Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 12-13, 2016

Feasibility analysis of the Primary Loop of Pool-Type Natural Circulating Nuclear Reactor Dedicated to Seawater Desalination

Woonho Jeong [a], Yong Hoon Jeong [a*]
[a]*Department of Nuclear & Quantum Engineering, KAIST, Daehak-ro, Yuseong-gu, Daejeon*
[*]*Corresponding author: jeongyh@kaist.ac.kr*

1. Introduction

Natural circulation capability of the nuclear reactor system is one of the most important factors to achieve passive safety of the system. In most cases, natural circulation was regarded as an option only for the accident situation which accompany with the shutdown of the reactor. However, with certain design condition, it is possible to operate nuclear reactor only by the natural circulation. This design increases the integrity of reactor vessel and allows fewer systems to maintain compared to the typical light water reactors which circulate reactor coolant by the reactor coolant pump.

In this study, the feasibility of natural circulation was evaluated for the reference plant AHR400 (Advanced Heating Reactor 400MWth) [1]. AHR400 is a pool-type desalination-dedicated nuclear reactor. As a consequence, AHR400 has low operating pressure and temperature which provides large safety margin.

Removal of the reactor coolant pump from the AHR400 will enforce integrity of the reactor vessel and passive safety feature. Therefore, the study also tried to find out optimized primary loop design to achieve total natural circulation of the coolant.

2. Methods

Feasibility of the primary loop of the AHR400 was analyzed by MATLAB with pressure drop model and MARS simulation. Fig. 1 shows schematics of the primary loop of the AHR400. Reactor coolant heated through the core, passes riser, and get into the hot pool. The coolant transfer heat to the secondary loop at four Intermediate Heat Exchangers (IHX) which are located in the reactor vessel. After the heat exchange, the coolant passes through the downcomer and reach to the cold pool.

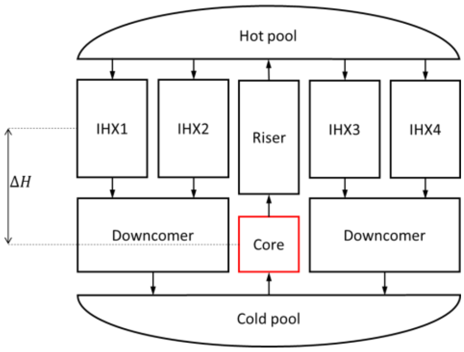

Fig. 1. Schematics of the primary loop of the AHR400

Design parameters of the AHR400 are shown on the Table 1.

Table 1: Design parameters of AHR400

Core	Type	Half length PLUS7
	Number of fuel rods	16284
	Thermal power	388.3MW
	Rod average heat flux	$419.5 kW/m^2$
	Active fuel length	1.905m
	Flow area	1.668m
IHX	Type	Single pass shell&tube heat exchanger
	Number of tubes	3598
	Number of baffles	9
	Length	9.144m
	Shell inside diameter	1.695m
	Tube outside diameter	19.05mm
PRZ	Saturation pressure	15bar
	Required volume	$35.5m^3$

2.1 MATLAB Analysis

Natural circulation feasibility of the AHR400 was checked by comparison of buoyancy pressure head and friction pressure head loss. As shown as Fig. 1, central height difference ΔH between the center of reactor core and the center of IHXs was used as a variable. Four ΔH values (10m, 15m, 20m, 25m) were selected for the calculation because the ΔH of AHR400 is 10m and bigger ΔH is demanded to enhance natural circulation. Mass flow rates through the loop and core inlet temperatures were calculated with given ΔH and core outlet temperature. Core outlet temperature was fixed as 180℃ to avoid boiling. The detailed algorithm of calculation is presented in Fig. 2.

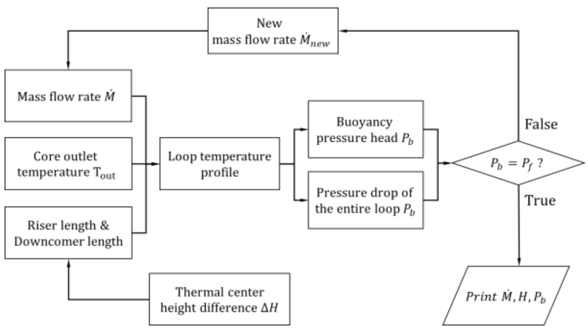

Fig. 2. Calculation algorithm of the analysis using the MATLAB

Four assumptions were introduced to simplify the calculation. These assumptions were used because they are considered as giving minor effect to the calculation results.

- 1-D flow along the loop axis
- Loop resistance at the pools was neglected
- Pressure loss from the mixing vanes were neglected
- Uniform heat flux at the core and IHX

Used correlations are as followings.

- Pressure loss at the shell&tube heat exchanger was predicted by the Bell-Delaware method. [2]
- McAdams friction factor correlation (valid for 30,000<Re<1,000,000)
- Loss through sudden contraction, $K_c = 0.42$
- Loss through sudden expansion, $K_e = 1$

Identical to the MATLAB analysis, mass flow rate through the loop and core inlet temperature were chosen as the output values.

2.2 MARS Analysis

MARS code was used to compare with the results of friction – buoyancy balancing analysis done by MATLAB. MARS code shows limitation for 3-D modeling and simulation, thus pool-type nuclear reactor AHR400 was described as loop-type in this simulation for the convenience. Flow resistance from the implemented virtual loops were neglected. Pressure losses at IHXs were implemented as form loss factors at each part for the convenience. The factors were chosen by comparing the pressure loss factors to the pressure loss predicted by the Bell-Delaware method. The other assumptions are as followings.

- 1-D flow along the loop axis
- Pressure loss from the mixing vanes were neglected
- Uniform heat flux at the core and IHX

Used correlations is as following.

- Zigrang-Sylvester approximation of the Colebrook-White correlation (valid for 3,000<Re)
- Loss through sudden contraction, $K_c = 0.42$
- Loss through sudden expansion, $K_e = 1$

3. Results and Discussions

Feasibility analysis results are shown on the Table 2 and 3. Mass flow rate and core inlet temperature are listed through various central height difference. Reference case of AHR400 which has reactor coolant pump in the loop, is shown on the top of the table to conveniently observe the difference. The results of MATLAB analysis derived by balancing pressure loss through the entire loop and buoyancy driven pressure are shown on the Table 2. The results of MARS simulation are also shown on the Table 2.

Table 2: Mass flow rate, core inlet temperature driven by MATLAB and MARS analysis

ΔH[m]	Mass flow rate [kg/s]		Core inlet T [℃]	
	MATLAB	MARS	MATLAB	MARS
AHR400	3063	3063	150	150
10	1306	1252	112	110
15	1510	1464	122	120
20	1711	1641	129	126
25	1867	1788	133	131

Results from the MARS simulation shows conservative results than the results from the MATLAB analysis. Therefore, discussions will be based on the MARS simulation results. At all cases, core inlet temperatures are out of the constraints because the lowest temperature at the secondary loop (135℃) is higher than the core inlet temperature. Therefore, without any design modification, AHR400 is not feasible to be operated only by the natural circulation of the coolant.

In order to found out the possible modification of AHR400 to increases natural circulation capability, mass flow rate and core inlet temperature were calculated at the 200MWth core power which is about half of the core power of the AHR400.

Table 3: Mass flow rate, core inlet temperature driven by MARS simulation with 200MWth and 400MWth core power

ΔH[m]	Mass flow rate [kg/s]		Core inlet T [℃]	
	200MW	400MW	200MW	400MW
10	1090	1252	140	110
15	1276	1464	146	120
20	1414	1641	149	126
25	1525	1788	151	131

With 200MWth power no case violate the core inlet temperature constraint. However, when ΔH = 10, required heat transfer area of a IHX is $9200m^2$ which is bigger than the $8900m^2$ heat transfer area of reference case. Therefore, central height difference of 15m will be the optimum height which comply with design constraints.

4. Conclusions

Natural circulation capacity of the primary loop of the desalination dedicated nuclear reactor AHR400 was evaluated. It was concluded that to remove RCP from the AHR400 and operates the reactor only by natural circulation of the coolant is impossible. Decreased core power as half make removal of RCP possible with 15m central height difference between the core and IHXs. Furthermore, validation and modification of pressure loss coefficients by small-scaled natural circulation experiment at a pool-type reactor would provide more accurate results.

REFERENCES

[1] Y. H. Jung, Y. H. Jeong, J. Y. Choi, A. F. Wibisono, J. I. Lee, and H. C. No, Feasibility study of a small-sized nuclear heat-only plant dedicated to desalination in the UAE, Desalination, Vol.337, p. 83, 2014.

[2] M. Sema, A. Jimenez, A Compact Formulation of the Bell-Delaware Method for Heat Exchanger Design and Optimization, Chemical Engineering Research and Design, Vol.83, p.539, 2005.