Comparison of Truly Optimized PWR Lattice Designs for Natural Circulation Reactor

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1. Introduction

Integrated and compact modules resulting in reduced capital cost and enhanced safety performance make Small Modular Reactor (SMR) one of the attractive options for energy mixes. The SMR safety is enhanced further by integrating a passive cooling system into the reactor vessel [1]. In passive cooling SMR, the mass flow rate is much lower than the pump-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, and the operation state of the heat exchanger influence the passive cooled reactor thermal-hydraulic performance [3].

Recently, a Truly Optimized PWR (TOP) lattice has been demonstrated to be able to enhance the neutronic performance of a Soluble-Boron-Free (SBF) SMR named ATOM [4]. The pin pitch is enlarged to increase the hydrogen-to-uranium (HTU) number density ratio. SBF is necessary to assure negative Moderator Temperature Coefficient (MTC) for the reactor's inherent safety under any condition. Currently, there are two types of TOP lattice design being considered. The fuel pin pitch is enlarged to 1.4 cm in the first design without changing the fuel pellet radius. While in the second design, the fuel pellet radius is reduced to 0.38 cm without changing the fuel pin pitch. In the first design, the core size needs to be adjusted to accommodate the enlarged fuel pin pitch, while such adjustment is not required in the second design.

Previously, the impact of the TOP lattice with an enlarged fuel pin pitch on the power of the natural circulation core was investigated [5]. The second TOP design with a smaller fuel pellet radius is preferred when the TOP lattice is implemented to the existing core design with restrained core size. In this study, the impact of both TOP designs on the natural circulation SMR will be compared. NuScale reactor, a soluble-boron SMR, will be utilized as the base model design, assuming that the NuScale core can be successfully converted to SBF core [6]. Preliminary studies have been performed regarding the implementation of enlarged pin pitch TOP lattice design to the passive cooled SBF SMR core based on the NuScale core [7,8]. In this study, both the first and second TOP designs are implemented to compare the impact of both designs on the core pressure drop, system mass flow, and the reactor power under the constraint of the same inlet and outlet temperature. For an apple-to-apple comparison, the same in-house code [5] utilized in the previous study will be used for the analysis. The calculation models are briefly introduced in the next section.

2. Calculation Models

In this study, the steady-state condition is considered. The reactor is modeled (thermal-hydraulically) consisting of laterally closed parallel channels. The analysis will be done only on the primary coolant circulation loop.

2.1 Core Heat Transfer Model

In the current study, the thermal-hydraulics (TH) code is uncoupled with the neutronic code. A chopped cosine function is utilized to determine the axial power distribution. 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 are utilized [2]. The analysis will be done in one-dimensional, steady-state conditions and accommodating the local boiling. Dittus-Boelter correlation [9] and Jens-Lotte correlation [10] are utilized for the sub-cooled and nucleate-boiling regions, respectively. The correlations are written as follows:

Dittus-Boelter:

$$Nu = 0.023\,\mathrm{Re}^{0.8}\,\mathrm{Pr}^{0.4} \quad (1)$$

Jens-Lotte:

$$\frac{q''(W/m^2)}{10^6} = \frac{\exp(\dfrac{4P(Pa)}{6.2x10^6})}{25^4}(Tw(C) - Tsat(C))^4 \quad (2)$$

2.2 Core Mass Flow Rate Model

In natural circulation system, the one-dimensional steady-state natural primary loop momentum equation [2] is formulated as:

$$\Delta P_{loss} = \Delta P_{buoyancy} \quad (3)$$

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 3 is the buoyancy force, which is the driving force of system coolant, and derived using Boussinesq approximation as follows:

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

where ρ_{cold} is the coolant density at the cold pool, ρ_{hot} is the coolant density at the hot pool and g is the gravity acceleration constant and ΔH is the thermal center differences between the reactor core and primary heat exchanger. The left-hand side of equation 3 is the summation of all irreversible pressure drops in the primary circulation and can be written as:

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

The coolant mass flow in the primary flow is calculated using equation 3 as the constraint. Section 2.5 describes the detailed calculation model.

2.2.1. Core Pressure Drop Model.

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

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

where

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

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

f_{core} is the friction factor at the core, K is the loss coefficient term corresponding to the inlet and outlet, L is the core length, D_e^{core} is the equivalent core diameter, and v is the coolant velocity. The spacer pressure drop is calculated using Rehme's formula [11] as follows:

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

where N_{spacer} is the number of spacer grids, C_v is the drag coefficient, V_v is the average bundle fluid velocity, A_s is the projected frontal area of the spacer, and A_v is the unrestricted flow area. The drag coefficient is calculated using the Dalle Donne formulation [12] as follows:

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

2.2.2 Steam Generator Pressure Drop.

The Helical Coil-type SG (HCSG) is utilized in the NuScale core as the primary heat exchanger (PHX). In this study, an in-line tube configuration of the HCSG is assumed. Therefore, the pressure drop of the primary coolants flow through the tube with an in-line configuration is calculated using the Gaddis-Gnielinski correlation [13] as follows:

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

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

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

where Eu is the Euler number, $u_{\max}$ is the maximum velocity in minimum cross-section area, u_{mean} is the mean velocity, a is a transversal pitch to outer tube diameter ratio, N is the number of tube columns, 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:

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

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

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

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

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

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

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

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

where b is the longitudinal pitch to outer tube diameter ratio, Re is the Reynold number, d is the outer tube diameter 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 friction pressure drops of the lower plenum, riser, upper plenum, and down-comer are much smaller than the pressure drop of the core and HCSG. Therefore, these pressure drops are neglected in the current study.

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 scoping analysis, the HCSG heat transfer is modeled with several simplifications utilizing the predetermined secondary side condition (uncoupled). The secondary system is not modeled explicitly, and the temperature at the secondary system is adjusted to ensure the HCSG heat transfer equal to the generated reactor heat. The HCSG heat transfer equation in the steady-state conditions is defined as follows:

$$G\frac{dh}{dz} = \frac{q"P_h}{A_f}, \quad (22)$$

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

where,

$$\frac{1}{R_{SG}} = \frac{Q}{A_h \Delta T_m}, \tag{24}$$

$$\Delta T_m = \frac{\Delta T_{\max} - \Delta T_{\min}}{\ln \frac{\Delta T_{\max}}{\Delta T_{\min}}}, \tag{25}$$

$$A_h = N_{tubes} P_h^{tubes} l, \tag{26}$$

$$P_h^{tubes} = \pi D_o \tag{27}$$

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

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

2.4 Parallel Channel Flow Distribution Model

By considering N uniform, vertical, interconnected, parallel channels, the pressure drop [14] from inlet to outlet in any channel can be written as:

$$\Delta P_{ch,n} = P_{ch,n}^{in} - P_{ch,n}^{out} \tag{28}$$

Assuming the inlet and outlet pressures of each channel are equal, the pressure equilibrium among the channels can be written as:

$$\Delta P_{ch,1} = \Delta P_{ch,2} = \Delta P_{ch,n}; n = 1, 2, 3, \ldots N \tag{29}$$

The mass conservation equation can be written as:

$$W_{total} = \sum_{n=1}^{N} W_n; n = 1, 2, 3, \ldots N \tag{30}$$

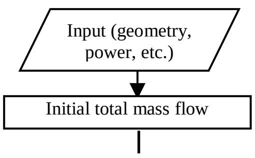
The energy conservation equation can be written as:

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

The mass flow rate and enthalpy rise of each channel are determined by solving equations (29) to (31).

2.5 Calculation Algorithm

The in-house code reads the input data regarding the system geometry, power parameter, and other necessary data. The detailed calculation flow chart is shown in Fig. 1.



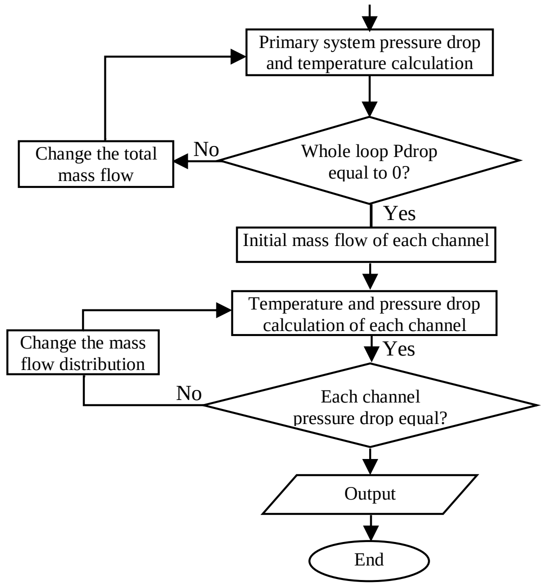


Fig. 1. Calculation flow chart

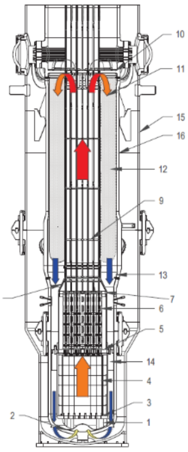
3. Numerical Result

NuScale reactor design is based on the standard PWR 17x17 FA with 160 MWth power for one Nuclear Power Module utilizing HCSG as the primary heat exchanger and water coolant. The reactor pressure vessel with 17.7 m height and 2.7 m diameter contains the reactor core with five spacer grids per FA, pressurizer, and HCSG. Table I describes the key parameters of the NuScale reactor system.

Table I: Key parameters of NuScale reactor [6]

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	1,380
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	1,665 m^2
Total primary coolant flow path	2,673 cm

Nuscale reactor coolant flow system is described in Fig. 2. Detailed information regarding the NuScale reactor system can be found in Reference [6]. However, as NuScale is going to be a commercial reactor, several vital parameters, especially for the HCSG parameter, are not published. Utilizing the known temperature difference (ΔT) and approximated thermal center difference (8.354 m), the buoyancy force of the NuScale reference design is calculated. Based on equation 3, the buoyancy force is equivalent to the system total pressure drop in the steady-state condition. Furthermore, in general, the core P_{drop} ratio to the total P_{drop} is maximum 30% at best. These two parameters are 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 [15]

Table II. Comparison of both TOP designs

Parameter	Pin pitch/fuel radius (cm)		
	1.26/0.405	1.26/0.38	1.4/0.405
Equivalent core radius	73.78 cm	73.78 cm	81.95 cm
P_{drop}^{core}	2,353 Pa	2,085 Pa	1,276 Pa
P_{drop}^{HCSG}	5,070 Pa	5,315 Pa	6,056 Pa
P_{drop}^{other}	444 Pa	467 Pa	535 Pa
$\dot{m}$	587 kg/s	602 kg/s	645 kg/s
$v_{coolant}^{core}$	0.87 m/s	0.83 m/s	0.66 m/s
Coolant flow area	245 m^2	265 m^2	354 m^2
$T_{coolant}^{hot}$	310 C	310 C	310 C
$T_{coolant}^{cold}$	259 C	259 C	259 C
Q	160 MWt	164 MWt	175 MWt

In Table II, it is observed that the 1.26/0.405 cm results are close to the reference result. The P_{drop}^{core} ratio to the total $P_{drop}$is around 29.9%. Therefore, the P_{drop} model is validated and can be used for the TOP design analysis.

The TOP design analysis is performed by using the identical ΔT as the constraint. By enlarging the coolant flow area in the core, the core P_{drop} is reduced, resulting in an increase of the total mass flow rate and power. The power is increased by 2.5% and 9.8% for 1.26/0.38 cm and 1.4/0.405 cm cases, respectively. It is observed that the 1.26/0.38 TOP design has a lower power increase due to the smaller increase in coolant flow area. The power increase is proportional to 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 for a consistent driving force with consistent hot and cold temperatures. As the core coolant speed is reduced with the increase of coolant flow area in the core, the CHF and DNBR value might be reduced. It is observed that the core coolant speed reduction of the 1.26/0.38 cm design is not as significant as the 1.4/0.405 cm case. Therefore, the DNBR reduction of the 1.26/0.38 cm design might be lower than the 1.4/0.405 cm design.

3. Conclusions

The coolant flow area in the core is the primary variable that dictates the impact of the TOP design on the core. Under the same ΔT constraint, reactor power can be increased by 2.5% and 9.8% utilizing 1.26/0.38 cm and 1.4/0.405 cm TOP designs, respectively. The 1.26/0.38 cm design has a smaller reduction of core coolant speed and is suitable for the existing core with a fixed size. Nevertheless, a comprehensive TH-analysis coupled with the neutronics code will be done to determine the exact CHF and DNBR value of both TOP designs to justify the benefits of both designs.

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