Local Scaling Analysis for Helical Steam Generator of i-SMR Integral Effect Test Facility

Hwang Bae [a*], Jin-Hwa Yang [a], Byong Guk Jeon [a], Sang Gyun Nam [b], Byongjo Yun [b], Kyoung-Ho Kang [a]
[a] *Korea Atomic Energy Research Institute, 111, Daedeok-Daero 989Beon-Gil, Yuseong-Gu, 34057, Republic of Korea*
[b] *Pusan National University, 2, Busandaehak-ro 63Beon-Gil, Geumjeong-gu, Busan, 46241, Republic of Korea*
[*]*Corresponding author: hbae@kaeri.re.kr*

***Keywords :** scaling analysis, helical tube, steam generator, integral effect test facility, i-SMR

1. Introduction

An i-SMR [1, 2] is the innovative small modular PWR which has been newly developed by the Korean nuclear community since 2021. An independent reactor of the i-SMR with four module design produces a 170 MW electric power, and will provide the improved safety, economy, and operational flexibility. The major primary components of this typical integral reactor consist of a pressurizer, eight reactor coolant pumps (RCPs) with canned motor, a helical once-through steam generator (SG), and a core with 69 nuclear fuel assemblies and are contained in a single reactor pressure vessel (RV). The i-SMR has the three types of passive safety systems which are the passive auxiliary feedwater system (PAFS), passive containment cooling system (PCCS), and passive emergency core cooling system (PECCS). The heat exchanger of the PAFS in the emergency cooling tank (ECT) connects with the steam generator secondary side and removes the core residual heat during the transient event or loss-of-coolant accident through the natural circulation by the gravity head. The PCCS operates to constantly maintain the pressure and temperature of the CV during normal conditions and keeps to lower the pressure and temperature under an accident. The passive emergency core cooling system (PECCS) consists of two emergency depressurization valves (EDVs) and two emergency recirculation valves (ERVs). The EDVs releases the high-temperature and high-pressure reactor coolant into the CV, and the ERV recirculates the released reactor coolant to the RV. Fig. 1 shows the conceptual alignment of the i-SMR with the PCCS, PAFS, PECCS, and ECT.

To validate the performance and the safety of the i-SMR, an integral thermal-hydraulic effect test is being prepared in KAERI. The basic design for the integral effect test facility based on the scaling analysis are carried out. It is important to maintain the similarity of the thermal hydraulic phenomena in the main components and individual systems between the prototype reactor and the test facility. In this paper, the scaling methodology adopted on the integral effect test facility for the i-SMR is briefly introduced. The local scaling analysis for the steam generator is presented to concern the local phenomena such as the pressure drop and heat transfer behavior. The scale ratios for the major design parameters have been derived and will be applied to the basic design for the steam generator.

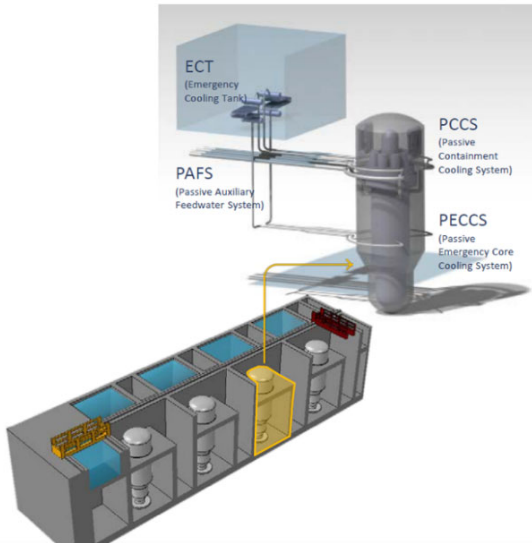

Fig. 1 Conceptual Alignment of i-SMR

2. Scaling Analysis

The basic design of the integral effect test facility for the i-SMR is based on the three-level scaling method proposed by Ishii [3]. The global scaling is applied to the major components to consist of the loop type system such as a reactor coolant system (RCS), a secondary system, and a passive safety system [4, 5, 6, 7]. The global scale ratio for the integral effect test facility is height of 1/2, diameter of 1/7, area of 1/49, and volume of 1/98. Boundary flow scaling is applied to the charging or letdown system to conserve the total mass and energy for the flow behavior to be generated by the pressure difference. Local scaling is applied to the conservation of local thermal hydraulic phenomena in a specific component such as a heat exchanger, a steam generator, and a CV.

2.1 Global Scaling

The global scaling is beginning with length scale ratio, 1/2. Assuming that the Richardson number is conserved with prototypic reactor and test facility and applying length scale to the Richardson number, the scale ratios of velocity, time, and flow rate are derived. Since the pressure and temperature are maintained identically in the prototype and the facility, the Boussinesque approximation, which considers only the density difference due to the temperature difference, can be applied to the buoyancy under single-phase normal operation conditions. Assuming that the heat source number is conserved, the scale ratios of the power/volume and core power are derived. Assuming that the friction number and modified Stanton number are conserved, the scale ratios of the friction and tube diameter are derived, respectively. Scale ratios for major design parameters are listed in Table 1.

· Axial Length Scale,

(1) $l_{oR} \equiv \frac{l_m}{l_p} = \frac{1}{2}$

· Richardson Number,

(2) $R_{oR} = \left(\frac{g\beta\Delta T_o l_o}{u_o^2}\right)_R = l_{oR} u_{oR}^{-2} = 1$

· Heat Source Number,

(3) $Q_{oR} = \left(\frac{q_s''' l_o}{\rho_s c_{ps} u_o \Delta T_o}\right)_R = q_{s,oR}''' l_{oR} u_{oR}^{-1} = 1$

· Friction Number,

(4) $F_{oR} = \left(\frac{fl}{d} + K\right)_R = \frac{f_{oR} l_{oR}}{d_{t,oR}} = 1$

· Modified Stanton Number,

(5) $St_{oR} = \left(\frac{4hl_o}{\rho_f c_{pf} u_o d}\right)_R = l_{oR}\, u_{oR}^{-1}\, d_{t,oR}^{-1} = 1$

2.2 Local Scaling for steam generator

· Pressure drop inside the tube

The similarity of the pressure drop inside the heating tube is evaluated by the local phenomenon scaling method, which selects the heating tube diameter and the helical diameter that satisfy the pressure drop scaling ratio using equation (6) proposed by Xiao [8].

(6) $\Delta P_{2\phi} = \phi_{lo}^2 f \frac{L}{d} \frac{G^2}{2\rho'}$

(7) $f = \frac{0.3164}{Re_{lo}^{0.25}}\left(1 + Re_{lo}^{0.053}\left(\frac{d}{D_c}\right)^{0.404}\right)$

(8) $\phi_{lo}^2 = (0.377 + 6.79x - 5.66x^2)A$

(9) $A = \left[1 + x\left(\frac{\mu''}{\mu'} - 1\right)\right]^{0.25}\left[1 + x\left(\frac{\rho'}{\rho''} - 1\right)\right]$

In the similarity evaluation between the model (m) and the prototype (p) using the two-phase frictional pressure drop ($\Delta P_{2\varphi}$) correlation proposed by Xiao [8], the equilibrium quality of the fluid (*x*), viscosity (μ', μ'') of saturated liquid and saturated vapor, and density (ρ') of saturated liquid have the same values in the model and prototype because the pressure and temperature are conserved in the model and the prototype. Therefore, the liquid-only two-phase frictional pressure drop multiplier (ϕ_{lo}^2) is ignored in the similarity evaluation. Only the length (*L*), mass flux (*G*), tube diameter (*d*), and helical diameter (*Dc*) are considered. The scale ratio of the two-phase frictional pressure drop ($\Delta P_{2\varphi,R}$) is arranged as in the equation (10) and the scale ratio of the frictional factor (f_R) is defined as follows.

(10) $\Delta P_{2\phi,R} = \left(f \frac{L}{d} G^2\right)_R = f_R l_R d_R^{-1} l_R \equiv l_R$

$\Rightarrow f_R = d_R l_R^{-1}$

Equation (7) for the frictional factor (f) is rearranged in terms of the ratio (f_R) of the test facility to the prototype reactor as in equation (11), which is rearranged by the ratio of the tube diameter and helical diameter of the test facility as in equation (12).

Table I: Scale Ratio for Major Design Parameters

Non-Dimensional Number	Scale Ratio	Design Parameter	Symbol	Scale Ratio	Scale Ratio for i-SMR ITL
Axial length scale	1/2	Length	l_{oR}	l_{oR}	1/2
Diameter scale	1/7	Diameter	d_{oR}	d_{oR}	1/7
Area scale	1/49	Area	a_{oR}	d_{oR}^2	1/49
Volume scale	1/98	Volume	V_{oR}	$l_{oR} d_{oR}^2$	1/98
Richardson number	1.00	velocity	u_{oR}	$l_{oR}^{1/2}$	$1/\sqrt{2}$
		time	t_{oR}	$l_{oR}^{1/2}$	$1/\sqrt{2}$
		flow rate	$\dot{m}_{oR}$	$a_{oR} l_{oR}^{1/2}$	$\frac{1}{49}\frac{1}{\sqrt{2}} = \frac{1}{69.3}$
Heat source number	1.00	Power/Volume	$q_{s,oR}'''$	$l_{oR}^{-1/2}$	$\sqrt{2}$
		Core Power	q_{oR}	$a_{oR} l_{oR}^{1/2}$	$\frac{1}{49}\frac{1}{\sqrt{2}} = \frac{1}{69.3}$
Modified Stanton number	1.00	Tube Diameter	$d_{t,oR}$	$l_{oR}^{1/2}$	$1/\sqrt{2}$
Friction number	1.00		f_{oR}	$d_{t,oR}\, l_{oR}^{-1}$	$\sqrt{2}$

$$(11)\quad f_R = \left(\frac{Re_{lo,m}}{Re_{lo,p}}\right)^{-0.25} \frac{\left(1+Re_{lo,m}^{0.053}\left(\frac{d}{D_c}\right)_m^{0.404}\right)}{\left(1+Re_{lo,p}^{0.053}\left(\frac{d}{D_c}\right)_p^{0.404}\right)} \equiv d_R l_R^{-1}$$

$$(12)\quad \left(\frac{d}{D_c}\right)_m = \left(d_R l_R^{-1}\left(\frac{Re_{lo,m}}{Re_{lo,p}}\right)^{0.197}\left(Re_{lo,p}^{-0.053} + \left(\frac{d}{D_c}\right)_p^{0.404}\right) - Re_{lo,m}^{-0.053}\right)^{1/0.404}$$

The tube thickness ratio is equivalent between the test facility and the prototype reactor when the average tube diameter ratio is conserved ($d_{t,oR} = 1$). However, when the average diameter ratio ($d_{t,oR} = l_{oR}^{1/2}$) is applied as a calculated value by the global scale analysis, the tube thickness ratio can be considered as 1 or $1/\sqrt{2}$. The helical diameter of the test facility can be determined by equation (12).

· Heat transfer coefficient inside the tube

The conservation of heat transfer phenomena inside the helical tube is confirmed by performing a local scaling analysis using Guo correlation [9] related to the heat transfer coefficient (h_{st}) for single liquid flow in helical coiled tube. Since the similarity between the nucleate boiling and the phase change into superheated steam occurring inside the tube is maintained by preserving the similarity of the heat transfer area determined by the inner diameter and length of the tube, the similarity of the heat transfer coefficient inside the heat transfer tube is sufficient to evaluate the single-phase heat transfer characteristics. Assuming that the Prandtl number (Pr) and thermal conductivity (k) are preserved in the prototypical reactor and the test facility, and applying the similarity to equation (13), the ratio of the heating tube diameter and the helical diameter of the test facility is arranged as equation (14) and (15).

$$(13)\quad h_{st} = \frac{0.21kRe^{0.8}Pr^{0.4}\left(\frac{d}{D_c}\right)^{0.1}}{2r}$$

$$(14)\quad h_{st,R} = \frac{h_{st,m}}{h_{st,p}} = \left(\frac{Re_m}{Re_p}\right)^{0.8}\left(\frac{\left(\frac{d}{D_c}\right)_m}{\left(\frac{d}{D_c}\right)_p}\right)^{0.1}\left(\frac{d_m}{d_p}\right)^{-1}$$

$$= \left(l_o^{1/2}\right)^{0.8}\left(d_{t,R}\right)^{-0.2}\left(\frac{\left(\frac{d}{D_c}\right)_m}{\left(\frac{d}{D_c}\right)_p}\right)^{0.1} = 1$$

$$(15)\quad \left(\frac{d}{D_c}\right)_m = \left(\frac{d}{D_c}\right)_p \left[\left(l_R^{1/2}\right)^{-0.8}\left(d_{t,R}\right)^{0.2}\right]^{1/0.1}$$

· Heat transfer coefficient on the primary side (outside the tube)

The conservation of the primary side heat transfer coefficient of the steam generator corresponding to the outside of the tube is confirmed by using the Zukauskas correlation [10] to determine the similarity of the horizontal/vertical pitch of the helical tubes for the staggered arrangement and the relationship between the diameter and length of the helical tubes for the in-line arrangement, respectively. Based on the Reynolds number, it is separately considered as the normal operation ($Re \geq 2 \times 10^5$) and accident operation ($10^3 \leq Re < 2 \times 10^5$) conditions, and the Prandtl number (Pr) and thermal conductivity (k) are assumed to be conserved in the prototype and the test facility. First, the Zukauskas correlation for the staggered arrangement is derived as in equation (18) and (20) according to the pitch ratio depending on the Reynolds numbers.

$$(16)\quad h = 0.031\left(\frac{S_1}{S_2}\right)^{0.2}\frac{k}{d}Re_d^{0.8}Rr_d^{0.4}\left(\frac{Pr_d}{Rr_{wall}}\right)^{0.25}$$
for $Re \geq 2 \times 10^5$

$$(17)\quad h_R = \left(\frac{S_1}{S_2}\right)_R^{0.2} d_R^{-1}u_R^{0.8}d_R^{0.8} = \left(\frac{S_1}{S_2}\right)_R^{0.2} d_R^{-0.2}l_R^{0.4}$$
$= 1$

$$(18)\quad \Rightarrow \left(\frac{S_1}{S_2}\right)_R = d_R^{0.2}l_R^{-0.4}$$

$$(19)\quad h = 0.35\left(\frac{S_1}{S_2}\right)^{0.2}\frac{k}{d}Re_d^{0.6}Rr_d^{0.36}\left(\frac{Pr_d}{Rr_{wall}}\right)^{0.25}$$
for $10^3 \leq Re < 2 \times 10^5$

$$(20)\quad \Rightarrow \left(\frac{S_1}{S_2}\right)_R = d_R^{2}l_R^{-3/2}$$

S_1: Radial (horizontal) pitch, S_2: Axial (vertical) pitch, Pr: Prandtl number, d_R: Ratio of tube diameter, k: Thermal conductivity

Second, the Zukauskas correlation for the in-line arrangement is derived as in equation (22) and (24) according to the relationship between the diameter and length of the tubes depending on the Reynolds numbers.

$$(21)\quad h = 0.033\frac{k}{d}Re_d^{0.8}Rr_d^{0.4}\left(\frac{Pr_d}{Rr_{wall}}\right)^{0.25}$$
for $Re \geq 2 \times 10^5$

$$(22)\quad h_R = d_R^{-1}u_R^{0.8}d_R^{0.8} = d_R^{-0.2}l_R^{0.4}$$

$$(23)\quad h = 0.27\frac{k}{d}Re_d^{0.63}Rr_d^{0.36}\left(\frac{Pr_d}{Rr_{wall}}\right)^{0.25}$$
for $10^3 \leq Re < 2 \times 10^5$

$$(24)\quad h_R = d_R^{-1}u_R^{0.63}d_R^{0.63} = d_R^{-0.37}l_R^{0.315}$$

· Pressure drop on the primary side (outside the tubes)

The pressure drop on the primary side (outside the tubes) of the steam generator is affected by the scale ratio of the vertical/horizontal pitch of the heat transfer tubes. Using the heat exchanger shell-side pressure drop correlation proposed in the Idelchik's Handbook [11] in which the diagram 12-27 is represented by equation (25) and the diagram 12-28 is represented by equation (32) in this paper, the pressure drop characteristics according to the pitch is evaluated. The arrangement of the helical tubes of the steam generator in the prototype reactor can be considered as a hybrid of in-line tubes and staggered tubes. First, if the pressure correlation coefficients for the in-line arrangement (diagram 12-28) are rearranged to satisfy the pressure ratio, equation (25) becomes equation (26) and (27). The temperature functions ($\Delta\zeta_t$) of the shell inlet and outlet, and the density (ρ_{av}), are the same in the prototype reactor and the model for test facility, and ψ is a constant coefficient. Equation (27) is rearranged as equation (28) and (29). When the equation is rearranged for the vertical pitch ($S_{2,m}$) in the test facility by applying the case of the range for $S_1/d \geq S_2/d$ and $1 < \bar{S}_1 \leq 8.0$, it is as equation (30) and (31).

$$(25)\quad \frac{\Delta P}{\rho_{av} u_{av}^2/2} = \psi A Re_{av}^n z_r + \Delta\zeta_t$$

$$(26)\quad \Delta P_R = \frac{\Delta P_m}{\Delta P_p} \equiv l_R$$

$$(27)\quad \Rightarrow\ \frac{(\rho_{av} u_{av}^2)_m}{(\rho_{av} u_{av}^2)_p} \frac{[\psi A Re_{av}^n z_r + \Delta\zeta_t]_m}{[\psi A Re_{av}^n z_r + \Delta\zeta_t]_p} \equiv l_R$$

$$(28)\quad \Rightarrow\ \frac{[A Re_{av}^n z_r]_m}{[A Re_{av}^n z_r]_p} = 1$$

$$(29)\quad \Rightarrow\ A_m = A_p \left(\frac{Re_m}{Re_p}\right)^{-n} \frac{(z_r)_p}{(z_r)_m}$$

Where, $A = 0.34(\bar{S}_1 - 0.94)^{-0.59}(S_1/d - 1)^{-0.5}$
for $S_1/d \geq S_2/d$ and $1 < \bar{S}_1 \leq 8.0$

$$\bar{S}_1 = \frac{S_1 - d}{S_2 - d}$$

$$n = 0.2\,(\bar{S}_1)^2$$

$$(30)\quad \Rightarrow (\bar{S}_1 - 0.94)_m^{-0.59} \left(\frac{S_1}{d} - 1\right)_m^{-0.5} = \frac{c}{(z_r)_m}$$

Where, $c = (\bar{S}_1 - 0.94)_p^{-0.59} \left(\frac{S_1}{d} - 1\right)_p^{-0.5} (l_R)^{-n} (z_r)_p$

$$(31)\quad S_{2,m} = \frac{(S_1 - d)_m}{\left(\frac{c}{(z_r)_m}\right)^{-1/0.59} \left(\frac{S_1}{d} - 1\right)_m^{-0.5/0.59} + 0.94} + d_m$$

S_1: Radial (horizontal) pitch, S_2: Axial (vertical) pitch, d: Heat tube diameter, z_r: Number of transverse rows (the number of heating tube arranged in a radial (horizontal) row)

Second, if the pressure correlation coefficients for the staggered arrangement (diagram 12-27) are rearranged to satisfy the pressure ratio, equation (32) becomes equation (33). The temperature functions ($\Delta\zeta_t$) of the shell inlet and outlet, and the density (ρ_{av}), are the same in the prototype reactor and the model for test facility, and ψ is a constant coefficient. Equation (33) is rearranged as equation (34) and (35). When the equation is rearranged for the vertical pitch ($S_{1,m}$) in the test facility by applying the case of the range for $3 < S_1/d \leq 10$ and $\bar{S}_1 > 1.7$, it is as equation (36) and (37).

$$(32)\quad \frac{\Delta P}{\rho_{av} u_{av}^2/2} = \psi A Re_{av}^{-0.27}(z_r + 1) + \Delta\zeta_t$$

$$(33)\quad \frac{(\rho_{av} u_{av}^2)_m}{(\rho_{av} u_{av}^2)_p} \frac{[\psi A Re_{av}^{-0.27}(z_r+1) + \Delta\zeta_t]_m}{[\psi A Re_{av}^{-0.27}(z_r+1) + \Delta\zeta_t]_p} \equiv l_R$$

$$(34)\quad \Rightarrow\ \frac{[A Re_{av}^{-0.27}(z_r+1)]_m}{[A Re_{av}^{-0.27}(z_r+1)]_p} = 1$$

$$(35)\quad \Rightarrow\ A_m = A_p \left(\frac{Re_m}{Re_p}\right)^{0.27} \frac{(z_r+1)_p}{(z_r+1)_m}$$

Where, $A = 1.83(S_1/d)^{-1.46}$
for $3 < S_1/d \leq 10$ and $\bar{S}_1 > 1.7$

$$(36)\quad \Rightarrow \left(\frac{S_1}{d}\right)_m^{-1.46} = \frac{e}{(z_r+1)_m}$$

Where, $e = \left(\frac{S_1}{d}\right)_p^{-1.46} (l_R)^{0.27} (z_r + 1)_p$

$$(37)\quad S_{1,m} = d_m \left(\frac{e}{(z_r+1)_m}\right)^{-1/1.46}$$

· Conduction heat transfer coefficient of tube

The conduction heat transfer coefficient for the steam generator tubes is evaluated based on the thickness and thermal conductivity.

$$(38)\quad q''_{cond} = \frac{k\Delta T}{r\ln(r_2/r_1)}$$

$$(39)\quad q''_{cond,R} = \frac{q''_{cond,m}}{q''_{cond,p}} = \frac{k_m}{k_p} \frac{r_p \ln(r_{2,p}/r_{1,p})}{r_m \ln(r_{2,m}/r_{1,m})} = 1$$

k_p: thermal conductivity of the heating tube in the prototype reactor, k_m: thermal conductivity of the heating tube in the test facility, r_1: tube inner radius, r_2: tube outer radius

· Heat transfer area

The heat transfer area ratio of the heat transfer tubes of the steam generator can be derived by evaluating the heat transfer rate and the number of heat transfer tubes. The heat transfer rate ratio must satisfy the core power ratio derived from the number of heat sources in the global scaling analysis. Assuming that the heat transfer coefficient and temperature change are conserved in the prototype and the test facility as in equation (40), the number of heat transfer tubes and the heat transfer area ratio in the test device are derived as in equation (41) and (38), respectively.

$$(40)\quad \frac{q_m}{q_p} = \frac{h_m A_{h,m} \Delta T_m}{h_p A_{h,p} \Delta T_p} = \frac{A_{h,m}}{A_{h,p}} \equiv a_R l_R^{1/2}$$

$$(41)\quad A_R = \frac{A_{h,m}}{A_{h,p}} = \frac{d_m L_m N_m}{d_p L_p N_p} = d_R l_R N_R = a_R l_R^{1/2}$$

$$(42)\quad N_R = \frac{a_R l_R^{1/2}}{d_R l_R} = \frac{a_R l_R^{1/2}}{(l_R^{1/2}) l_R} = a_R l_R^{-1}$$

The scale ratios for several design parameters were derived through the local scaling analysis and are listed in Table II. The correlation of the tube diameter and helical diameter was determined by the method to conserve the similarity related to the pressure drop and the heat transfer coefficient for the secondary tube side. The correlation of the vertical and horizontal pitch was determined by the method to conserve the similarity related to the pressure drop and the heat transfer coefficient for the primary shell side. By evaluating the correlation of thermal conductivity of the tubes, it was possible to evaluate the similarity of thermal conductivity characteristics according to the different materials. By evaluating the similarity of heat transfer area, it was possible to derive the number of the tubes.

Table II: Local Scale Ratio for Major Design Parameters

Design Parameter	Scale Ratio	Correlation for Design Parameter
Pressure drop inside the tube	$\Delta P_{2\varphi,R} = l_R$	$\left(\frac{d}{D_c}\right)_m = \left(d_R l_R^{-1}\left(\frac{Re_{lo,m}}{Re_{lo,p}}\right)^{0.197}\left(Re_{lo,p}^{-0.053} + \left(\frac{d}{D_c}\right)_p^{0.404}\right) - Re_{lo,m}^{-0.053}\right)^{1/404}$
Heat transfer coefficient inside the tube	$h_{st,R} = 1$	$\left(\frac{d}{D_c}\right)_m = \frac{d}{D_c}\left[\left(l_R^{1/2}\right)^{-0.8}\left(d_{t,R}\right)^{0.2}\right]^{1/0.1}$
Heat transfer coefficient on the primary side (outside the tube)	$h_R = 1$	for $Re \geq 2 \times 10^5$ staggered arrangement: $\left(\frac{S_1}{S_2}\right)_R = d_R^{0.2} l_R^{-0.4}$ in-line arrangement: $h_R = d_R^{-0.2} l_R^{0.4}$
		for $10^3 \leq Re < 2 \times 10^5$ staggered arrangement: $\left(\frac{S_1}{S_2}\right)_R = d_R^2 l_R^{-3/2}$ in-line arrangement: $h_R = d_R^{-0.37} l_R^{0.315}$
Pressure drop on the primary side (outside the tubes)	$\Delta P_R = l_R$	in-line arrangement: $S_{2,m} = \frac{(S_1 - d)_m}{\left(\frac{c}{(z_r)_m}\right)^{-1/0.59}\left(\frac{S_1}{d} - 1\right)_m^{-0.5/0.59} + 0.94} + d_m$
		staggered arrangement: $S_{1,m} = d_m\left(\frac{e}{(z_r+1)_m}\right)^{-1/1.46}$
Conduction heat transfer coefficient	$q''_{cond,R} = 1$	$\frac{k_m}{k_p}\frac{r_p \ln(r_{2,p}/r_{1,p})}{r_m \ln(r_{2,m}/r_{1,m})} = 1$
Heat transfer area	$A_R = a_R l_R^{1/2}$	$N_R = a_R l_R^{-1}$

Under normal operating conditions, single-phase water supplied as feedwater flows through the inside of the heat transfer tubes, undergoing phase changes into a two-phase water-steam mixture, saturated single-phase steam, and superheated steam, and is discharged to the steam pipe. The sequential and complex thermal-hydraulic phenomena occurring inside the heat transfer tubes must be designed to occur in the scaled-down test facility in the same way as in the prototypic reactor. In order to preserve these phenomena, the most important factor is to design the length and heat transfer area of the heat transfer tubes to satisfy the scale ratio. The length ratio of the heat transfer tubes was determined to be 1/2 by the global scaling analysis. The heat transfer area ratio is determined by the diameter ratio and number ratio of the heat transfer tubes. The diameter of the heat transfer tubes in the test facility must be designed in a way that preserves the similarity of the pressure drop characteristics and the heat transfer coefficient. In the local phenomenon scale analysis of a heat transfer tube, the process of determining the inner diameter and helical diameter by evaluating the similarity between the pressure drop and the heat transfer coefficient inside the heat transfer tube, and the process of determining the outer diameter and pitch by evaluating the similarity between the pressure drop and the heat transfer coefficient outside the heat transfer tube was introduced, and the number of heat transfer tubes and the heat transfer area derived from the results were introduced. If the global and local scaling analysis presented in this paper is directly applied to the design of the steam generator of the test facility, it is expected that the thermal-hydraulic phenomena occurring under normal operating conditions and accident conditions of an actual prototypic reactor will occur in the test facility as well.

3. Conclusions

The methodology of the local scaling analysis for steam generator with helical tubes was proposed based on Ishii's 3-level scaling approach. The scale ratios of the major design parameters for the helical tubes of the steam generator were derived though the way to conserve the similarity of the pressure drop and heat transfer coefficient between the inside and outside of the tube. The basic design for the integral test facility will be practically carried out using these major scaling ratios.

ACKNOWLEDGMENTS

This work was supported by the Innovative Small Modular Reactor Development Agency grant funded by the Korea Government (MSIT R&D) (No. RS-2023-00257680).

NOMENCLATURE

Abbreviation

CV Containment Vessel
i-SMR Innovative Small Modular Reactor
ECT Emergency Cooling Tank
EDV Emergency Depressurization Valve
ERV Emergency Recirculation Valve
PAFS Passive Auxiliary Feedwater System
PCCS Passive Containment Cooling System
PECCS Passive Emergency Core Cooling System

PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RV	Reactor Pressure Vessel
SG	Steam Generator

Greek Letters

β	volumetric coefficient of thermal expansion [K^{-1}]	
f	friction loss coefficient	
g	gravitational acceleration	[m^2/s]
K	form loss coefficient	
ρ	density	[kg/m^3]

Subscripts

f	saturated liquid
m	condition in model
o	reference component
p	condition in prototype
R	model to prototype quantity ratio
s	solid
t	tube
$cond$	conductivity

REFERENCES

[1] IAEA, Advances in Small Modular Reactor Technology Developments, Vienna, Austria 2022.

[2] S.Y. Kim et al., "i-SMR NSSS Design Bases," KAERI Report, 9-000-NZ-402-001 (Rev. 1), Korea Atomic Energy Research Institute, Daejeon, Republic of Korea, 2023.

[3] M. Ishii et al., The three-level scaling approach with application to the Purdue University Multi-Dimensional Integral Test Assembly (PUMA), Nuclear Engineering and Design, 186, 177–211, 1983.

[4] S. K. Moon et al., "A Sate-of-the-Art Report on the Study of the Nuclear Reactor Thermal Hydraulics Using Integral Test Facilities", KAERI Report, KAERI/AR-509/98, Korea Atomic Energy Research Institute, Daejeon, Republic of Korea, 1998

[5] K. Y. Choi et al., "Scaling Analysis Report of the ATLAS Facility", KAERI Report, KAERI/TR-5465-2014, Korea Atomic Energy Research Institute, Daejeon, Republic of Korea, 2014.

[6] J. H. Yang, B-U Bae, H. Bae, and K-H Kang, Scaling analysis and basic design of metal containment vessel in i-SMR integral effect test facility", Korean Society for Fluid Machinery Winter Meeting, Nov. 29 – Dec. 1, 2023, Jeju ICC, Korea.

[7] J. H. Yang, B-U Bae, H. Bae, and K-H Kang, Preliminary Scaling Analysis for Containment Vessel of Integral Effect Test Facility for i-SMR," ICAPP 2024, June 16-19, 2024, Las Vegas, NV, USA.

[8] Y. Xiao, Z. Hu, S. Chen, and H. Gu, Experimental study of two-phase frictional pressure drop of steam water in helically coiled tubes with small coil diameters at high pressure, Applied Thermal Engineering, 132, 18-29, 2018.

[9] L. Guo, X. Chen, C. Xu, C. Guo, and K. Lai, Forced convection boiling heat transfer and dryout characteristics in helical coiled tubes with various axial angles, Heat Transfer Conference, Proceeding of 11th IHTC, Vol.2, 261-266, 1998, Kyongju, Korea.

[10] A. Zukauskas, Heat Transfer from Tubes in Crossflow, advances in heat transfer, vol.18, pp.87-159, 1987.

[11] I. E. Idelchik, Handbook of Hydraulic Resistance, 3rd Edition, Begell House, Inc., USA, 1996.