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Effects of Truly Optimized PWR Lattice Core Design on Reactor Power and Critical Heat Flux in a Natural Circulation Small Modular Reactor

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1. Introduction

In order to mitigate global climate crisis, many policies are aiming to achieve carbon neutrality by 2050. In particular, carbon-free energy sources such as solar, wind, and nuclear power have been highlighted. Among them, renewable energies are intermittent and sensitive to the climate and weather conditions. In addition, large site area is required due to their low power density. Although large nuclear reactor can generate high power steadily, low power flexibility and high initial cost are the factors which many counties hesitate to introduce the large reactors. Accordingly, many related experts propose a well-balanced hybrid energy system of renewable energy with small modular reactors (SMRs) [1].

SMRs are a remarkably attractive option for energy system even without renewable energies due to their advantages such as high applicability to multi-purposes, lower initial cost due to mass production from factories and shorter construction periods. In addition, smaller power, less than 300 MWe, and modular, simplified system design enhance the safety of SMRs. Accordingly, as the concepts of system modularity and simplification were greatly improved, natural circulation SMRs (NC-SMRs) such as VOYGR[TM] SMR (formerly known as NuScale Power Module) and CAREM, whose primary system operates only with natural circulation by removing the reactor coolant pumps (RCPs), have been proposed [2, 3]. By replacing the RCPs with natural circulation, cost of the RCP and possibility of accidents induced by the RCP are entirely eliminated.

To improve the economic feasibility of NC-SMRs, the power generation rate per reactor module needs to be as large as possible. The buoyancy in the primary system is determined by the density difference and relative elevation change between core and steam generators (SGs), length of thermal center. The density difference is affected by the coolant temperatures at the core entrance and exit. Although a large temperature difference is required to enhance buoyancy without an increase in the total height of the reactor pressure vessel (RPV), the core temperature is not arbitrarily changeable from a reactor-physics viewpoint. In addition, the increase in the length of the thermal center causes difficulties in modular design and manufacturing in a factory. Thus, to increase the coolant flow rate, the pressure drop in the NC-SMR system needs to be reduced.

The major pressure drops occur at the regions of core and SGs in the NC-SMR systems due to structures for supporting large tubes, and large heat transfer areas. However, the core and SG designs are considerably related to the reactor performance. In other words, they are not changeable only from the perspective of the thermal-hydraulic performance in the primary system. Here, as a method to reduce the pressure drop through the core, Truly Optimized PWR lattice core design (TOP core design) can be proposed [4, 5].

The TOP core design was developed as a novel re-optimization of fuel assembly (FA) and new innovative burnable concepts are adopted [4]. A soluble-boron-free SMR could achieve small reactivity swing less than 1,000 pcm, long cycle length, and high fuel burnup. In the modification of the core, the geometric parameters such as total flow area and pitch to diameter were altered. The change in core structures are expected to affect the pressure drop and critical heat flux (CHF) in the core.

Due to the elimination of the RCPs, the effects of the TOP core design is more effective to NC-SMRs than forced circulation SMR. Wijaya showed that higher reactor power can be secured by adopting the TOP core design [5]. However, the heat balance in primary system was assumed to be satisfied with the same heat transfer area of SGs despite difference power generation and the effect on the CHF in natural circulation flow was not investigated.

The objective of this study is to investigate the effects of TOP core design on power generation rate with pressure balance and heat balance, and CHF in a NC-SMR system. To evaluate the power and CHF, an algorithm which embodies heat balance equation and was developed and implemented.

2. Methodology for Steady State Assessment

2.1 Reference and TOP core designs

In this study, we investigated the effects of the TOP core design on power generation capability in terms of thermal-hydraulics and CHF by comparing it to the previous design. The reference core design was the benchmark for the previous ATOM core, which consisted of 69 FAs with a 17x17 lattice, as shown in Fig. 1. The main differences between the reference and TOP core designs were the diameters of the fuel rods, which are listed in Table 1. These changes in geometric parameters had thermal-hydraulic effects on pressure drop, mass flux, and ultimate heat transfer capability in the core region. The thermal power was calculated through numerical analysis in this study. By reducing the total pressure drop, more mass flow rate can be secured, and a larger thermal power can be secured by increasing the mass flow in the primary system of the NC-SMR system. However, determining the critical heat flux (CHF) in the core region is more complicated because many parameters can affect it.

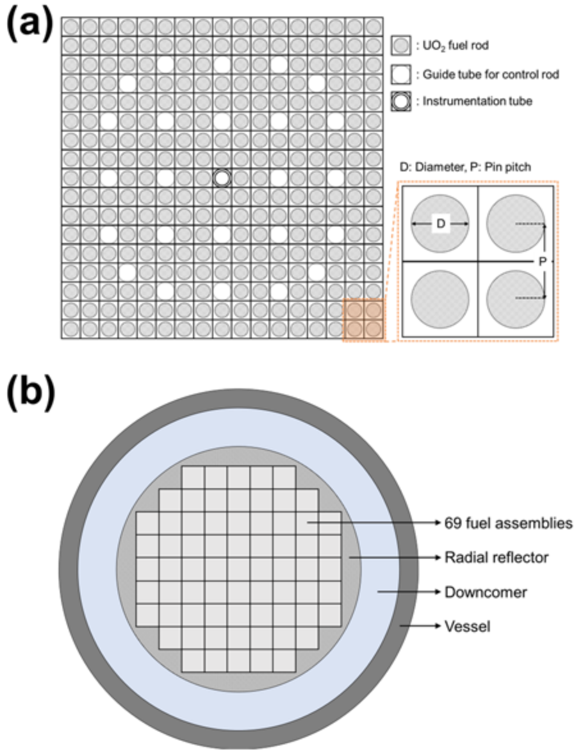

Figure 1. (a) 17×17 PWR fuel assembly, (b) radial core layout of [4, 5].

Table 1. Major parameters of NC-SMR.

Parameter	Reference core	TOP core
Thermal power	330 MWt	**calculated**
Fuel materials, enrichment	UO_2, 4.95w%	UO_2, 4.95w%
FA type, total number of FAs	17×17, 69	17×17, 69
Core height	2.40 m	2.40 m
Radial reflector	SS-304	SS-304
Diameters of pellet, cladding	8.1916 mm, 9.58 mm	**7.6000 mm, 8.9284 mm**
Diameter of control rod	12.3266 mm	12.3266 mm
Number of control rods	25/FA	25/FA
Pin pitch	12.6230 mm	12.6230 mm

In this study, to identify the position of the core, which is expected to occur the departure from nucleate boiling (DNB) the most, 10 axial nodes were adopted for core region. Table 2 shows the axial power distribution of the core. Assuming radial power flattening in NC-SMR core, the peaking factors were neglected to simplify the calculation. The power distributes more on the upper region compared to chopped cosine shape which is often used. The DNB is typically expected to occur at the upper section of the core due to the heated coolant from the lower section. Due to the biased power distribution, the coolant is expected to be less heated but the heat flux of the upper section is expected to be larger. Thus, the effect on departure from nucleate boiling ratio (DNBR), which is a measure of thermal safety margin, needs to be investigated.

Table 2. Nodalized axial core power distribution.

Axial node number	Height range	Normalized axial power
10	2.16-2.40 m	1.580
9	1.92-2.16 m	2.115
8	1.68-1.92 m	1.830
7	1.44-1.68 m	1.415
6	1.20-1.44 m	1.060
5	0.96-1.20 m	0.815
4	0.72-0.96 m	0.545
3	0.48-0.72 m	0.330
2	0.24-0.48 m	0.200
1	0-0.24 m	0.110

2.2 Steady state calculation for NC-SMR

Figure 2 presents a conceptual drawing of integral NC-SMR with operating temperatures and geometric parameters in the system. The parameters are largely related to pressure balance and heat balance under steady state. The pressure balance and heat balance in the NC-SMR system are satisfied as Eqs. (1) and (2) are the same and Eq. (3), respectively. In addition, the temperatures in primary and secondary systems can be the upper limits and lower limits to each other as shown in Ineq. (4).

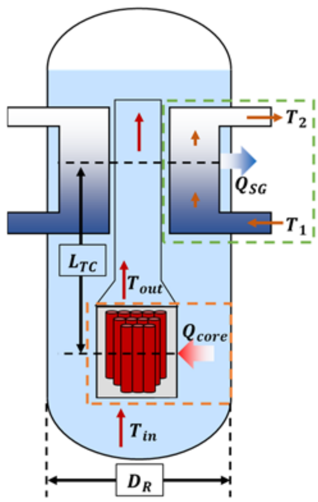

Figure 2. Conceptual drawing of integral NC-SMR.

$$\Delta P_{buo} = [\rho(T_{in}) - \rho(T_{out})]gL_{TC} \quad (1)$$

$$\Delta P_{drop} = \sum_j f_{D,j} \frac{L_j}{D_{h,j}} \frac{\rho_j v_j^2}{2} + \sum_i K_i \frac{\rho_i v_i^2}{2} \quad (2)$$

$$Q_{core} = Q_{SG} \quad (3)$$

$$T_{in} \geq T_1 \;\&\; T_{out} \geq T_2 \quad (4)$$

Here, ΔP_{buo}, ρ, g, L_{TC}, T_{in}, and T_{out} are buoyancy pressure, coolant density, gravitational acceleration, length of thermal center in the primary system, and temperatures at core inlet and outlet, respectively, in Eq. (1). ΔP_{drop}, f_D, L, D_h, v, and K are total pressure drop in the primary system, Darcy friction factor, length and hydraulic diameter of the channel, coolant velocity, and loss coefficient, respectively. The subscription i and j are component number for frictional pressure drop and form loss.

The coolant and cladding wall temperatures were determined by macro heat balance equation and Newton's cooling law as shown in Eqs. (5) and (6). The heat transfer coefficient was calculated by Dittus-Boelter correlation shown in Eq. (7).

$$Q = \dot{m}C_p(T_2 - T_1) \quad (5)$$

$$q" = h(T_{wall} - T_{bulk}) \quad (6)$$

$$h = 0.023 \frac{k}{D_h} Re^{0.8} Pr^{0.4} \quad (7)$$

Here, Q, $\dot{m}$, C_p, T_1, T_2, $q"$, h, T_{wall}, T_{bulk}, k, Re, and Pr are heat in the target channel, coolant mass flow rate, specific heat of coolant, temperature at outlet and inlet of the channel, heat flux, single-phase heat transfer coefficient, cladding wall and bulk liquid temperature, thermal conductivity of coolant, Reynolds number, and Prandtl number, respectively.

To compute pressure drop and heat transfer, empirical correlations and data were adopted for core region, spacer, and SG. In particular, frication factor correlation developed by Deissler and Taylor, frictional pressure drop correlations developed by Žukauskas, overall heat transfer coefficients in helical SGs, and CHF look-up table (LUT) were adopted to evaluate friction factor in core region, pressure drop in SG region, heat transfer in SG, and CHF [6-10]. The CHF was estimated through linear interpolation of the data in the LUT.

A calculation algorithm to evaluate the thermal-hydraulic performance and CHF of NC-SMRs was implemented. The calculation flow chart is described in Fig. 3. The operating temperature and mass flow rate at SG tubes inlet were set as 150°C and 134.1 kg/s, respectively. The calculation was iterated until the coolant temperatures at core inlet and outlet were evaluated as 252.30°C and 324.88°C, respectively. The achievable power generation rate and CHF of the NC-SMRs adopting reference and TOP core designs were compared under 15.5 MPa of operating pressure. The control variables for the assessment are summarized in Table 3.

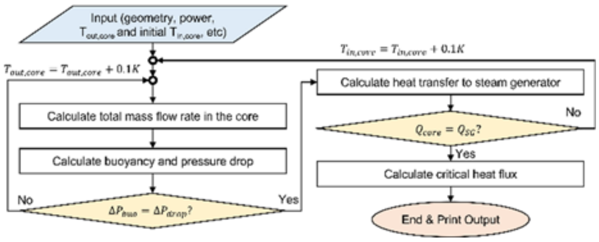

Figure 3. Calculation flow chart for steady state assessment of NC-SMR.

Table 3. Control variables for assessment.

Parameters	Values
Primary/Secondary pressures [MPa]	15.5 / 4.3
Core inlet/outlet temperature [°C]	252.40 / 324.68
SG inlet/outlet temperature [°C]	150 / 294.68
Diameter of RPV [m]	2.7

3. Results and Discussions

As shown in Table 4, smaller diameter of the fuel rods increase the flow area in core region. The pressure drop in the core region is expected to be lower due to 10.78% larger area. Accordingly, more mass flow rate was expected in the primary system. Consequently, the core heat transfer capability and thermal power increased. However, the increased mass flow caused more pressure drop in other components including SG. In addition, more thermal power requires more heat transfer of the SG. Thus, the length of SG and RPV, total and pressure drop in primary system increased with the TOP core design. Nonetheless, the results showed that the TOP core design can reduce the core pressure drop and increase power generation capcability of the NC-SMR. The power increase ratio with the TOP core was estimated as 3.18% which is very similar value with the Wijaya's evaluation [5].

Table 4. NC-SMR calculation results.

Parameter	Reference core	TOP core
Thermal power	330 MWt	**340.5 MWt**
Flow area in core	1.64 m^2	**1.82 m^2**
Mass flow rate in primary system	857.58 kg/s	**884.87 kg/s**
Mass flux in core	522.9 kg/m^2s	**486.2 kg/m^2s**
Mass flow rate in secondary system	143.41 kg/s	**147.98 kg/s**
Length of RPV	23.76 m	**24.39 m**

Heat transfer area of SG	3,427.53 m^2	**3,542.03 m^2**
Pressure drop in core region	3,278.56 Pa	**2,744.57 Pa**
Pressure drop in SG region	8,615.70 Pa	**9417.8 Pa**
Total pressure drop	12,044 Pa	**12,322 Pa**

Figure 4 shows the DNBR of the core designs. The DNBR of 7.0476, the lowest value, is certainly acceptable for safe operation. However, it is only possible in the design. In other words, the core design secured larger heat transfer area compared to other NC-SMRs.

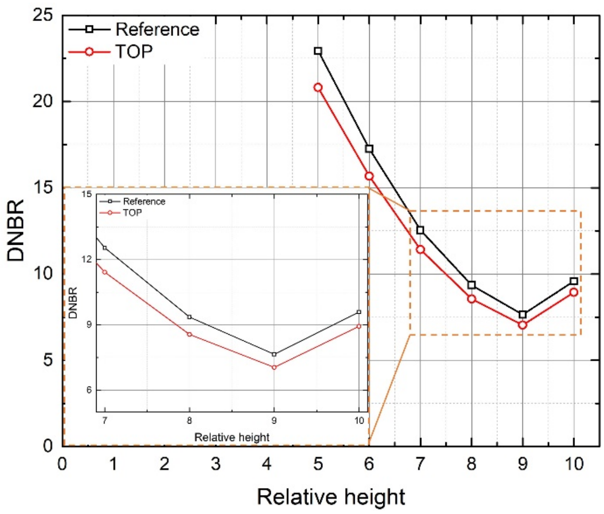

Figure 4. DNBR in reference and TOP core.

There still exist several limitations in the assessment in terms of strictness. To confirm thermal-hydraulic feasibility of TOP core design, more investigation and experimental validation are required. However, it is noted that the TOP lattice core design certainly affect the thermal-hydraulic behavior. Thus, major parameters for stable and safe operation in a NC-SMR, such as flow, power generation, and CHF need to be investigated in more details since the TOP design is advantageous in neutron economy and stable operation in soluble-boron-free core.

4. Summary and Conclusion

In this study, the effects of truly optimized PWR lattice designs on power generation rate and CHF in the natural circulation SMR was investigated. To compare the previous and modified core designs, an evaluation algorithm was developed and implemented. The major outcomes of this study can be summarized as follows:

- ✓ 3.18% more thermal/electric power generation is expected as the TOP core design is adopted only with increase of 0.63 m in the length of RPV.
- ✓ The lowest value of the DNBRs was expected to change from 7.6475 to 7.0476 as the TOP core design is adopted. Although the value is acceptable from a safety point of view, only possible in the design.

This study showed that the TOP core design is effective to secure more reactor power and the CHF was acceptable under steady state. As the future work, more detailed steady state analysis and transient analysis need to be carried out by using MARS-KS code.

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