Brute Force Thermal Margin Analysis of Plate-type Fueled Research Reactor

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1. Introduction

A core thermal hydraulic design of a research reactor ensures that the core has enough thermal margin to maintain its safety. In conventional design process, a hot channel (or assembly) is created and analyzed. The hot channel is formed by conservatively considering a core flow distribution, power profile data both in radial and axial directions, and various uncertainties coming from computation and manufacture. Since the above various factors are combined intentionally to yield most conservative thermal margins, the resulting hot channel is rather far from real and sometimes overly conservative. If the reactor core is designed to have unnecessarily high safety margin, it can sometimes adversely affect the core performance. Therefore, it is worthwhile to quantify the conservatism involved in the thermal margin analysis process, because this can be utilized to find a balance in terms of core safety and performance. For this purpose, Korea Atomic Energy Research Institute (KAERI) has been developing a thermal hydraulic analysis code CORAL (Code Optimized for Research Reactor Thermal Hydraulic Analysis)[1]. In this study, this code is used to analyze the thermal margin of a typical plate-type fueled research reactor core. Instead of carrying out the hot channel analysis, actual power distribution data are directly used to estimate the thermal margin.

2. Methods and Results

In this section, an analysis methods and results are explained.

2.1 Analysis Code

In this study, the CORAL code developed by KAERI is utilized to analyze the thermal margin of the reactor core. The code is developed to overcome following shortcomings of existing research reactor thermal hydraulic analysis codes: first, it is difficult to modify and to enhance the existing codes due to its complicated internal code structures; second, the old-fashioned input format is difficult to understand and simulating arbitrary geometry is almost impossible. The developed code has user friendly input/output formats and simplified internal structures. The code solves steady-state momentum and energy conservation equation for a one-dimensional flow path. The code can also calculate the temperature distribution of the plate and rod type fuel geometry. Light water properties and widely used heat transfer and flow related correlations are embedded into the code. Engineering Hot Channel Factors (EHCFs) can be utilized for estimating the safety margin. Lastly, the code is also equipped with a batch run capability where the user can vary the mass flow (or pressure drop) and thermal boundary conditions and check the effects on the thermal margin.

2.2 Analysis Model

In this study, geometry and operating conditions of a fuel assembly of the typical plate-type fueled research reactor core are utilized. The model reactor is an open-tank-in-pool type reactor which its core generates 5 MW of thermal power from 18 fuel assemblies[2]. Each fuel contains 21 plate-type fuels which form 22 rectangular coolant channels. In this study, an average thermal-hydraulic behavior of the single fuel assembly is simulated using the CORAL code. A right side of Figure 1 shows the nodalization of the single fuel assembly in the core. In order to lump the geometry into one-dimensional flow path, the geometry is divided into several axial regions of same or similar cross-sections, and equivalent hydraulic dimeter is calculated. To check whether the code input generates reliable hydraulic data, as shown in the left side of Fig. 1, the assembly pressure drop is calculated and compared with that of an experiment[3]. For a nominal operating condition (average channel velocity= 2.5 m/s), the difference between the code and the experiment was less than 3%. The deviation seems to come from uncertainties of friction factor correlation and form losses, and these uncertainties are considered in the hot channel analysis. Then, the core pressure drop corresponding to the nominal mass flow rate from the experiment is found and used as a boundary condition for the subsequent thermal margin analysis.

In the hot channel analysis, the highest plate power and the axial power shape with the largest Sum-To-Peak (STP) value is used altogether. Since the STP is a normalized power axially integrated from an inlet to the maximum local power position, the larger STP gives more conservative thermal hydraulic results[4]. In the actual core state, the plate with the largest STP and the plate with the highest power may not match. This causes additional conservatism. In this study, each plate-wise axial power distribution is directly utilized in the CORAL simulation. The total number of the axial power distribution is more than 14,000 considering every core states and cycles where the batch calculation feature of the CORAL code come in handy[5].

In this study, an Onset of Nucleate Boiling (ONB) temperature margin and a Critical Heat Flux Ratio (CHFR) margin are estimated. For ONB temperature, Bergles-Rohsenow correlation is used and Sudo-Kaminaga correlation is utilized for CHFR calculation[6-8].

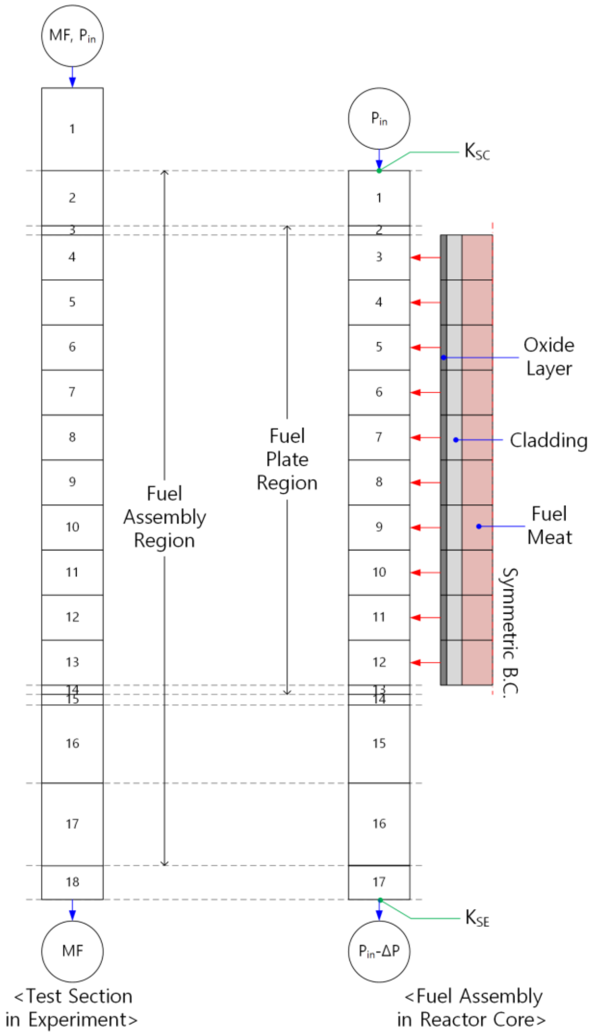

Fig. 1. Discretized Geometry for CORAL Simulation.

2.3 Analysis Results

Figures 2 and 3 show the simulation results. On personal desktop environment (Intel® Core™ i9-9900K CPU), each simulation took roughly 20 milliseconds totaling 270 seconds for entire batch run. The ONB temperature margin was calculated to be larger than 55 ºC, and the MCHFR was estimated to be higher than 4. The assembly mass flow variation due to plate power difference was minor (less than 0.3%). Since hot channel flow reduction penalty and the engineering hot channel factors are not considered, the results corresponds to the thermal margin of the average assembly. Nevertheless, the analysis results indirectly show that the simulated core has sufficient thermal margin.

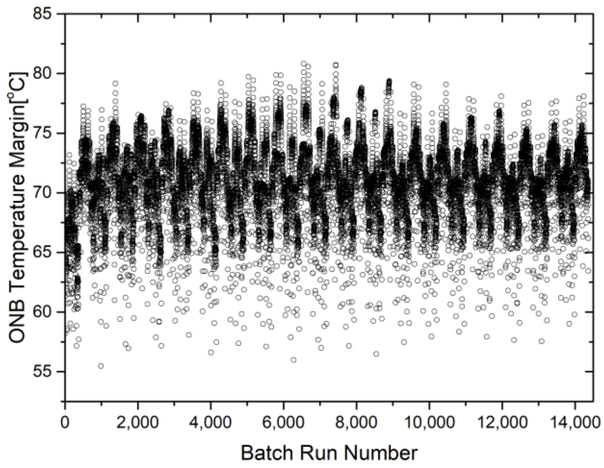

Fig. 2. Simulation Results: ONB Temperature Margin.

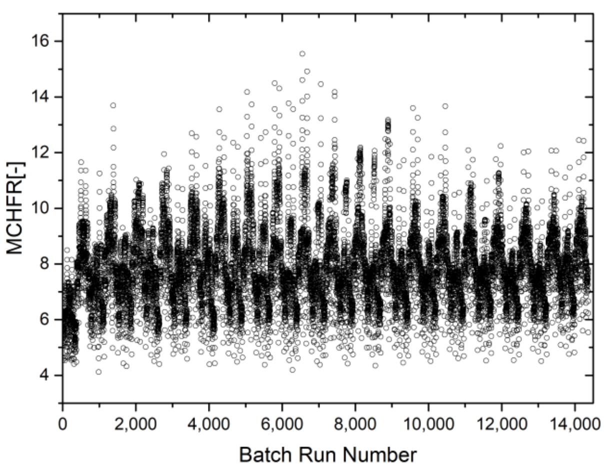

Fig. 3. Simulation Results: Minimum CHFR.

3. Conclusions

In this study, the thermal margin analysis of a typical plate-type fueled research reactor core is performed using in-house code and direct substitution of the raw power distribution data. Comparing fuel assembly pressure drop value from the experiment and the calculation showed that the developed model can simulate hydraulic characteristic of the fuel assembly reasonably. The analysis results showed that the code is able to quickly estimate thermal margins for the entire axial power distributions. The results also revealed that the model core has enough thermal margin in terms of ONB and CHF. The analysis methods and results covered in this paper are the first steps to quantifying the conservatism of the hot channel analysis method. Currently a code update is in progress, which will allow the hot channel analysis to be performed based on the average power assembly calculation results.

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