Preliminary Computational Fluid Dynamics Analyses of 2-loop RVI Baffle Structure

Hyun-Gon Kim[a], Jae-min Jyung[a], and Yoon-Suk Chang[a*]
[a]*Department of Nuclear Engineering, Kyung Hee University, 1732 Deogyeong-daero, Giheung-gu, Yongin-si, Gyeonggi-do 17104, Republic of Korea*
[*]*Corresponding author: yschang@khu.ac.kr*

1. Introduction

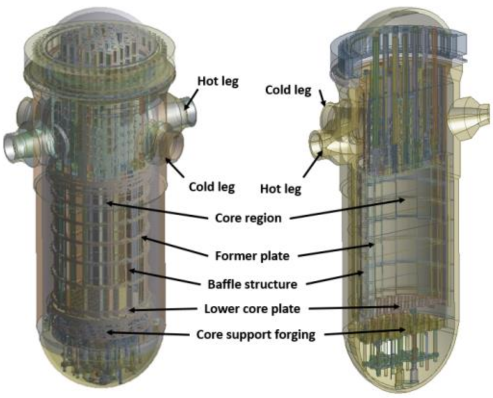
The RVIs (Reactor Vessels Internals) baffle structure surrounds the core region where high neutron flux and temperature are induced by nuclear fuel assemblies. In particular, BFBs (Baffle Former Bolts) are jointing baffle and former plates of the baffle structure. BFB related studies have been conducted since the first BFB cracking had been found in a French plant in late 1980s.

BFBs are located in core region. The flow of fluid in this region is complex and can affects BFBs. If these bolts degrade in this environment for long cycles, the deformation of fuel cladding could be occurred [1]. The structural integrity should be performed considering the fluid characteristics. In the previous study, velocity distribution depending on spacer grid length were compared by using different turbulence models [2, 3].

In this study, preliminary CFD (Computational Fluid Dynamics) analyses of baffle structures were carried out. Among several types of reactors, the Westinghouse 2-loop reactor was adopted. The pressure and velocity distributions were derived according to turbulence models. As results, the parameters were compared and vulnerable region was discussed.

2. Analysis Model and Conditions

Fig.1 presents the schematics of Westinghouse 2-loop reactor. Core support forging is the passage through which the fluid moved to the bottom of the reactor starts moving upward. Lower core plate is the passage where the core region is beginning. Fig. 2 shows fluid model description in height. Analysis conditions are presented in Table 1.



(a) Real arrangement (b) CFD model

Fig. 1. Schematics of Westinghouse 2-loop reactor

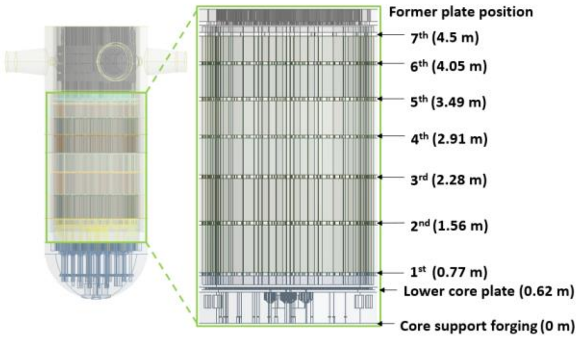


Fig. 2. Positions of baffle structure

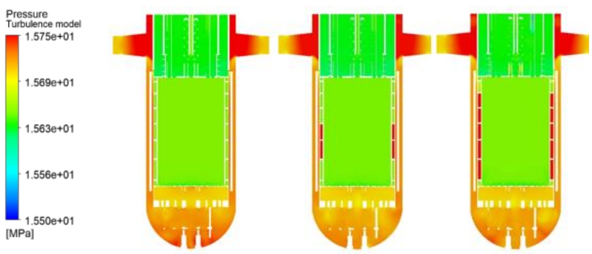
Table 1. Analysis conditions	
Analysis type	Steady state
Fluid temperature	274 ℃
Inlet mass flow rate	5600 kg/s
Outlet pressure	15.5 MPa

3. Analysis Results

As mentioned earlier, the fluid velocity and pressure can act as factors to judge the vulnerable area in the baffle structure. Accordingly, the fluid pressure and velocity profiles were examined.

3.1 Pressure Distributions

Fig. 3 and Fig.4 shows the pressure distributions of inlet and outlet section of the reactor according to three different turbulence models. The pressure of the core region was about 15.6 MPa. There was difference in bypass region depending on the turbulence model. It is expected that mesh optimization of this region will be required to clarify the pressure distribution in this region.



k-ε model k-ω model SST model

Fig. 3. The pressure distributions of inlet section

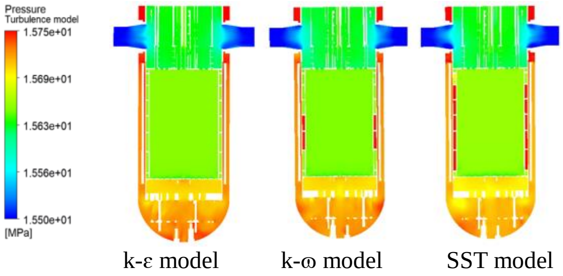


Fig. 4. The pressure distributions of outlet section

3.2 Velocity Distributions

Fig. 5 and Fig. 6 shows the fluid velocity distributions of inlet and outlet section of the reactor according to three different turbulence models. The fluid velocity distribution looked complex in the lower part of the core region. Therefore, velocity profile was extracted from the nodes along the height of each case. The fluid velocity distributions of the inlet section were different from those of the outlet section.

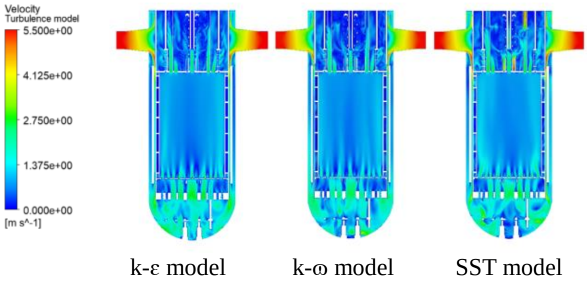


Fig. 5. The velocity distributions of inlet section

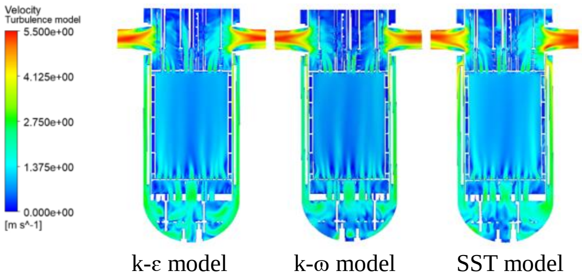


Fig. 6. The velocity distributions of outlet section

3.3 Vulnerable Region

Fig. 7 presents velocity profiles of the core region with three different turbulence models. Dispersions of the k-ε model, k-ω model and SST model were 0.966, 0.877, 0.884, respectively. Among seven former plates, the fluid velocity near the first one was the highest that has about 5.5 m/s.

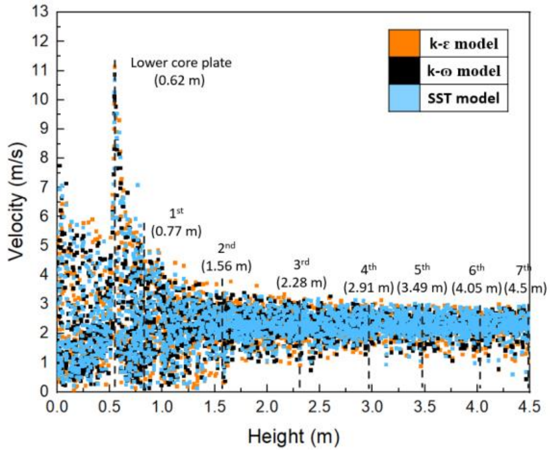


Fig. 7. Velocity profiles of the core region with different turbulence models.

4. Conclusions

In this study, fluid flow characteristics were compared by using different turbulent model. The followings were observed as the key findings.

(1) The fluid pressure distribution was similar except bypass region in three turbulence models.
(2) The fluid velocity distributions of the inlet section were different from those of the outlet section.
(3) Dispersion of velocity profile with k-ω model was smaller than other models. And the most vulnerable region in core region was evaluated near the first plate because of the high flow.
(4) Further grid optimization and structural integrity of BFBs will be carried out.

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REFERENCES

[1] U.S. NRC, Degradation of baffle-fromer bolts in pressurized-water reactors, ML16225A341, 2016.
[2] Yingjie Wang, Mingjun Wang, Haoran Ju, Minfu Zhao, Dalin Zhang, Wenxi Tian, Tiancai Liu, Suizheng Qiu, G.H. Su, CFD simulation of flow and heat transfer characteristics in a 5×5 fuel rod bundles with spacer girds of advanced PWR, Nuclear Engineering and Technology, available online, 2019.
[3] M. Zaidabadi, G.R. Ansarifar, M.H. Esteki, Thermal hydraulic analysis of VVER-1000 nuclear reactor with dual-cooled annular fuel using k-ω SST Turbulence model, Annals of Nuclear Energy 101, 118-127, 2017.