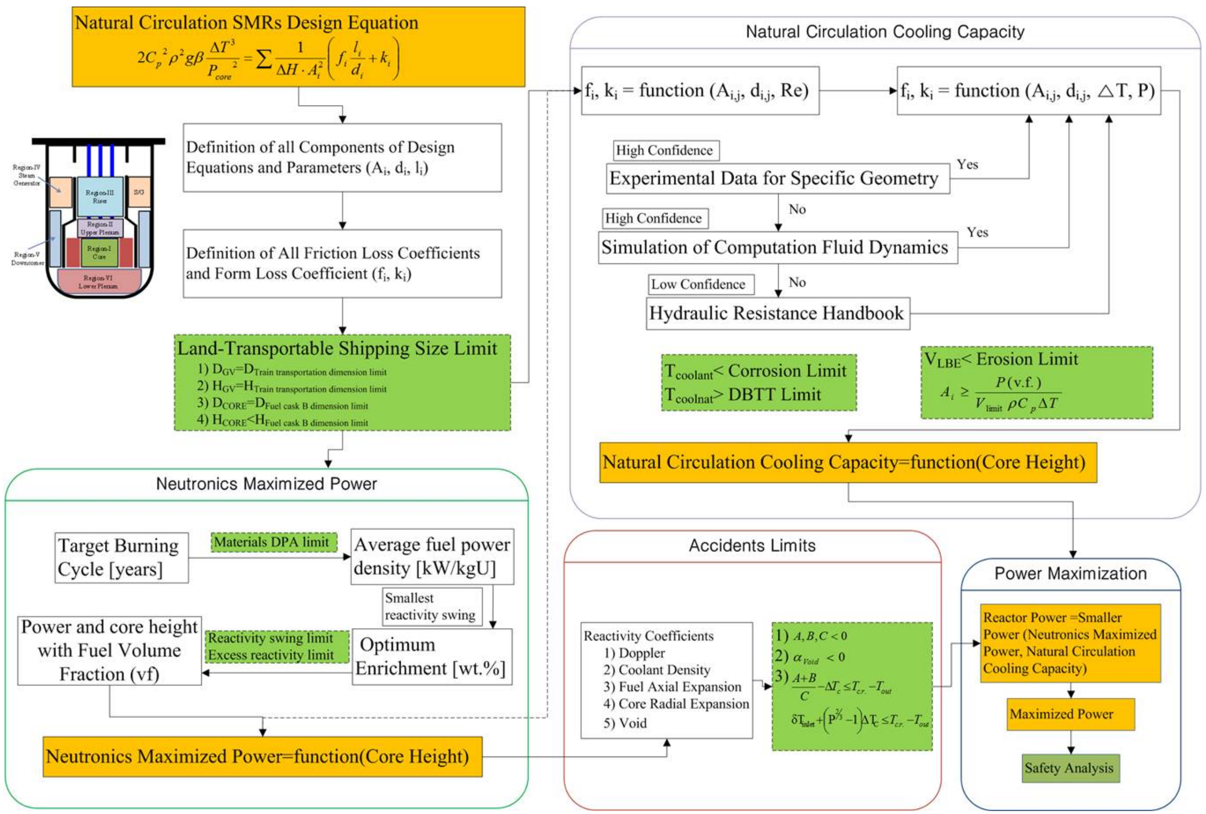


Figure 6.6 Power optimization method

Chapter 7 Case Study: URANUS

Concepts and goals of URANUS-40 are described as follows.

1) Ubiquitous

   Adequacy of safety margins with URANUS-40 permits the flexibility in siting, to include those near civic center, those remote isolated areas with extreme climates. This flexible site options greatly expand marketability, with the freedom from electrical grids or in connection with Smart Grid. Reliable seismic isolation design allows for an earthquake of 0.5g zero period acceleration (ZPA) for the Safe Shutdown Earthquakes (SSE).

2) Rugged

   The rugged capsular design of a whole core assures the safeguards for sensitive nuclear fuel materials. The containment including URANUS-40 is built under the ground which assures rugged features in support of air defense, explosion proof, and protection from external terrors.

3) Accident-forgiving

   The URANUS-40 has significantly greater safety margin against core damage from all conceivable accident conditions. It combines the chemical inertness of heavy liquid metal coolant and the passive cooling capability of a small reactor with high surface to volume ratios. It is designed to have the core damage frequency less than

10^{-8} per reactor year that is 1,000 times lower than that of Generation III NPP. The seismic system of URANUS-40 can tolerate the severest earthquake in one million years for Korean peninsula. Under a hypothetical core disruption accident (HCD), URANUS-40 eliminates the chance for re-criticality due to the floating nature of nuclear fuel elements. Natural circulation design eliminates Loss of Flow Accident (LOFA) and low primary system pressure lead to “leak-before-break”. In addition the presence of guard vessel eliminates the chance for LOCA.

4) Non-proliferating

The proliferation risk can be effectively managed by limiting access to individual or a bundle fuels by the one-core capsule design. Therefore, refueling of URANUS-40 is performed by replacing the entire reactor vessel (no on-site refueling). The lead-bismuth eutectic coolant ahs Pu doubling time that is much longer than sodium case. In addition, URANUS-40 has a long refueling period making much less chance for accumulating spent nuclear fuels on site.

5) Ultra-lasting

Each fuel loading in URANUS-40 can be operated for up to 25 years. It can contribute to assure the energy security as well as decrease of fuel cost, increase of capacity and suppression of spent nuclear fuel.

6) Sustainer

   The modular URANUS-40 can be used to replace old coal thermal power stations, thereby reducing CO_2 emission to the environment. Volume of spent nuclear fuel can be reduced to over 20 times and the safety management time can be cut to about 300 year by closely working with a regional recycling facility.

The special features of the URANUS-40 are as follows:

1) Land-based nuclear power station

   The URANUS-40 is a land-based nuclear power station with the reactor building basically embedded underground for assured security and physical protection while enhancing passive cooling characteristics. [4]

2) Ultra-long refuelling interval and whole core refueling

   The URANUS-40 is designed for a long core life (for up to 25 years) and the entire reactor vessel with the core can be replaced and a reactor cover seal-welded to the vessel makes its core physically inaccessible. [5]

3) Shop-fabrication, transportability, and modular construction

   In order to reduce costs and to enhance quality of production, all components of URANUS-40 are shop-fabricated. Capsular design of URANUS-40 permits transportation by barge and/or truck. The production process will cost less with fast learning effect, to economically compete with existing large power systems. [4]

4) Naturally-circulated primary coolant

   Lead-bismuth eutectic (LBE) coolant of URANUS-40 is naturally-circulated by employing large hexagonal open lattice and by adequate elevation difference between the reactor core and the steam generator. Primary pumps are avoided facilitating long-term reliable operation. This also eliminates any chance of the loss of flow accidents (LOFA). [9] Potential slow fouling can be reversed by special chemistry systems.

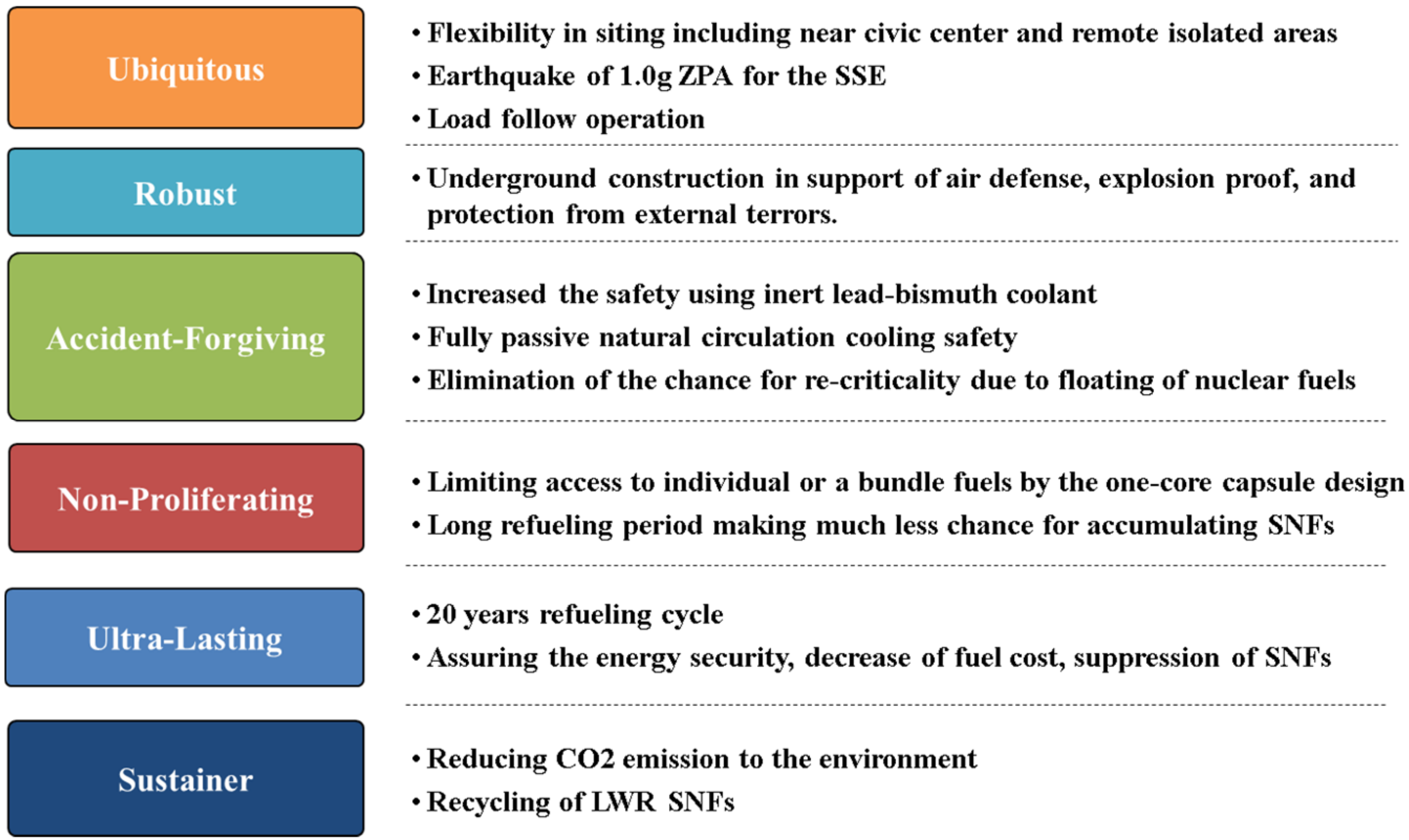


Figure 7.1. Concept of URANUS-40

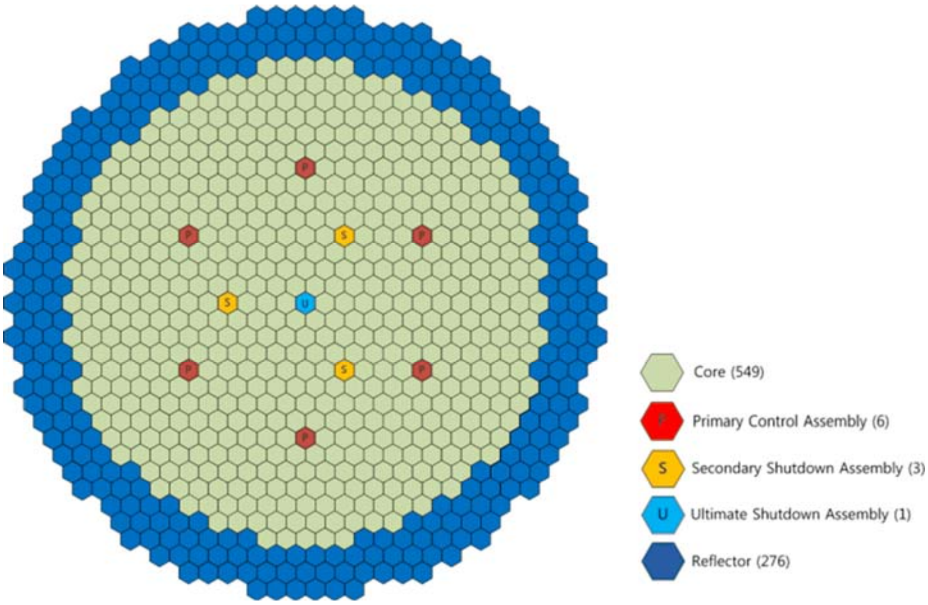


Figure 7.2. Core configuration for case study

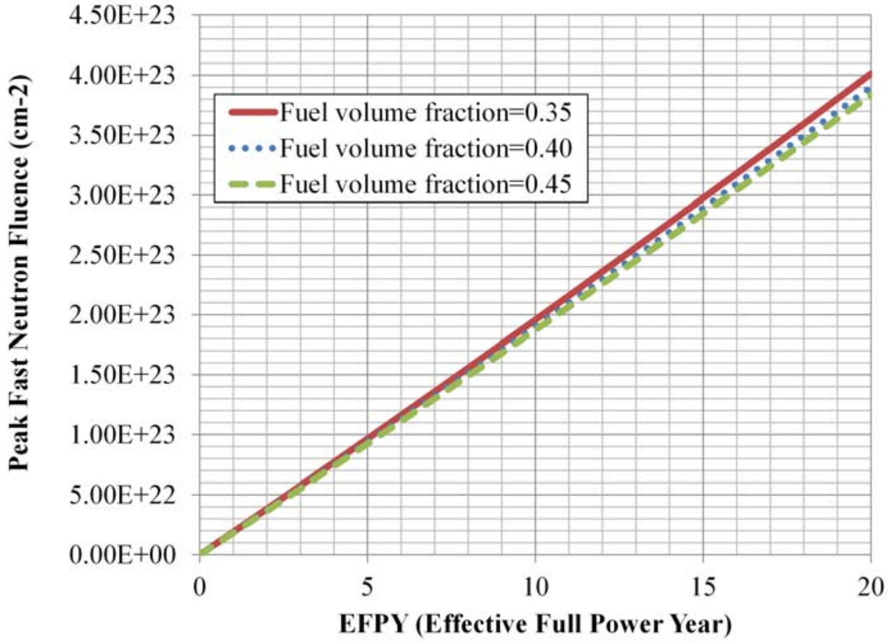
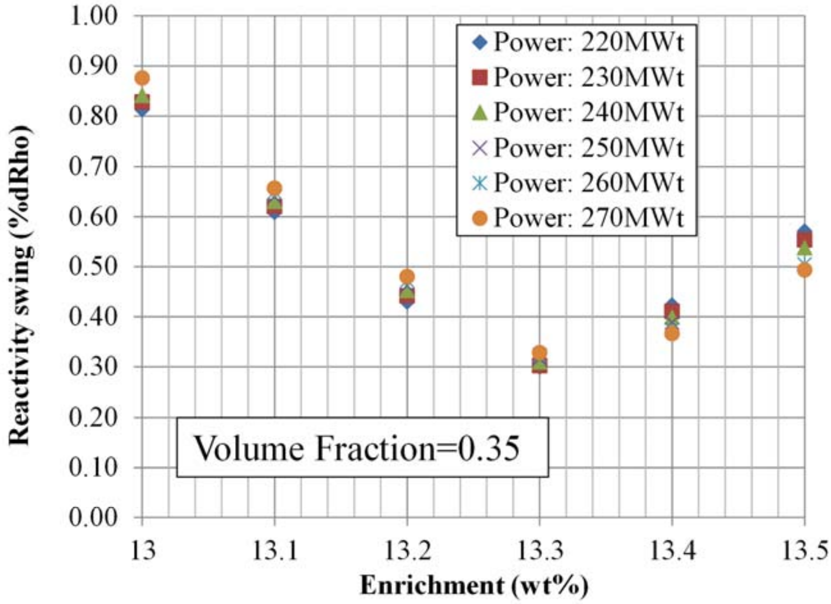


Figure 7.3. Peak fast neutron fluence for fuel volume fraction



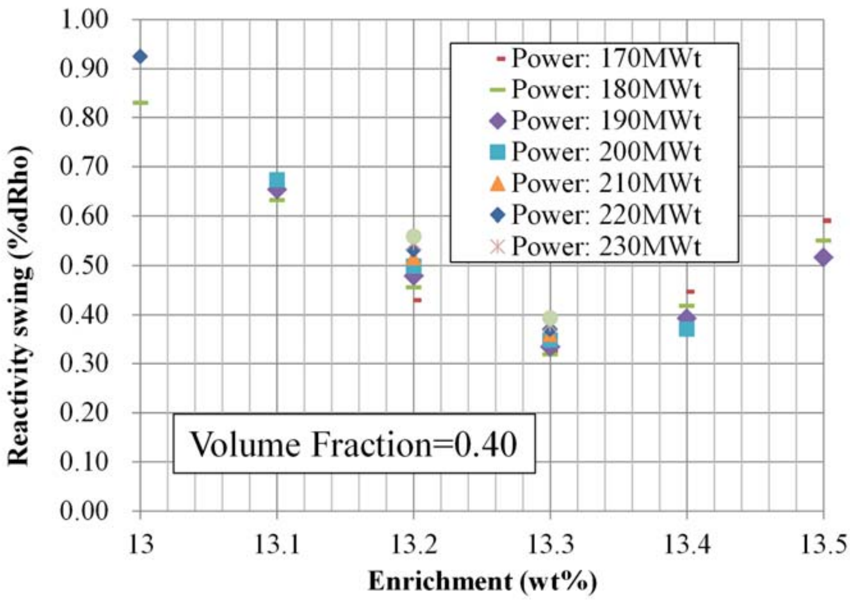


Figure 7.4. Optimized reactivity swing

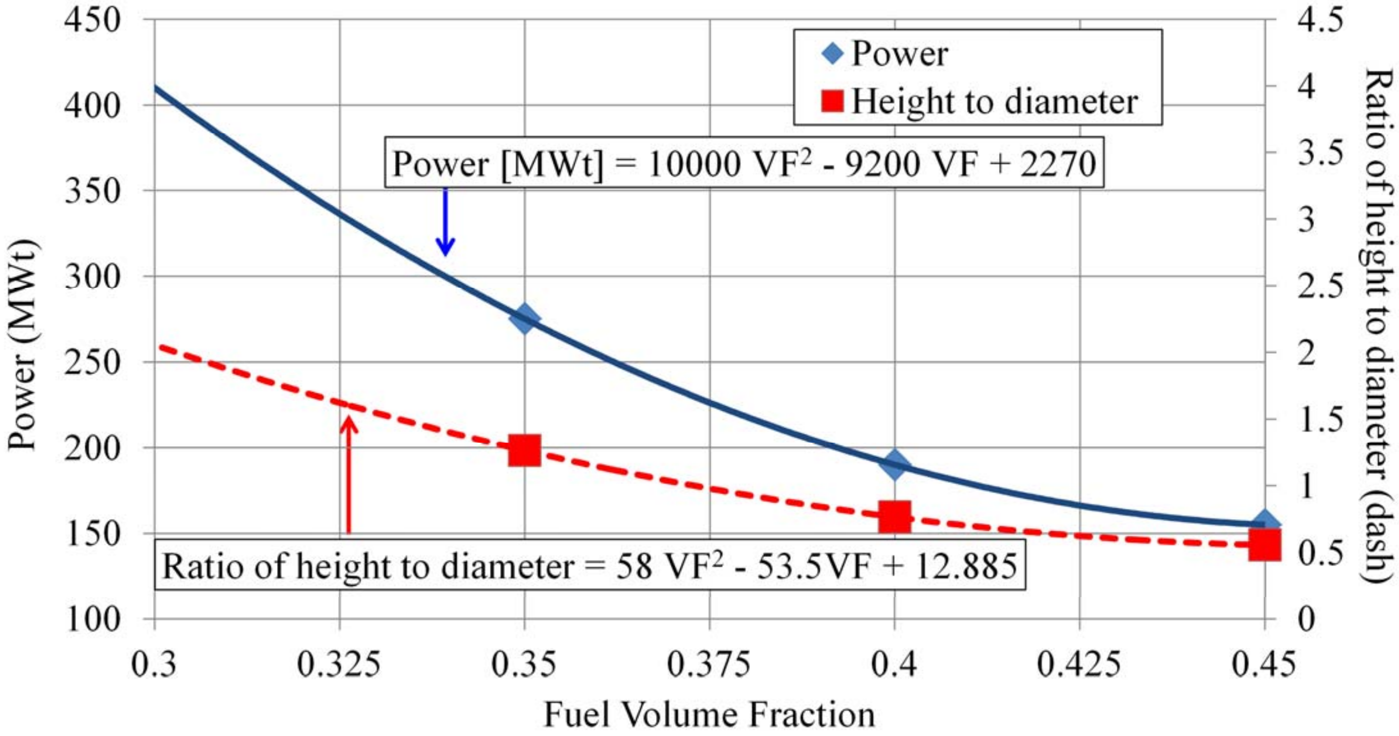


Figure 7.5. Power and height as function of fuel volume fraction

Chapter 8 Summary and Conclusion

Steady state limitations including land-transportable shipping size limits, materials limits, and neutronics limits are determined based on “10CFR 50 Appendix A General Design Criteria” of NRC. The quasi-static reactivity balance equation is used to obtain the limitations of reactivity in accidents conditions for selected BDBEs; UTOP, ULOHS.

- $\delta T_{out}=\frac{A+B}{C}-\Delta T_C < T_{C.R.}-T_{out}$
- $\delta T_{out}=\frac{-\Delta\rho_{TOP}}{C} < T_{C.R.}-T_{out}$
- $\delta T_{out}=\left[\left(1+\frac{-\Delta\rho_{TOP}}{A+B}\right)^{\frac{2}{3}}-1\right]\Delta T_C < T_{C.R.}-T_{out}$
- Void coefficients should be negative because steam could penetrate into core in unprotected STGR.

LBE coolant experiments using HELIOS facility are conducted. In the forced convection tests, pressure losses of core, orifice, gate valve, and expansion tank are obtained. In the natural circulation tests, temperature distribution and mass flow rate are obtained for specific core heat. Long-term stability of LBE natural circulation is confirmed by 600-hours experimental tests. Predictions for hydraulic-resistance and natural circulation behavior of experimental results are conducted. Pressure loss coefficients of measured data are agree well with CFD results within ±15%.

Natural circulation experimental results are agree well with MARS-LBE predictions using recommended pressure loss coefficients. Predictions by natural circulation governing equation are agree well with measured data when they are updated with recommended pressure loss coefficients.

Power optimization method is developed with design flow chart. Using this method, easy pre-conceptual design of natural circulation based SMRs could be available.

Application of developed power optimization method to other SMRs design could be available. Case study for 20 years long-burning small modular reactor with LBE natural circulation using the power optimization method shows the maximized power is 206MWt.

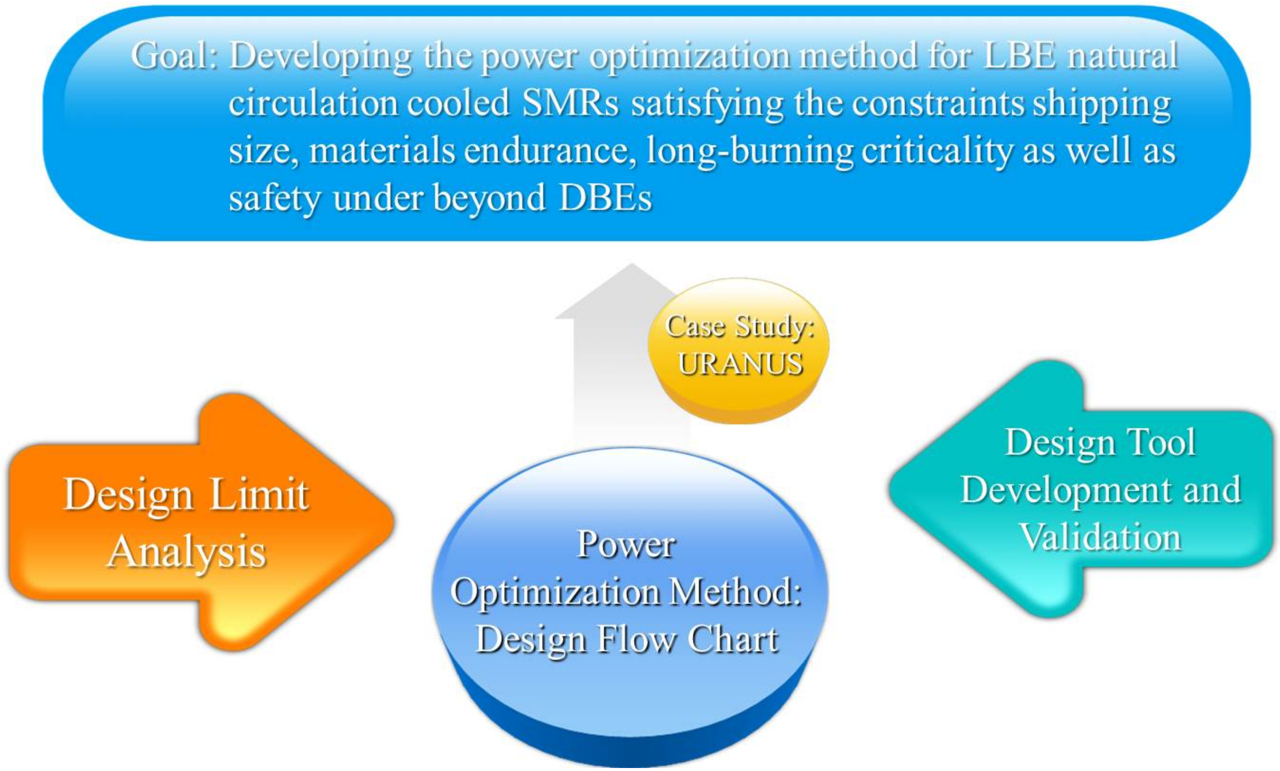


Figure 8.1 Summary diagram

Chapter 9 Future Work

1) Obtaining the maximum power as function of core diameter by case study for various core diameter (Dcore=2m in this dissertation)

2) Providing favorable reactor power and physical size package for various energy demand from developing nations and developed nations with electricity, desalination, and nuclear ships

3) Quantitatively evaluation for maximum power of SMRs with sodium and water natural circulation. Design index of nuclear energy for various energy spectrum demand highlighting the weaknesses and strengths of each coolant.

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초 록

피동 안전성을 확보하고, 모듈화, 대량생산을 통해 경제성을 높이고, 거대 전력망이 연결되어 있지 않은 벽지에 설치하여 유연하게 각 지역에 전기를 공급하는 것이 가능하다는 장점으로 소형모듈화 원자로에 대한 수요가 전 세계적으로 발생하고 있다. 특히 후쿠시마 사태 이후에 안전성의 극대화가 원자력발전소의 최우선 목표가 되면서 피동안전계통이 강조되고 있어 소형모듈화원자로에 펌프를 제거한 완전피동 자연 순환 냉각계통이 제시 되고 있다. 한편, 전 세계적으로 지속적으로 누적된 가압경수로의 사용후핵연료의 처분문제가 심각한 상황이므로, 이를 위해서 고속중성자를 이용하여 사용후핵연료를 핵 변환하여 중저준위화 하는 장주기 소각원전의 연구가 진행 중에 있다. 장주기 소각 원자로의 냉각재로서 납-비스무스는 물, 공기와 반응하지 않으며, 자연순환에 유리하며, 소형 노심에 대해서 음의 기포 반응도 계수를 확보할 수 있는 등 여러 가지 고유의 안전성이 있으므로 장주기 소형모듈화원자로에 적합하다. 따라서 납-비스무스 자연순환 냉각 소형모듈화원자로는 현재 경수로가 가진 현안인 안전성, 경제성, 및 핵폐기물 문제를 해결할 수 있는 대안으로 판단된다.

납-비스무스 냉각 소형모듈화원자로는 미래의 에너지 수요에 맞추어 여러 가지 활용이 가능한데, 대표적으로 선진국 시장에서의 친환경 분산형 전원 시스템 (스마트 그리드)에 맞는 에너지원, 개발도상국의 에너지원, 중동 및 아프리카 지역의 해수담수화, 원자력 선박이용 등이 있다. 이러한 다양한 에너지 수요에 대해서 기존 가압경수로의 역할과는 차별화되게 소비자가 원하는 대로 유연하게 출력을 제공하여 경제성 및 지역 수용성을 최적화하는 것이 소형모듈화원자로의 큰 장점이다. 그러나 납-비스무스 냉각 소형모듈화원자로가 갖는 여러 가지 제한 조건에 의하여 출력을 무한히 높이는 것이 제한된다. 공장에서 제작된 모듈들이 벽지까지 운

송이 가능해야 하므로, 모듈의 크기가 육상운송가능 크기에 제한되며, 재료의 부식, 침식, 및 방사화 제한조건을 만족하기 위하여 계통의 온도, 냉각재 속도, 및 노심의 중성자 밀도 등이 제한되며, 장주기운전 동안의 임계 유지를 위하여 노심의 형상 및 농축도 등이 특성화되어진다.

따라서 본 논문의 연구 목표는 납-비스무스 냉각 소형모듈화원자로의 여러 제한 조건들을 만족하면서, 최대로 낼 수 있는 출력을 구하는 방법론을 도출하는 것이다. 이를 이용하여 사례연구를 수행하여 납-비스무스 냉각 소형모듈화원자로의 출력 유연성을 정량적으로 도출해내고, 더 나아가 출력 최적화 방법론을 다른 냉각재 또는 다른 제한 조건들에 적용하여 자연순환 기반 소형모듈화 원자로의 최대 출력을 구할 수 있다는 데 본 논문의 의의가 있다.

이를 위해 본 논문에서 다루게 될 연구 질문은 1) 납-비스무스 자연순환 냉각 소형모듈화원자로의 설계 제한조건들이 무엇인가? 2) 어떤 코드 및 식을 이용하여 설계를 수행할 것이며, 검증되어 있는가? 3) 설계 제한조건들과 설계 도구를 이용하여 어떻게 출력 최적화 방법론을 개발할 것인가? 이다. 이에 대한 답을 찾기 위해 크게 네 단계의 과정을 통해 연구를 진행하였다. 첫 째, 제한 조건을 정상상태 제한치와 사고 제한치로 나누어, 정상상태 제한치에 대해서는 NRC에 제시 되어 있는 10CFR50 Appendix A General Design Criteria의 철학을 바탕으로 납-비스무스 냉각재, 자연순환, 및 소형 모듈화의 개념에 맞게 설계 조건들을 정의하였고, 사고 제한치에 대해서는 설계 기준 초과 사고를 정의하고 이에 대해 준정적 반응도평형식에 의하여 계산된 냉각재 출구온도의 변화를 제한조건으로 설정하였다. 둘 째, MARS 코드에 납-비스무스 냉각재 물성치 및 열전달 상관식을 수정하여 MARS-LBE 코드를 구축하고, 예측한 값과 납-비스무스 냉각재 자연순환 실험치와 비교하여 코드의 예측능력을 검증하였다. 셋 째, 앞 서 제시된 제한조건 및 설계도구를 이용하여 출력을

최적화하는 설계 순서도를 제시하여 방법론을 개발하고, 이 방법론을 적용하여 납-비스무스 냉각 소형모듈화원자로가 갖는 최대 출력을 도출했다.

long burning, fast reactor, lead-bismuth eutectic, small modular reactors, power maximization, natural circulation

주요어: 고속로, 납-비스무스, 소형모듈화원자로, 출력 최대화, 자연순환
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