# Prediction of Low-Pressure Onset of Nucleate Boiling using SPACE-RR Code

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## 1. Introduction

Onset of nucleate boiling (ONB) is the phenomena that distinguishes single- and two-phase heat transfer regimes. The ONB condition is usually expressed in wall temperature limit. If the thermal-hydraulic condition is continuously deteriorated (increasing wall heat flux, decreasing coolant flow rate, for example) past this limit, the boiling may progress into more severe and active two-phase phenomenon such as onset of significant void (OSV), onset of flow instability (OFI), and eventually departure from nucleate boiling (DNB). For conservative reasons in both thermal-hydraulic and neutronics point of view, many research reactors adopt ONB as one of steady-state thermal-hydraulic limits and even Limiting System Safety Settings (LSSS) are constructed upon it[1,2,3]. For this importance, several ONB correlations are embedded into SPACE-RR code, which is newly developed system code based on SPACE for research reactor safety analysis application. The ONB temperature margin is evaluated from correlations which are analytically developed or semi-empirically constructed utilizing experimental data. In order to determine whether the calculated margin is safe enough, the prediction capability of the correlation should be checked. In this study, Bergles-Rohsenow correlation currently embedded in SPACE-RR code, is used to predict the ONB temperature of low-pressure experiment from literature[4]. This correlation is selected due to its popularity in safety margin evaluation of low-pressure research reactor cores[5].

## 2. Methods and Results

In this section, the tests performed by Basu et al. (2002) is briefly described and the ONB prediction results are presented.

### *2.1 ONB Experiment by Basu et al. (2002)*

Figure 1 shows the cross-section view of the rod bundle test section taken from the literature[4]. Total 9 rod type heaters are installed in 3 by 3 square array configuration, where Zircalloy-4 cladding is heated by 100 kW DC power. Thin thermocouples are installed in the inner surface of the cladding in 6 axial locations. These are used along with conduction equation to obtain wall temperature at ONB condition. Additional termocouples are installed at the subchannels to measure coolant temperature. Mass flow and the subcooling at the inlet is controlled by valves and preheater. During the test, the heater power is supplied after the desired flow and subcooling conditions are reached. In the test, the ONB point was detected visually along with analyzing temperature and heat flux data. Table I summarizes the test section specification.

Table I: Test Section Specification

| Item | Value |
|---|---|
| Geometry | |
| Heater configuration | Square, 3x3 |
| Heater pitch [m] | 1.429e-2 |
| Heater diameter [m] | 1.11e-2 |
| Heated length [m] | 0.91 |
| Material | |
| Cladding | Zr-4 |
| Test condition (40 cases) | |
| Pressure [bar] | 1.03 |
| Mass flux [kg/m$^2$-s] | 186~631 |
| Inlet subcooling [$^o$C] | 1.7~38.6 |
| Heat flux (uniform) [W/cm$^2$] | 1.6~14.3 |

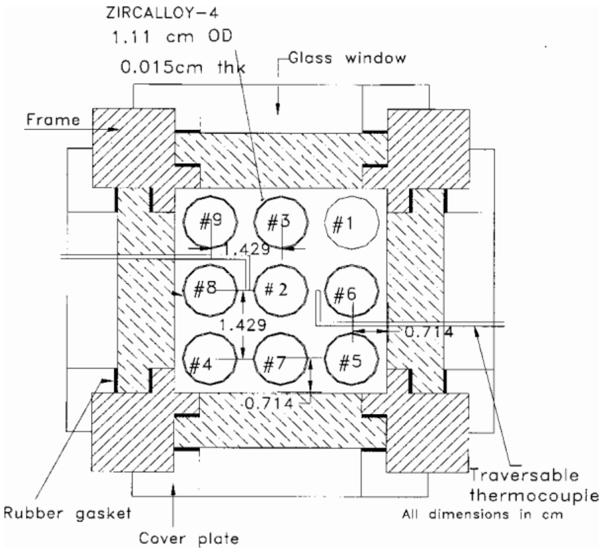


Fig. 1. Cross-section view of the test section (take from [4]).

### *2.2 Prediction by SPACE-RR Code*

In order to simplify the problem, only an interior subchannel is simulated by the code. Figure 2 shows the nodalization. A pipe component with 22 sub-volumes (including 2 volumes for upstream/downstream unheated regions) is used for simulating the subchannel. Temporal face boundary conditions (TFBC) are connected to inlet and outlet of the pipe to obtain desired flow rate. A heat structure is attached to the pipe to simulate heater. For each simulation case, the code is run by first raising the flow rate to the desired value for 200s, and then the heat structure power is slowly increased (5 W/s) until the wall temperature reaches the ONB temperature predicted by the embedded correlation.

Figure 3 compares the predicted wall superheat temperature (= wall temperature - saturation temperature) with the test data. Mean and sample standard deviation of measured-to-predicted value ratio (M/P) was 1.05 and 0.31, respectively. From the simulation results, it seems that the embedded correlation in the code is able to predict overall wall superheat behavior of the experiment. Figure 4 shows the comparison results of wall heat flux at the ONB condition which exhibit different trends. The mean and sample standard deviation of M/P was 1.71 and 0.91, respectively. This means that the code gives conservative predictions in terms of the heat flux. However, considering that the present predictions are obtained from the simplified geometry, it is rather presumptuous to draw conclusions and further study is needed.

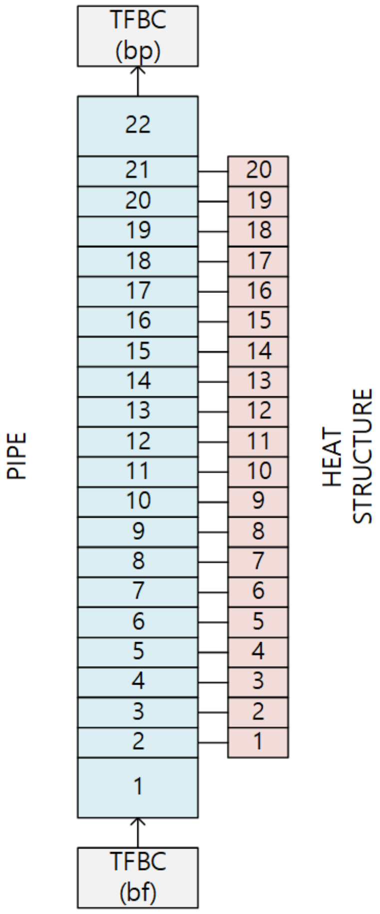


Fig. 2. Code nodalization (not to scale).

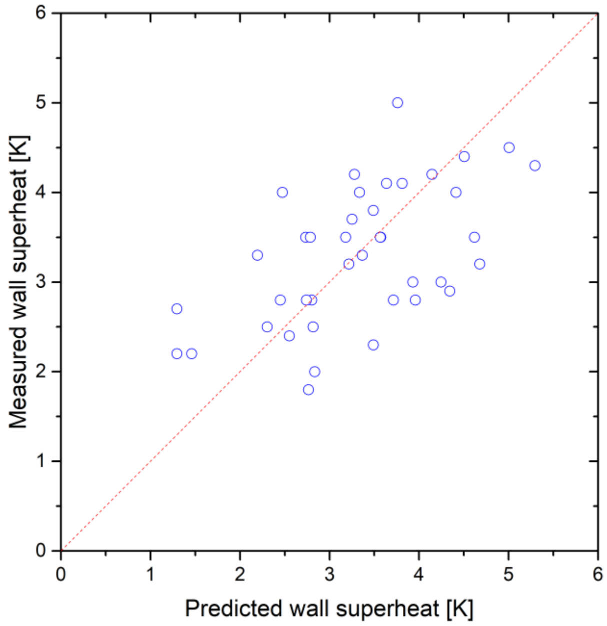


Fig. 3. ONB wall superheat comparison results.

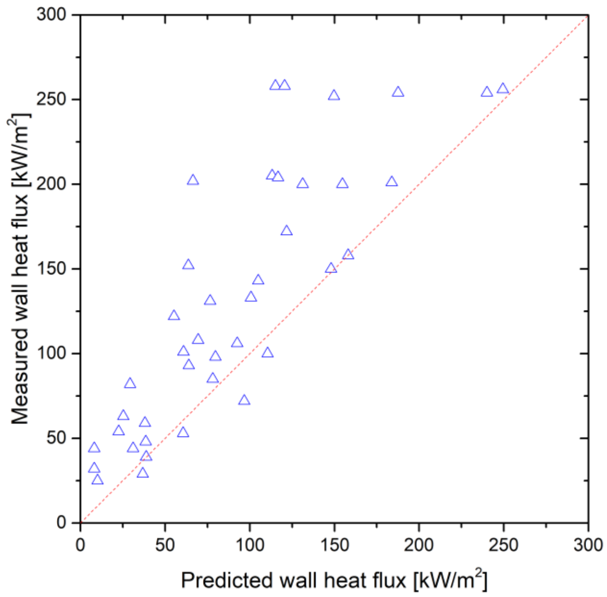


Fig. 4. ONB wall heat flux comparison results.

## 3. Conclusions

In this study, the low-pressure ONB experiments were simulated by safety analysis code SPACE-RR using embedded Bergles-Rohsenow correlation. The simulation results showed that the correlation seem to follow the overall trend of ONB wall temperature measurement data. However, the code gave lower wall heat flux predictions with relatively larger deviation. Considering the current simulation is only carried out for single interior subchannel region, it seems that additional work is required to clarify the applicability of the embedded correlation for low-pressure conditions.

## ACKNOWLEDGEMENT

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## REFERENCES

[1] A.J. Dave, K. Sun, L.-W. Hu, S.H. Pham, E.H. Wilson, D. Jalovka, Thermal-hydraulic Analyses of MIT Reactor LEU Transition Cycles, Progress in Nuclear Energy, Vol.118, 103117, 2020.
[2] A. Alyan, M.S. El-Koliel, Power Upgrading of WWR-S Research Reactor using Plate-type Fuel Element Part I: Steady-state Thermal-hydraulic Analysis (Forced Convection Cooling Mode), Nuclear Engineering and Technology, Vol.52, p.1417, 2020.
[3] Jordan Research & Training Reactor Final Safety Analysis Report, Rev.4, Jordan Atomic Energy Commission, 2016.
[4] N. Basu, G.R. Warrier, V.K. Dhir, Onset of Nucleate Boiling and Active Nucleation Site Density During Subcooled Flow Boiling, Journal of Heat Transfer, Vol.124, p.717, 2002.
[5] Research Reactor Core Conversion from the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels Guidebook, IAEA-TECDOC-233, Vienna, IAEA, 1980.