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Augmentation of single-phase forced convection heat transfer in tightly arrayed rod bundle with twist-vane spacer grid

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ABSTRACT

The investigation on the augmentation of convection heat transfer of water rod bundle flow was performed experimentally. The 4×4 tightly arrayed rod bundle with narrow gap was tested and its P/D (Pitch-to-Diameter ratio) was $\sim$1.08. As the spacer grid, the twist-vane spacer grid was used. The axial mean bundle-flow velocity was $U \sim 1.0$ m/s, and the corresponding Reynolds number was $\sim$10,400. Circumferential and axial distributions of rod-wall temperatures were measured. Upstream of twist-vane spacer grid, the rod-wall temperature at the subchannel center was measured to be lower than that at the rod-gap center. Downstream of twist-vane spacer grid, the rod-wall temperature toward the tip of twist-vane was lowest, which might be due to the deflected flow caused by twist-vane. The twist-vane spacer grid enhanced significantly the convection heat transfer in the tight-lattice rod bundle, and its maximum enhancement value was $\sim$77% near the downstream of spacer grid. Based on the present experimental data, the previous correlations of Nusselt number enhancement were compared and assessed. Then, the enhancement correlation of convection heat transfer for the twist-vane spacer grid in the tightly arrayed rod bundle was proposed.

1. Introduction

In the nuclear fuel assembly for the PWR (Pressurized Water Reactor), a number of fuel rods are arrayed to a square or triangular rod bundle configuration, and are supported by the spacer grid. The coolant water flows parallel with the nuclear fuel rods through the non-circular subchannels formed between the nuclear fuel rods. To improve the heat transfer of rod bundle flow and mixing rate between the adjacent subchannels, the spacer grid generally adopts the mixing-vane. On the other hand, in general, the mixing-vane spacer grid is known to increase the pressure drop. Therefore, the final goal of mixing-vane spacer grid may be to achieve the smaller pressure drop and higher heat transfer performance. The mixing-vane spacer grid induces the complicated coolant flow pattern and distribution in the rod bundle flow, which has a significant impact on the thermal–hydraulic characteristics of coolant flow around the nuclear fuel assembly. In the nuclear power industry, an understanding of the heat transfer performance downstream of the various kinds of mixing-vane spacer grid can be one of important research subjects to ensure the safety operation and high performance of the nuclear reactor [1].

Several mechanisms of the heat transfer enhancement near the spacer grid in rod bundle flow of nuclear reactor have been reported, which are summarized in the previous report [2]: The spacer grid affects the thermal and momentum boundary layers. The spacer grid contributes to breaking up the local boundary layer, which results in increasing the local heat transfer. In addition, since the spacer grid obstructs the coolant flow, the flow velocity at the spacer grid location becomes higher than that at the unobstructed portion. Hence, the heat transfer coefficient at the spacer grid would tend to become higher. The spacer grid may also act as fin structure contacting the heated surface, and consequently, can increase the local heat transfer. If the spacer grid surface is much cooler than the nuclear fuel surface, the radiation may have influence on the local heat transfer. Also, the spacer grid provides the sudden contraction and expansion within the bundle configuration, which can change the local heat transfer due to eddy generation. The swirl and turbulence generation caused by the spacer grid contributes to improving the heat transfer performance of rod bundle flow.

Most of researchers have made efforts to invent and propose the uniquely designed mixing-vane to increase significantly the

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Nomenclature

A	area (m^2)
D	diameter (m)
F	fraction of the projected area of the vanes to the flow cross-section (–)
h	convection heat transfer coefficient (W/m^2 K)
k	thermal conductivity (W/m K)
L	length (m)
Nu	Nusselt number (–)
P	pitch (m)
p	wetted perimeter (m)
q''	heat flux (W/m^2)
P/D	pitch-to-diameter ratio (=pitch between rods/rod diameter) (–)
Re	Reynolds number (–)
T	temperature (°C)
ΔT	temperature difference (°C)
t	thickness (m)
U	axial mean bundle-flow velocity (m/s)
z	axial position (m)

Greek letters

ε	blockage ratio of spacer grid (=A_p/A_f) (–)
$\varnothing$	vane angle with respect to the axial direction (°)

Subscripts

exp	experimental data
f	undisturbed flow cross-sectional area in a bare rod bundle
h	hydraulic diameter
in	inlet
o	outer diameter
p	projected cross-sectional area of the spacer grid
pre	prediction
sub	subchannel
wall	rod-wall
wetted	wetted perimeter

convection heat transfer performance with a minimal increase of pressure drop. Some investigations on the effect of spacer grid on the convection heat transfer and flow mixing between subchannels have been performed experimentally [3–9] and numerically [1,10–13] using the various types of spacer grids. The detailed flow behavior and heat transfer characteristics induced by the mixing-vane spacer grid are known to be not easily predictable due to their complexities. Therefore, to apply newly designed mixing-vane spacer grid (e.g., twist-vane spacer grid used in this work) to the nuclear fuel assembly, the experiments on its heat transfer performance need to be carried out. Also, such experimental data can be used for validation and verification of the CFD (Computational Fluid Dynamics) code.

On the other hand, some researchers have tried to propose the prediction correlation of convection heat transfer performance of mixing-vane spacer grid. Yao et al. [14] proposed the prediction correlation of Eq. (1) for the heat transfer enhancement of swirling-vane spacer grid, based on the experimental data. They considered the effect of blockage ratio and mixing-vane angle on the heat transfer enhancement (Nu/Nu_{in}) in the rod bundle flow.

$$\frac{Nu}{Nu_{in}} = \left[1 + 5.55\varepsilon^2 e^{-0.13(z/D_h)}\right] \cdot \left[1 + F^2 \tan^2 \varnothing e^{-0.034(z/D_h)}\right] \quad (1)$$

Here, ε and $\varnothing$ are the blockage ratio of the spacer grid and angle of the mixing-vane, respectively. F is the fraction of the projected area of the vanes to the flow cross-section. z and D_h indicates the axial distance from the spacer grid and hydraulic diameter, respectively. On the other hand, Miller et al. [15] proposed a single-phase convection heat transfer augmentation correlation of Eq. (2) based on the steam cooling experimental data. They reported that the heat transfer enhancement caused by the mixing-vane spacer grid is influenced by the Reynolds number and spacer grid blockage ratio.

$$\frac{Nu}{Nu_{in}} = 1 + 465.4(Re)^{-0.50}\varepsilon^2 e^{-7.31\times10^{-6}Re^{1.15}(z/D_h)} \quad (2)$$

The correlation of convection heat transfer enhancement is known to strongly depend on the type of mixing-vane spacer grid. Hence, the application of the previous correlations to the newly designed mixing-vane spacer grid can be limited, which should be examined.

Among various types of mixing-vane spacer grids, the split-vane spacer grid [16] is widely used in the nuclear fuel assembly for PWR, which is known to increase the crossflow mixing between the neighboring subchannels. Our research group has proposed the twist-vane spacer grid [17], which is uniquely designed to increase a crossflow mixing between adjacent subchannels as well as a swirl mixing within the subchannel. Recently, our group performed the experiments of flow mixing and convection heat transfer to check the performance of the twist-vane spacer grid. In et al. [18] measured the axial and lateral velocities downstream of the twist-vane spacer grid in a simulated non-heated 4 × 4 rod bundle using LDV (Laser Doppler Velocimetry) and PIV (Particle Image Velocimetry) techniques. In et al. [19] checked the convection heat transfer performance of twist-vane spacer grid in the regular rod bundle configuration of P/D (=pitch between rods/rod diameter) ~ 1.35 using the partially heated single rod. The ~30%-enhancement of convection heat transfer was measured. However, in the different rod bundle geometry (e.g., different P/D condition), the convection heat transfer performance of twist-vane spacer grid may be altered.

KAERI (Korea Atomic Energy Research Institute) has been developing a dual-cooled annular fuel to increase a significant amount of the reactor power in PWR [20]. The dual-cooled annular fuel array was proposed to become structurally compatible with the conventional cylindrical solid fuel array under the same array size and guide tube locations. The conventional cylindrical solid fuel is cooled by the water coolant of the outer channel. On the other hand, the dual-cooled annular fuel allows the water coolant to flow through the inner channel as well as the outer channel. Hence, the dual-cooled annular fuel exhibits the larger outer diameter, and smaller P/D in the fuel assembly, as compared with the conventional cylindrical solid fuel and assembly, respectively. The difference of P/D can significantly affect the thermal–hydraulic characteristics of rod bundle flow with mixing-vane spacer grid. To apply the twist-vane spacer grid to the dual-cooled annular fuel assembly, its heat transfer performance should be tested under the tightly arrayed rod bundle configuration.

In this work, the convection heat transfer experiments of water flow in the tightly arrayed rod bundle with the twist-vane spacer grid were conducted. The P/D of test bundle prepared is ~1.08, which is simulated to the dual-cooled annular fuel assembly. The averaged axial bundle-flow velocity is $U \sim 1.0$ m/s, which corresponds to $Re \sim 10{,}400$. To examine the convection heat transfer performance, the circumferential and axial rod-wall temperatures are measured. In addition, the variations of Nusselt number along