Impact of Truly Optimized PWR Lattice on Maximum Power of Natural Circulation Reactor

Steven Wijaya [a] and Yonghee Kim [a*]
[a]*Department of Nuclear and Quantum Engineering, KAIST, 291 Daehak-ro Yuseong-gu, Daejeon, Korea, 34141*
[*]*Corresponding author: yongheekim@kaist.ac.kr*

1. Introduction

Recently, Small Modular Reactor (SMR) becomes one of the attractive options for energy mixes due to the reduced capital cost, compact module and enhanced safety performance. The SMR safety is enhanced further by integrating passive cooling system into the reactor vessel [1]. In passive cooling SMR, the heat is completely removed by natural circulation without reactor pump. As the mass flow rate is lower, the passive cooled SMR has lower power than the pump-cooled SMR. In passive cooled SMR, the core mass flow rate is determined by the balance of the driving force and resistance force of the primary cooling system [2]. The reactor power, geometrical design of the reactor system, particularly the fuel assembly (FA) design, and the operation state of heat exchanger influence the reactor thermal-hydraulic performance [3].

Recently, a high-performance Soluble-Boron-Free (SBF) SMR, named ATOM, has been developed [4]. ATOM's neutronic performance is significantly enhanced by using an optimal PWR lattice, so-called Truly Optimized PWR (TOP) lattice, and the SBF operation is achieved by utilizing Disk-type Burnable Absorber (DiBA). If the core size is restricted, a TOP lattice still can be achieved by reducing the fuel rod diameter for a given pitch. However, the optimization of TOP has not considered the thermal-hydraulic condition, particularly under passive cooling conditions.

In this paper, the application of TOP on a natural circulation cooled SMR is investigated. NuScale reactor will be utilized as the base model design. NuScale developed a natural circulated SMR based on the well-established PWR technology since 2000 and currently under the US-NRC review for commercial licensing [5]. However, as the TOP model is applied for SBF core and the NuScale reactor is utilizing the soluble boron, it is assumed that the NuScale Core can be successfully converted to the SBF core. An in-house code is developed for the analysis. The fuel pin pitch is varied to observe the impact of the reduced pressure drop at the core to the improvement of the system mass flow with the same temperature difference as the constraint. Therefore, the generated power will be increased. This study will be a preliminary step to find the TOP lattice for SMR with natural circulation.

2. Calculation Models

The steady-state model is being considered in this study. The reactor is modeled as parallel channels with the individual flow and the cross-flow between the internal components are not considered. The analysis will be done within the primary circulation loop following the primary coolant circulation.

2.1 Core Heat Transfer Model

In the current study, the thermal-hydraulics (TH) code is not coupled with the neutronic code yet. Therefore, the axial power distribution is determined by a chopped cosine function. The axial heat conduction is neglected to allow the analysis to be done at the axial level, channel by channel basis. The standard heat conduction and convection equation derived from the energy transport equation is utilized [2]. The analysis will be done in one-dimension, steady state condition and accommodating the local boiling. Dittus-Boelter correlation will be used at the sub-cooled region and Jens-Lottes correlation will be used at nucleate boiling region, which are written as follow:

Dittus-Boelter [6]:

$$Nu = 0.023\,\mathrm{Re}^{0.8}\,\mathrm{Pr}^{0.4} \tag{1}$$

Jens-Lotte [7]:

$$\frac{q"}{10^6} = \frac{\exp(\frac{4P}{6.2x10^6})}{25^4}(Tw - Tsat)^4 \tag{2}$$

2.2 Core Mass Flow Rate Model

The one-dimensional steady-state natural primary loop momentum equation for natural circulation system [2] can be written as:

$$\Delta P_{loss} = \Delta P_{buoyancy} \tag{3}$$

The coolant in the system is driven by the density differences between the hot inlet and cold inlet (buoyancy force). The right-hand side of equation 4 is the buoyancy force, which is the driving force of system coolant, and defined as:

$$\Delta P_{buoyancy} = (\rho_{cold} - \rho_{hot})g\Delta H \tag{4}$$

where ΔH is the thermal center differences between the reactor core and primary heat exchanger, ρ_{cold} is the coolant density at cold pool, ρ_{hot} is the coolant density at hot pool and g is the gravity acceleration constant. The left-hand side of equation 4 is the summation of all pressure drop in the primary circulation and can be written as:

$$\Delta P_{loss} = \Delta P_{lowplenum} + \Delta P_{core} + \Delta P_{riser} + \Delta P_{upplenum} + \Delta P_{SG} + \Delta P_{downcomer} \tag{5}$$

The coolant mass flow in the primary flow is calculated by using equation 3 as the constraint. The detail calculation flow is described in section 2.5.

2.2.1. Core Pressure Drop Model.

The total pressure drop in the core is calculated using following formula:

$$\Delta P_{core} = \Delta P_{inlet} + \Delta P_{friction} + \Delta P_{spacer} + \Delta P_{outlet} \quad (6)$$

where

$$\Delta P_{inlet} + \Delta P_{oulet} = (K_{inlet} + K_{oulet})\frac{1}{2}\rho v^2, \quad (7)$$

$$\Delta P_{fric} = f_{core}\frac{L_{core}}{D_e^{core}}\frac{1}{2}\rho v^2 \quad (8)$$

K is the loss coefficient term, v is the coolant velocity, f_{core} is the friction factor at core, D_e^{core} is the equivalent core diameter, and L is the core length. The spacer pressure drop is calculated using Rehme's formula [8] as follow:

$$\Delta P_{spacer} = N_{spacer} C_v (\frac{\rho V_v^2}{2})\left(\frac{A_s}{A_v}\right)^2 \quad (9)$$

where C_v is the drag coefficient, V_v is the average bundle fluid velocity, N_{spacer} is the number of spacer grids, A_s is the projected frontal area of spacer, and A_v is the unrestricted flow area. The drag coefficient is calculated using Dalle Donne formulation [9] as follow:

$$Cv = MIN[3.5 + \frac{73.14}{\mathrm{Re}^{0.264}} + \frac{2.79x10^{10}}{\mathrm{Re}^{2.79}}, \frac{2}{(\frac{A_s}{A_v})^2}] \quad (10)$$

2.2.2 SG Pressure Drop.

NuScale uses the Helical Coil-type SG (HCSG) as the primary heat exchanger (PHX). It is assumed that the tube configuration in the HCSG is in-line. Therefore, the pressure drop of the primary coolants flow through the tube with in-line configuration is calculated using Gaddis-Gnielinski correlation [10] as follow:

$$\Delta P_{HCSG} = Eu\frac{1}{2}\rho u_{\max}^2, \quad (11)$$

$$u_{\max} = \frac{a}{a-1}u_{mean}, \quad (12)$$

$$Eu = \xi N \quad (13)$$

where Eu is the Euler number, $u_{\max}$ is the maximum velocity in minimum cross-section area, u_{mean} is the mean velocity, N is the number of tube column, a is transversal pitch to outer tube diameter ratio and ξ is the drag coefficient. The drag coefficient is the summation of the contribution of drag loss due to laminar flow (ξ_{lam}), turbulent flow (ξ_{turb}), inlet and outlet effects (f_n). The drag coefficient is calculated using following equations:

$$\xi = \xi_{lam} + (\xi_{turb} + f_n)[1 - \exp(-\frac{\mathrm{Re}_d + 1000}{2000})], \quad (14)$$

$$\xi_{lam} = 280\pi\frac{(b^{-0.5} - 0.6)^2 + 0.75}{a^{1.6}(4ab - \pi)\mathrm{Re}_d}, \quad (15)$$

$$\xi_{turb} = \frac{f_t}{\mathrm{Re}_d^{0.1(\frac{b}{a})}}, \quad (16)$$

$$f_t = [0.22 + \frac{1.2(1-(\frac{0.94}{b}))^{0.6}}{(a-0.85)^{1.3}}]10^{0.47}(\frac{b}{a} - 1.5) \quad (17)$$
$$+0.03(a-1)(b-1),$$

$$f_n = \frac{1}{a^2}(\frac{1}{N} - \frac{1}{10});\ for 5 \le N \le 10$$
$$f_n = 0;\ for N \ge 10 \quad (18)$$

$$D_e^{HCSG} = (\frac{4a}{\pi} - 1)d; b > 1 \quad (19)$$

$$D_e^{HCSG} = (\frac{4ab}{\pi} - 1)d; b < 1 \quad (20)$$

$$\mathrm{Re}_d = \frac{D_e^{HCSG} u_{\max}\rho}{\mu} \quad (21)$$

where b is the longitudinal pitch to outer tube diameter ratio, Re is the Reynold number and μ is the coolant dynamic viscosity.

2.2.3 Pressure Drop of Lower Plenum, Riser Upper Plenum and Down Comer.

In the primary circulation loop, the pressure drop due to the friction contribution of lower plenum, riser, upper plenum and down comer are much smaller compared with the pressure drop of the core and HCSG. Therefore, these pressure drops are neglected.

2.3 HCSG Heat Transfer Model

HCSG consists of helical tubes carrying the secondary water and the primary coolant flows through the helical tubes. For the preliminary study, the HCSG heat transfer is modelled with several simplifications utilizing the predetermined secondary side condition (uncoupled). The secondary system is not modelled explicitly and the temperature at the secondary system is adjusted to ensure the HCSG heat transfer equal to the generated reactor heat. The equations to model the HCSG heat transfer in steady state condition are defined as follow:

$$G\frac{\partial h}{\partial z} = \frac{q'' P_h}{A}, \quad (22)$$

$$G\frac{(h(T_z^{primary}) - h(T_{z-1}^{primary}))}{z_i - z_{i-1}} - \frac{\left(T_z^{primary} - T_z^{secondary}\right)P_h}{R_{SG}A} \simeq 0 \quad (23)$$

where,

$$\frac{1}{R_{SG}} = \frac{Q}{A\Delta T_m}, \quad (24)$$

$$\Delta T_m = \frac{\Delta T_{\max} - \Delta T_{\min}}{\ln \dfrac{\Delta T_{\max}}{\Delta T_{\min}}}, \quad (25)$$

$$A = N_{tubes} P_h l, \quad (26)$$

$$P_h = \pi D_o \quad (27)$$

where h is the coolant enthalpy, P_h is the heated perimeter, A is the total heat transfer area, R_{sg} is the thermal resistance, Q is the total heat to be transferred to secondary side, $\Delta T_{\max}$ and $\Delta T_{\min}$ indicate the maximum and minimum temperature difference between primary and secondary side. N_{tubes} is the total number of tubes, l is the tube length and D_o is the outer diameter of helical tubes.

It is assumed that the heat transfer at lower and upper plenum, riser, and down comer is negligibly small

2.4 Parallel Channel Flow Distribution Model

Parallel channel flow distribution model [11] is utilized in this study. Considering N uniform, vertical, interconnected, parallel channels, the pressure drop from inlet to outlet in any channel can be written as:

$$\Delta P_{ch,n} = P_{ch,n}^{in} - P_{ch,n}^{out} \quad (28)$$

Utilizing the assumption that the inlet and outlet pressure of each channel are equal, the pressure equilibrium among the channels can be written as:

$$\Delta P_{ch,1} = \Delta P_{ch,2} = \Delta P_{ch,n}; n = 1, 2, 3, ...N \quad (29)$$

The mass conservation equation can be written as:

$$W_{total} = \sum_{n=1}^{N} W_n; n = 1, 2, 3, ...N \quad (30)$$

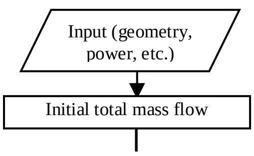
The energy conservation equation can be written as:

$$Q_n = W_n (h_{out,n} - h_{in,n}); n = 1, 2, 3, ...N \quad (31)$$

The mass flow rate and enthalpy rise of each channel can be determined by solving equation (27) to (29). As the current code is not coupled with neutronic code, a chopped cosine function is utilized to determine the axial power distribution.

2.5 Calculation Algorithm

The code reads the input data regarding the system geometry, power parameter and other necessary data. It is well-understood that there will be a non-linear iteration within the primary circulation loop pressure drop calculation because the core pressure drop calculation is also performed there. The detailed calculation flow chart is shown in Figure 1.



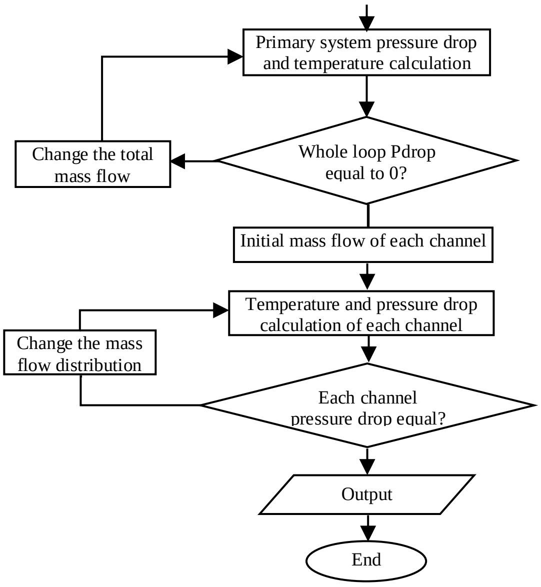


Fig. 1. Calculation flow chart

3. Numerical Result

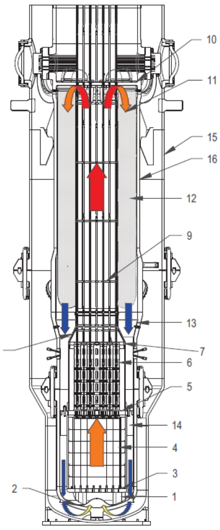
NuScale reactor uses water as the primary coolant for the natural circulation system. It's based on the standard PWR 17x17 FA with 160 MWth power for one nuclear power module (NPM) utilizing HCSG as the primary heat exchanger. The reactor pressure vessel with 17.7m height and 2.7m diameter contains the reactor core with 5 spacer grids per FA, pressurizer and HCSG. Table I describes the key parameter of NuScale reactor system.

Table I: Key parameter of NuScale reactor [5]

Parameter	Value
Core power	160 MWth
Height of active core	2 m
System pressure	12.75 MPa
Inlet temperature	531.5 K
Best estimate flow	587.15 kg/s
Core average coolant velocity	0.82 m/s*
Number of FA	37
FA pitch	21.5 cm
Fuel rod pitch	1.26 cm
Fuel rod diameter	0.95 cm
Number of helical tubes per NPM	1380
Tube column per NPM	21
Steam temperature	574.8 K
Feedwater temperature	422 K
HCSG Tube outer diameter	15.875 mm
HCSG Total heat transfer area	1665.57 m^2

*Using the reference mass flow rate, average core temperature and core flow area, we found that the average core coolant velocity is around 0.868 m/s

Figure 2 describes the overall NuScale reactor system operation. More information regarding the NuScale reactor system can be found at Reference [5]. However, as NuScale is going to be a commercial reactor, several key parameters, especially for the HCSG parameter, are not available. Utilizing the known temperature difference (ΔT) and approximated thermal center difference (8.354 m), the buoyancy force of NuScale reference design is calculated. Based on equation 3, the buoyancy force is equal to the system total pressure drop in steady state. Furthermore, in general the core P_{drop} ratio to the total P_{drop} is maximum 30% at best. These two parameters can be used as the constraints for the P_{drop} calculation despite of the incomplete information provided by the reference.



No.	Stage
1	Core support blocks in downcomer
2	Downcomer to lower plenum turn
3	Lower core plate
4	Core
5	Upper core plate
6	Control rod assembly guide tubes
7	Control rod assembly guide tube support plate
8	Riser transition
9	Control rod drive shaft support
10	Pressurizer baffle
11	Upper riser turn to annulus
12	Downcomer through steam generator
13	Downcomer transition
14	Upper core support blocks
15	Containment vessel
16	Reactor vessel

Fig. 2. NuScale reactor coolant system flow diagram [12]

Table II. Comparison of each pitch variant

Parameter	Pitch (cm)			
	Reference	1.26	1.35	1.4
Equivalent core radius (cm)	73.78	73.78	79.04	81.95
P_{drop}^{core} (Pa)	N/A	2332.40	1587.70	1293.40
P_{drop}^{HCSG} (Pa)	N/A	5090.50	5772.30	6041.70
P_{drop}^{other} (Pa)	N/A	443.992	506.83	531.77
$\dot{m}$ (kg/s)	587.15	587.06	627.23	642.48
converged $v_{coolant}^{core}$ (m/s)	0.820	0.869	0.732	0.666
Hot Temperature (C)	310.00	310.00	310.00	310.00
Cold Temperature (C)	258.11	258.87	258.87	258.87
Q (MWt)	162.23	160.00	170.94	175.11

In Table II, it is observed that for 1.26 cm pitch, the in-house code result is close to the reference result. The calculated power is slightly under the reference due to the slightly smaller temperature difference. The core P_{drop} ratio to the total P_{drop} is around 29.65 %. Therefore, the P_{drop} model is validated and can be used for the other pitch variants calculation.

In this study, the pitch is enlarged without changing the other geometry. As the pitch is enlarged, the core equivalent radius is also enlarged. The pitch variants analysis is done using the same ΔT as the constraint. It is observed that by enlarging the pitch, the core P_{drop} is reduced resulting in the increase of the total mass flow rate and power. By enlarging the fuel pitch, the power is increased by 6.8% and 9.4% for 1.35 cm and 1.4 cm pin pitch cases, respectively. The power increase is consistent with the mass flow rate increase under the same ΔT constraint. As the core P_{drop} decreases and the mass flow rate increases, the HCSG and the other loss form P_{drop} increase to satisfy the momentum conservation. As the core coolant speed is reduced with the increase of pitch and the Critical Heat Flux (CHF) is also function of $\sim v^{0.62}$, then the CHF is reduced by 10% and 15% for 1.35 cm and 1.4 cm pitch size, respectively. Furthermore, due to the lower average core coolant speed, the DNBR can be reduced.

3. Conclusions

Under the same ΔT constraint, power can be increased by 6.8% and 9.4% utilizing 1.35 cm and 1.4 cm fuel pin pitch, respectively. It is also well-understood that the core P_{drop} ratio affects the percentage of power gain. However, as the pitch size is enlarged, the average core coolant speed is decreased resulting in a lower CHF. As the CHF is also function of the flow diameter, pressure and speed, a comprehensive TH-analysis will be done in the future to determine the optimal TOP lattice for natural circulation cooled SMR. As the reactor is cooled utilizing natural circulation, Churchill-chu correlation will be implemented for comparison and finally, the neutronics code will be coupled for a high-fidelity multi-physics calculation.

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