Influence of single-phase heat transfer correlations on safety analysis of research reactors with narrow rectangular fuel channels

A. Rawashdeh[1], B. Lee[2]*, Y.J. Chung[2], S. Park[2], R. Altamimi[1]

[1]Advanced Nuclear Engineering System Department, University of Science and Technology, 217 Gajeong-Ro Yuseong-Gu, Daejeon, 305-350, Republic of Korea

[2]Korea Atomic Energy Research Institute, 1045 Daeduk-Daero, Dukjin-Dong, Yuseong-Gu, Daejeon, 305-353, Republic of Korea

*Corresponding author: leebh@kaeri.re.kr

ABSTRACT

The influence of different single-phase heat transfer correlations on the fuel temperature and minimum critical heat flux ratio (MCHFR) during a typical accident of a 5 MW research reactor is investigated. A reactor uses plate type fuel, of which the cooling channels have a narrow rectangular shape. RELAP5/MOD3.3 tends to over-predict the Nusselt number (Nu) at a low Reynolds number (Re) region, and therefore the correlation set is modified to properly describe the thermal behavior at that region. To demonstrate the effect of Nu at a low-Re region on an accident analysis, a two-pump failure accident was chosen as a sample problem. In the accident, the downward core flow decreases by a pump coast-down, and then reverses upward by natural convection. During the pump coast-down and flow reversal, the flow undergoes a laminar flow regime which has a different Nu with respect to the correlation sets. Compared to the results by the original RELAP5/MOD3.3, the modified correlation set predicts the fuel temperature to be a little higher than the original value, and the MCHFR to be a little lower than the original value. Although the modified correlation set predicts the fuel temperature and the MCHFR to be less conservative than those calculated from the original correlation of RELAP5/MOD3.3, the maximum fuel temperature and the MCHFR still satisfy the safety acceptance criteria.

1. Introduction

Research reactors have several different features from those of nuclear power plants. Many research reactors use a plate type fuel for a high neutron flux. Their cooling channels are a narrow rectangular shape with a closed channel rather than a conventional circular shape with an open channel. In addition, the coolant is mostly single phase water since the reactors are usually operated under low-temperature atmospheric conditions. Although the correlations developed for a circular channel have been used for various channel shapes using the equivalent diameter of the channel, many researchers have studied the heat transfer characteristics of rectangular channels [1-3].

RELAP5/MOD3.3 is one of the system analysis codes for simulating a reactor accident, and has a correlation set for analyzing the thermal-hydraulic phenomena in the reactors. The code also contains a correlation set for a research reactor having a narrow rectangular channel [4]. However, the compatibility of the correlation set to the reactor channel should be investigated and guaranteed before using the code for a safety analysis of a research reactor.

In this paper, the suitability of the correlation set of the RELAP5/MOD3.3 code on narrow rectangular cooling channels is investigated for a new 5MW pool-type research reactor. The incompatibility of the code is then modified to be suitable compared to existing heat transfer experimental results for a narrow rectangular channel. A typical loss-of-flow accident of the reactor is simulated with the original and modified codes, and the maximum fuel temperatures and CHFRs are compared.

2. Comparison of RELAP5/MOD3.3 and Experiments

The REALP5/MOD3.3 code has a heat transfer correlation set for circular channels, i.e., the 101 correlation set, and a set for narrow rectangular channels, i.e., the 102 set. These correlation sets have differences in natural convection heat transfer model, single-phase forced convection heat transfer model, and critical heat flux (CHF) model. The 102 correlation set was implemented for an advanced neutron source (ANS) reactor design. The ANS core design has a narrow rectangular channel with an aspect ratio of 68.11.

The 101 correlation set includes the Churchill-Chu natural convection correlation developed at a vertical wall, whereas the 102 correlation set includes the Elenbass correlation developed for a vertical channel. For laminar forced convection, the 101 correlation uses Nu=4.36, which is the analytical value for a circular channel, and the 102 correlation set uses Nu=7.63 for the ANS channel geometry.

For turbulent forced convection, the 101 correlation set uses the Dittus-Boelter correlation composed of Re and Prandtl number (Pr) as

$$Nu = 0.023\,\mathrm{Re}^{0.8}\,\mathrm{Pr}^{0.4}, \tag{1}$$

and 102 correlation set uses the Petukhov correlation as

$$Nu = \frac{\left(\frac{f}{8}\times \mathrm{Re}\times \mathrm{Pr}\right)\times\left(\frac{\mu_f}{\mu_{ws}}\right)^{0.11}}{(1.0+3.4f)+\left(11.7+\frac{1.8}{\mathrm{Pr}^{1/3}}\right)\times\left(\frac{f}{8}\right)^{0.5}\times\left(\mathrm{Pr}^{2/3}-1.0\right)}, \quad f=\frac{1.0875-0.1125\left(\frac{Gap}{S}\right)}{\left(1.82\log_{10}\mathrm{Re}-1.64\right)^2} \tag{2}$$

where f is the friction coefficient. The Petukhov correlation was originally developed for a circular channel; however, RELAP5/MOD3.3 adopted f from the Filonenko correlation for a rectangular channel considering the channel gap and span, S.

Ma et al. [3] conducted a single-phase heat transfer experiment on a vertical rectangular channel, which is similar in shape with a new research reactor, and therefore the results are compared to the RELAP5/MOD3.3 calculation. The experimental channel has a 2 x 40 mm narrow rectangular shape and is 1200 mm long. The channel is heated by flowing a current through the metal wall. The channel geometry, heat flux, and flow condition of the RELAP5/MOD3.3 calculation was set to be the same as in the experiment, and the Nu of the calculations and the experimental results are compared.

Figure 1 shows the Nu of the experiment and the Nu calculated by RELAP5/MOD3.3 with the 101 and 102 correlation sets. In a turbulent region, the 102 correlation using the Petukhov correlation tends to over predict the Nu compared to that of the experiment, whereas the 101 correlation with the Dittus-Boelter correlation shows that the Nu is pretty close to the experimental results. The experimental Nu in the laminar region is close to the theoretical value, i.e., Nu=7.63; however, both the 101 and 102 correlation sets estimate a higher Nu than the experimental one. This is because RELAP5/MOD3.3 does not differentiate the flow regime by Re, but simply chooses the maximum Nu among those of natural convection, laminar forced convection, and turbulent forced convection. Since the correlation of turbulent forced convection predict a higher Nu even at a very low Re of less than 1000, the Nu over the entire Re range is governed by the turbulent one. Furthermore, the 102 correlation set has an algorithm in which the code uses Re =2300 for the Nu calculation when Re < 2300, making the Nu constant and much higher than the laminar one at a low Re region.

The RELAP5/MOD3.3 code is then modified to properly select the laminar, transition, and turbulent Nu with respect to Re. When Re≤2500, the code is modified to select the larger Nu between the natural convection and laminar forced convection. When Re>4000, the code selects the turbulent Nu of the Dittus-Boelter correlation, which shows better agreement with that by the experiment. For 2500<Re≤4000, the Nu is interpolated between the one at Re=2500 and that at Re=4000. Figure 1 shows that the modified RELAP5/MOD3.3 shows good agreement of Nu with that of the experiment.

3. Influence of *Nu* modification on Safety Analysis

The influence of *Nu* on the laminar and transition regions in the reactor safety analysis was investigated for a typical loss of flow accident in a new research reactor. The new research reactor is a 5MW, pool-type research reactor operated under atmospheric conditions. The reactor uses plate type fuel, and thus has a narrow rectangular channel for cooling. The coolant is circulated by two pumps, and the flow direction is a downward flow at the core.

A simultaneous failure of two pumps is selected as the scenario for the comparison of the RELAP5/MOD3.3 calculation with original and modified correlation sets of the. Figure 2 shows the core flow reduction by a pump coast-down during the accident. The core flow starts to decrease with the accident initiation and coast down by the inertia of the pump flywheel. The reactor is tripped about 2 s after the accident initiation by a low core differential pressure or low flow signal. The flow completely stops at 68 s after the initiation, and then reverses through natural convection. The fuel temperature and critical heat flux ratio are compared between the original RELAP5/MOD3.3 with 101 and 102 correlation sets, and the modified code.

Figure 3 shows the fuel temperature changes of three cases during the two-pump failure accident. At the beginning of the accident, the fuel temperatures start to increase as the core flow decreases. The fuel temperatures then reach the maximum level when the reactor trips, and then decrease rapidly. The flow is fully turbulent at the maximum fuel temperature, and therefore the 102 correlation set is used to evaluate the steady-state fuel temperature, as the maximum fuel temperature is lower than when evaluated by the 101 set. On the other hand, the 101 correlation set and the modified code show an identical steady-state temperature since they use the same correlation for turbulent forced convection. During the pump coast-down, the flow regime undergoes transition and laminar regions, and therefore the modified code shows a higher fuel temperature than the 101 correlation set. The temperatures reach the second peak a little bit before the flow reversal, and show small differences since the flow is in the laminar regime. After the second peak, the flow oscillates a little bit and then reaches a steady-state, and the fuel temperature keeps decreasing since the decay heat of the fuel decreases.

Figure 4 shows the critical heat flux ratios (CHFR) of three cases during the accident. The CHF is calculated using the Sudo-Kaminaga correlation, which was developed for a vertical narrow rectangular channel of the JRR-3 research reactor [5]. The CHFRs decrease a little and reach the minimum at the reactor trip, and increase rapidly after the trip. The CHFRs then decrease as the pump coasts down and reaches the second lowest peak before the flow reversal. Since the modified code predicts a smaller laminar and transition Nu than the others, the code shows the smallest CHFR at the second peak. However, the MCHFR of the modified code is the same as the MCHFR of the original RELAP5/MOD3.3 with the 101 correlation set, and the difference at the second peak is very small.

4 Conclusion

The influence of single-phase heat transfer correlations on an actual safety analysis of a new research reactor was investigated, since RELAP5/MOD3.3 tends to over predict the *Nu* at a low *Re* flow regime. The *Nu* by the 101 correlation set shows better agreement with the experimental data, compared to that by the 102 correlation set. However, the RELAP5/MOD3.3 with the 101 correlation set still predicts the *Nu* higher at the laminar and transition flow regimes, because it simply selects the maximum *Nu* among the natural convection, laminar forced convection, and turbulent forced convection. The RELAP5/MOD3.3 is modified to select the proper *Nu* with respect to *Re*, and the modified code shows good agreement with the experimental results. The modified RELAP5/MOD3.3 and the original ones with the 101 and 102 correlation sets are used to simulate the two-pump failure of a new 5MW research reactor. The influence of the modification shows a small increase in fuel temperature and a small decrease in MCHFR at the second peak near the flow reversal. However, the maximum fuel temperature and MCHFR, which are important for a safety analysis, are not affected by the modification because they are shown in a few second after an accident initiation, where the flow is fully turbulent.

Acknowledgement

This work has been carried out under the auspices of the Jordan Research and Training Reactor and Training Reactor Project being operated by Korean Atomic Energy Research Institute.

References

[1] Sudo, Y. et al., “Experimental Study of differences in Single-Phase Forced-Convection Heat Transfer Characteristics between Upflow and Downflow for Narrow Rectangular Channel”, Journal of Nuclear Science and Technology, 22(3), pp. 202-212, 1985

[2] Sudo, Y., Kaminaga, M. and Ikawa, H., “Combined Forced and Free Convective Heat Transfer Characteristics in Narrow Vertical Rectangular Channel Heated from Both Sides”, Journal of Nuclear Science and Technology, 24(5), pp. 355-364, 1987

[3] Ma, J. et al., “Experimental studies on single-phase flow and heat transfer in a narrow rectangular channel”, Nuclear Engineering and Design, 241, pp. 2865-2873, 2011

[4] NUREG/CR-5535, RELAP5/MOD3.3 CODE MANUAL, VOLUME IV: MODELS AND CORRELATIONS, December 2001

[5] Sudo, Y. and Kaminaga, M., “A New CHF Correlation Scheme Proposed for Vertical Rectangular Channels Heated from Both Sides in Nuclear research Reactors”, Transaction of the ASME, 115, pp. 426-434, 1993

Figures and tables

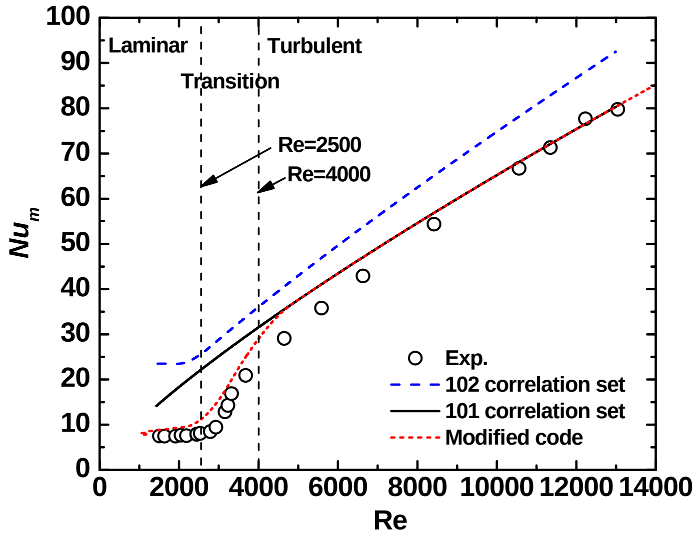

Fig 1. Nu calculated with original and modified RELAP5/MOD3.3 and that of experiment by Ma et al. [3]

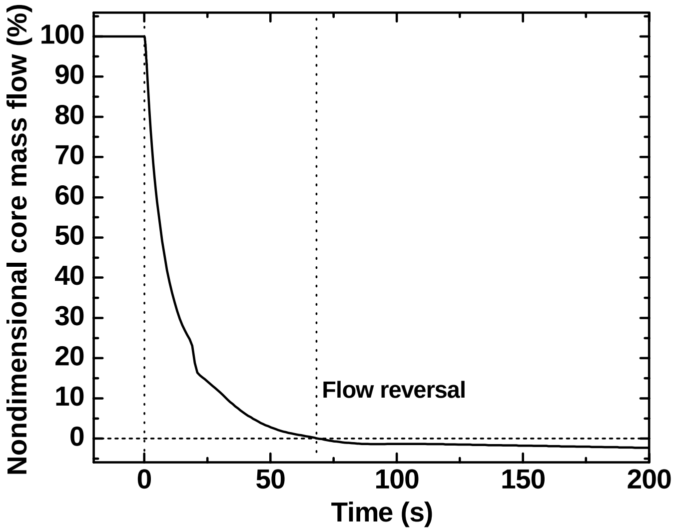

Fig 2. Core mass flow of 5MW research reactor during two-pump failure accident

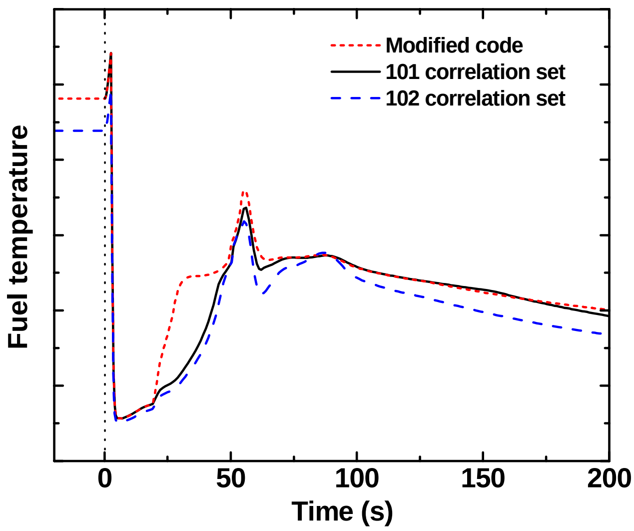

Fig 3. Fuel temperature during two-pump failure calculated using original RELAP5/MOD3.3 with 101 and 102 correlation set and modified RELAP5/MOD3.3

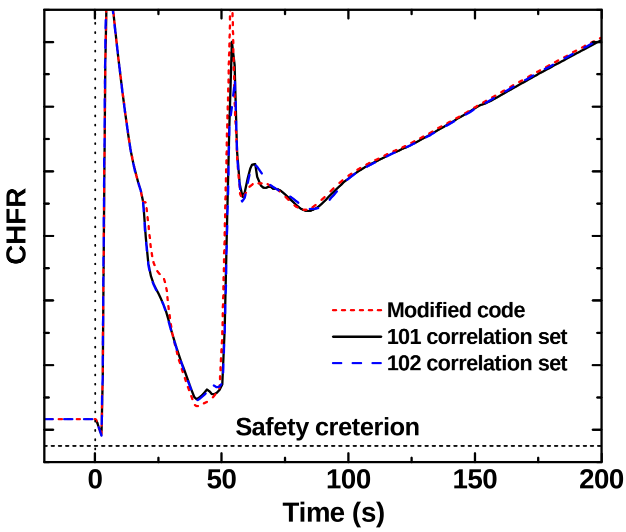

Fig 4. CHFR during two-pump failure calculated using original RELAP5/MOD3.3 with 101 and 102 correlation set and modified RELAP5/MOD3.3