Study on Thermal-Hydraulic Characteristics of Supercritical Water Reactor

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As the only water-cooled reactor among the six under the Generation-IV program, supercritical water-cooled reactor (SCWR) takes up attentions extensively all over the world. However, due to a rapid variation of the thermal-physical properties near the pseudo-critical line, the thermal-hydraulics characteristics of water at supercritical conditions differ obviously from that at sub-critical conditions of PWR and BWR. Simultaneously, because of high temperature, avoiding excessive hot spots of local cladding temperature becomes a big challenge in SCWR [1]. Therefore, sub-channel analysis is of crucial importance in designing fuel assemblies.

Based on the PWR subchannel analysis code of SNERDI (Shanghai Nuclear Engineering Research and Design Institute), a subchannel code for SCWR was developed. Using the developed subchannel code, the thermal-hydraulic characteristics of the typical SCWR fuel assembly with the moderator water rod, including the temperature, mass flux, fuel rod cladding temperature, heat transfer coefficients and so on, was investigated.

Typical outlet temperature distribution for the single enrichment with 50% flow through the moderator water rods is illustrated in Figure 1. The temperature profile closely follows the assembly power distribution. The highest outlet temperature appears in channels 2 and 14, which are located near the center of the fuel assembly. As for the uniform power distribution, the temperature distribution, which the highest outlet temperature is at the out corner of the fuel assembly, is more uniform than the single enrichment case.

Figure 2 shows the averaged coolant temperature and density profiles along the flow direction for different percents of flow through moderator water rods. The percents of flow through moderator water rods were varied from 0% to 75%. The results show that along the axial of the core, the coolant temperature firstly increases rapidly, forms a plateau because of very high thermal capacity near the pseudo-critical line, and then increases again rapidly with low thermal capacity of water. But the coolant density decrease sharply in the vicinity of the pseudo-critical line. For different percents of flow through the moderator water rods, the heat transfer from coolant channels to moderator water rods is different, which further induce the different inlet temperatures of coolant channels. With an increase of the percent of flow through moderator water rods, the coolant temperature rise and the density decrease obviously.

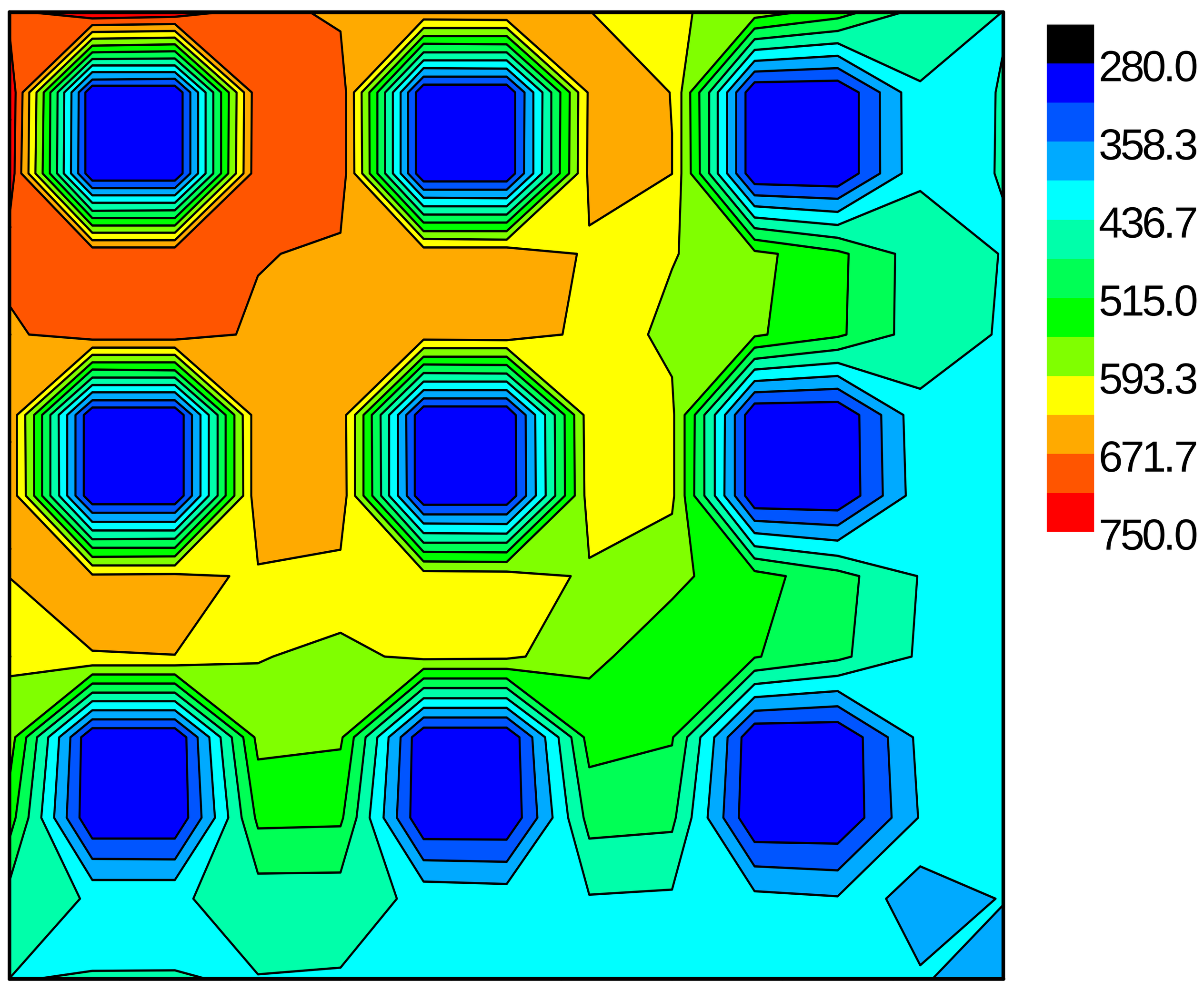

Figure 1 Typical outlet temperature distribution of 1/4 fuel assembly for single enrichment with 50% flow through moderator water rods.

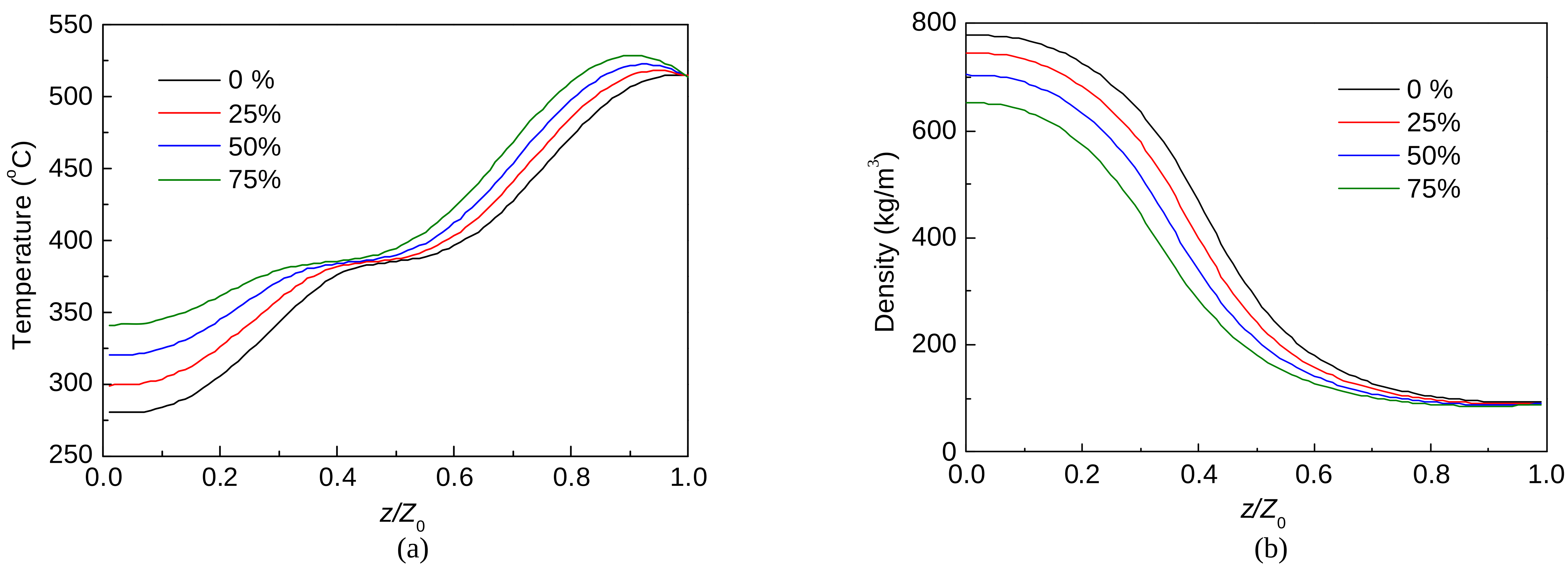

Figure 2 Typical thermal-hydraulic characteristic profiles for different percent of flow through moderator water rods: (a) coolant temperature; (b) coolant density.

Reference

1. Zhao D. J., Liao C. K., Shi G. B., Review of Sub-Channel Study and Coupled Neutronics/Thermal-hydraulic Study for Supercritical Water-Cooled Reactor, Nuclear Power Engineering and Technology, 3(2008), 34-41.

2. Bishop A. A., Sandberg R. O., Tong L. S., Forced Convection Heat Transfer at High Pressure After The Critical Heat Flux, ASME 65-HT-31, 1965.