Subchannel Code Benchmarking to Columbia University 4x4 and Pacific Northwest Laboratory 2x6 Bundle Test Data

Kang Hoon Moon^{a*}, Erdal Ö zdemir^a, Seung Jong Oh^a

^a KEPCO International Nuclear Graduate School, 658-91, Haemaji-ro, Seosaeng-myeon, Ulju-gun, Ulsan, 659-882, Korea *Corresponding author: moonriver0626@gmail.com

1. Introduction

The subchannel code is used to assess the safety of a reactor core at the steady-state and transient conditions. KEPCO Nuclear Fuel (KNF) has been developed new subchannel code, THALES [1], for PWR core design application. In this study, we are comparing the THALES result with VIPRE-01 code result utilizing bundle test data. VIPRE-01 was developed under EPRI sponsorship and has been used by U.S. PWR commercial nuclear utilities, historically.

2. Bundle Test Data Description

This section describes what the bundle test data are used for THALES and VIPRE-01 benchmarking which include the geometry, dimension, and test conditions of each bundle.

2.1 CU 4x4 Rod Bundle Test Data

The rod bundle tests with 4x4 square array was carried out at Columbia University [3]. The mass velocity and enthalpy were measured at the exits of hot and cold subchannels in steady-state runs which are combination sets with pressure, mass velocity, heat flux, and inlet enthalpy.

The geometry and dimension of the test cross section used here is shown in Fig. 1 and the upper half of bundle with fifteen subchannels is applied to THALES and VIPRE-01. The heat flux of each rod was axially uniform, but radially shifted to the hot and cold rods as the ratio of 1 to 0.86.



Fig. 1. Geometry and dimensions of CU 4x4 rod bundle

The number of 73 runs was classified with 6 cases according to selected power, pressure, and mass velocity. One case was chosen as an analysis target and indicated that of test conditions and exit mass velocities in Table I. There is no mention about spacer grid loss coefficient in the reference document. So, the space grid loss coefficient is assumed to be 0.7 by referring typical value of PWR.

Run #	Р	Hin	q"	G_{avg}	G5	G11
	[psi]	[Btu/lbm]	[MBtu/hr-ft ²]	[Mlbm/hr-ft ²]		
32	500	241	0.55593	2.01	1.97	2.09
32	500	286	0.57137	1.95	1.77	2.05
33	500	301	0.57137	1.98	1.66	2.06
73	500	342	0.59067	2.02	1.37	1.32
74	500	325	0.59067	2.00	1.27	1.48
75	500	305	0.58681	1.98	1.38	1.80
76	500	290	0.59067	2.00	1.63	2.00
77	500	270	0.58681	1.97	1.88	1.99

Table I: Test data of CU 4x4 rod bundle

2.2 PNL 2x6 Rod Bundle Test Data

An experiments was performed to obtain the measurements which are local temperature and fluid velocity in the condition of the combined free and forced convection at Pacific Northwest Laboratory, i.e. flow coastdown transients on the bundle of 2x6 rectangular array were collected [4]. There is also no mention about spacer grid loss coefficient in the reference document. So, the space grid loss coefficient is arbitrarily assumed to be 1.0.



Fig. 2 Geometry and dimensions of PNL 2x6 bundle

The configuration of PNL 2x6 bundle is shown in Fig.2 and the whole bundle with 21 subchannels is also applied to THALES and VIPRE-01. Local fluid velocities of twenty one subchannels were measured

i

through nine windows and those of temperature data were collected by thermocouple at the center of subchannel.

PNL 2x6 rod bundle test data was classified with twenty three steady-state and fifteen transient runs according to the combinations of fluid flow rate and heat flux. Two cases selected among fifteen transient cases here are shown in Table II. Transient time was varied from 45 to 500 seconds corresponding to convection flow regime in the reference. Case 1 and 2 corresponds to coastdown for 150 seconds and flow coastdown transient is stopped when the flow rate reaches to 35% of initial value. Analysis of Case 2 is performed for 140 seconds which is not to include recirculation mode. Case 1 is an isothermal test and Case 2 gets a different heating condition; unheated rods for #1~6 and heated rods for #7~12. The axial length of rod bundle is 72 inch, but the heated length, 48 inch is only considered for analysis.

Case #	Р	T _n	q"	Power Skew	G _{avg}
	[psi]	[°F]	[MBtu/hr-ft ²]	[-]	[Mlbm/hr-ft ²]
1	16.8	53.6	0	0:0	0.091273
2	16.8	53.6	0.00619	1:0	0.091273

Table Ⅱ: Test data of PNL 2x6 rod bundle

3. Benchmark Comparisons and Results

This section describes benchmark comparisons between THALES and VIPRE-01. The results of analysis are quantitatively evaluated.

3.1 Benchmark to CU 4x4 at the Steady-State Condition

We chose the same correlations from both codes if possible. Armand correlation was applied to THALES and VIPRE-01 as a two phase friction multiplier and Levy model was also applied to both codes as a quality model. And Chexal-Leollouche model [5] which was a drift flux model affiliated with Zuber-Findlay was also applied to THALES as a void model. The Zuber-Findlay void drift correlation with coefficients developed for EPRI model, so this void model was applied to VIPRE-01. And the constants of turbulent mixing was determined to 0.005.

Fig. 3 and 4 indicates that THALES predicts flow distribution closely to that of measurement using above described models. The maximum and average values of error rate were calculated by equation (1) and those results were 32.0% and 7.11%, respectively. The results of comparison between THALES and VIPRE-01 were 0.88% and 0.30% and the trend was very similar.

$$\varepsilon_i = \frac{|M_i - C_i|}{M_i} \times 100 \, [\%] \quad (1)$$

ε : Error rate*M* : Measured value*C* : Calculated value

where,



: Identification number of measurement

Fig. 3. Subchannel flow as a function of average exit quality at the hot region



Fig. 4. Subchannel flow as a function of average exit quality at the cold region

3.2 Benchmark to PNL 2x6 at the Transient Condition

The thermal hydraulic models were applied to all defaults such as homogeneous of two phase fiction multiplier and void model, and equilibrium of quality since Case 1 and 2 were to analyze low flow regime which contained convection flow condition.

THALES was evaluated to trace the overall behavior of fluid velocity profile and predict well test results at the low flow condition without convection shown in Fig. 5. Case 2 contained a convection condition had a different tendency that the velocity profile was biased to the higher fluid velocity level at the heated subchannel, SC6. This tendency was accelerated at the exit and this was caused by the results of buoyance shown in Fig. 6. The maximum and average values of error rate were also calculated by equation (1) and those results were around 30% and 10% for velocity profile of Case 1 and 2, respectively. The results of comparison between THALES and VIPRE-01 were less than 1%. THALES was also well matched to temperature profile shown in 6. The maximum and average values of error rate on the temperature profile were less than 23% and 5% of Case 1 and 2, respectively. Temperature profile seemed to be effected by velocity profile. The results of comparison between THALES and VIPRE-01 were all round 1%.



Fig. 5. The comparison of fluid velocity distribution at the specific axial height according to subchannel location – Case 1 of PNL2x6



Fig. 6. The comparison of fluid velocity distribution at the specific axial height according to subchannel location – Case 2 of PNL2x6



Fig. 6. The comparison of fluid temperature distribution at the specific axial height according to subchannel location – Case 2 of PNL2x6

3. Conclusions

THALES and VIPRE-01 codes were benchmarked to two kind of bundle test data which were at the steadystate and transient conditions. THALES predicted fluid velocity and temperature profile of bundle test data well and the error rate between THALES and VIPRE-01 was very small.

REFERENCES

[1] THALES Code Manual, KEPCO Nuclear Fuel, 2013.

[2] J. M. Cuta, A. S. Koontz, C. W. Stewart, S. D. Montgomery, and K. K. Nomura, "VIPRE-01: A Thermal Hydraulic Code for Reactor Cores", Battle, Pacific Northwest Laboratories, California, 1989.

[3] F. S. Castellana, J. E. Casteline, "Subchannel Flow and Enthalpy Distribution at the Exit of a Typical Nuclear Fuel Core Geometry", Columbia University, New York, 1971.

[4] J. M. Bates, E. U. Khan, "Investigation of Combined Free and Forced Convection in a 2x6 Rod Bundle During Controlled Flow Transients", Pacific Northwest Laboratory, Washington, 1980.

[5] B. Chexal, G. Lellouche, J. Horowitz and J. Healzer, "A Void Fraction Correlation for Generalized Applications", Progress in Nuclear Energy, Vol.27, No.4, pp.255-295, 1992.