Implementation of Helical Tube Heat Transfer Correlations in TRACE for Sodium-cooled Fast Reactor Analysis

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1. Introduction

TRACE code is a thermal-hydraulics code designed to analyze LOCAs and system transients in LWR. In order to extend the capabilities of TRACE to the generation IV reactors, many models have been developed and adapted recently. Until now, however, there are no proper heat transfer correlations for helical tube. Therefore, in this study, some helical tube heat transfer correlations are evaluated and adapted to TRACE code especially for analyzing steam generator of sodium-cooled fast reactor (SFR). SFR has helical type steam generator that the hot liquid sodium flows in the shell side (primary side) and the cold water flows in the tube side (secondary side). As a preliminary analysis, code calculations for DSFR's[1] steam generator are performed with modified TRACE code.

2. Implementation of Helical Tube Heat Transfer Models

2.1 Tube Side

The phase change occurs during the water flowing and the flow characteristics vary with the phase in the tube side. In the two phase region, TRACE code selects different heat transfer correlation for calculation according to three boiling regimes; nucleate, transition, film boiling. The quality value 0.8 is determined as a dry-out correlation in the helical tube. Since the overall dry-out quality in the helical tube is higher than straight tube, the heat transfer lengths of transition and film boiling region are relatively short. So the effects on heat transfer in that regions are insignificant compared with nucleate boiling region. Therefore, Mori-Nakayama's correlation[2] is employed instead of Gnielinski's for heat transfer correlation in nucleate boiling region of TRACE code whereas transition and film boiling region remain as it was. Mori and Nakayama suggested the heat transfer correlation based on experimental data of helical tube and Gnielinski's was based on straight tube. Equation 1 is adapted to source code of TRACE for nucleate boiling heat transfer in tube side.

$$h_c = 0.02439\ \frac{k_f}{d}\mathrm{Re}_f^{0.8333}\,\mathrm{Pr}_f^{0.4}\left(\frac{d}{D}\right)^{\frac{1}{12}}\left[1+\frac{0.061}{\left[\mathrm{Re}_f\left(\frac{d}{D}\right)^{2.5}\right]^{\frac{1}{6}}}\right]$$

Eq. 1. Mori-Nakayama's correlation

2.2 Shell Side

The liquid sodium flow perpendicular to the tube array causes the cross-flow in the shell side. This cross-flow increases the heat transfer to coolant. Ishiguro's correlation[3] is adapted instead of Subbotin to consider this phenomenon. Ishiguro performed an experimental study to clarify the characteristics of liquid sodium flowing uniformly across a circular cylinder and suggested the heat transfer correlation. Equation 2 is adapted to TRACE for heat transfer in shell side.

$$h = 1.125\ \frac{k}{d}(\mathrm{RePr})^{0.413}$$

Eq. 2. Ishiguro's correlation

3. Modeling For Steam Generator of DSFR

3.1 SNAP

The symbolic nuclear analysis package(SNAP) is a standard graphical user environment designed to simplify the use of analytical codes. It has highly extensible and flexible and can be adapted to any engineering codes such as TRACE, RELAP5, CONTAIN, etc. The modified TRACE model is created with SNAP version 2.2.2.

3.2 Model Description

The steam generator of DSFR is a vertically oriented and shell-and-tube type heat exchanger with fixed tube-sheet. In the TRACE model, the shell tube for the primary side is modeled by PIPE component having 25 cells. The helical tube for secondary side is simplified to angular tube and modeled by PIPE component having 29 cells. Primary and secondary sides are connected by heat structures. Sodium flow into primary side and water flow into secondary side are modeled by FILL and BREAK component. Feedwater mass flow is determined by using the control system options of SNAP. The boundary conditions and the nodding diagram are shown below.

Table 1: Boundary conditions

Sodium inlet temperature	502℃
Sodium flow rate	3073 kg/s
Feedwater temperature	215℃
Feedwater flow rate	344.7 kg/s

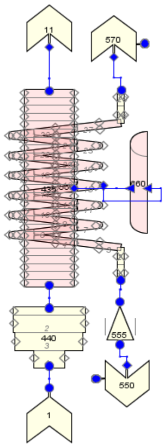

Fig. 1. Nodding diagram of DSFR's steam generator

4. Results and Conclusions

The code calculations for helical tube of DSFR's steam generator are performed by using both original TRACE(original model) and modified TRACE(helical model). Figure 2 shows the comparison results of heat transfer coefficient.

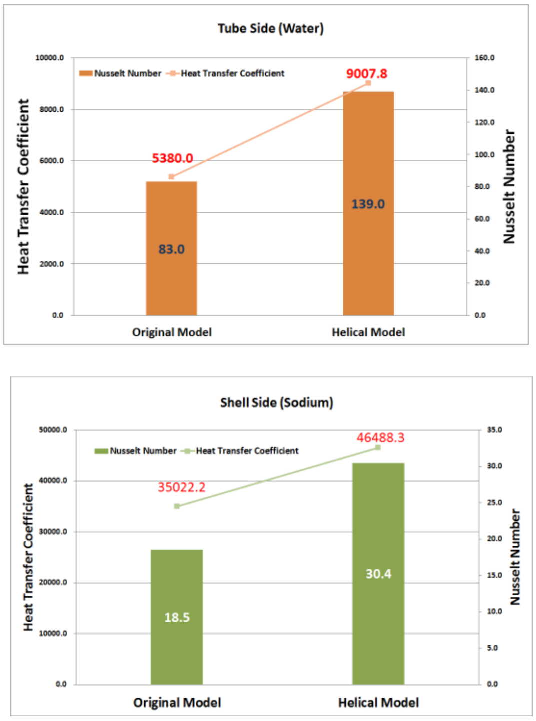

Fig. 2. Comparison results of heat transfer coefficient

When using the modified TRACE, the heat transfer rate is enhanced both tube side and shell side. Compared to the original model, the heat transfer coefficient is increased by about 67% in the tube side and 32% in the shell side. The improvement originates from implementation of proper heat transfer correlation for helical tube. In the helical tube, because of the centrifugal force of the fluid flow, the liquid and steam are separated. Then, the secondary flow is produced by steam at the center of the tube and it distributes the liquid to the outer surface of the tube internal wall. As a result, the dry out occurs from the inner side(to outer side of tube as the quality is increased. heat transfer is increased considerably compared to the straight tube. As a preliminary analysis, DSFR's SG performance with modified TRACE code is calculated and the result shows this feature very well without adjustment of heated diameter. Figure 2 shows the change of heat transfer coefficient in the tube side and shell side of helical steam generator of DSFR. Compared to the original model, the heat transfer coefficient is increased by about 67% in the tube side and 32% in the shell side. The reactor power properly reflects this increased heat transfer without modification of heated diameter, so the reliability of steady and transient analysis of DSFR can be improved. The modified TRACE code can also be used for another type of heat exchangers which has helical tube in the future.

NOMENCLATURE

d = inner diameter of tube
D = diameter of helical coil
k = thermal conductivity
Re = Reynolds number
Pr = Prandtl number

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