SPACE Code Assessment for FLECHT Test

Hyoung Kyoun Ahn^{*}, Ji Hong Min, Chan Eok Park, Seok Jeong Park, Shin Whan Kim Safety Analysis Department, KEPCO E&C, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Rep. of KOREA ^{*}Corresponding author:hkahn@kepco-enc.com

1. Introduction

According to 10 CFR 50 Appendix K, Emergency Core Cooling System (ECCS) performance evaluation model during LBLOCA should be based on the data of FLECHT test. Heat transfer coefficient (HTC) and Carryout Rate Fraction (CRF) during reflood period of LBLOCA should be conservative. To develop Mass and Energy Release (MER) methodology using Safety and Performance Analysis CodE (SPACE), FLECHT test results were compared to the results calculated by SPACE. FLECHT test facility is modeled to compare the reflood HTC and CRF using SPACE. Sensitivity analysis is performed with various options for HTC correlation. Based on this result, it is concluded that the reflood HTC and CRF calculated with COBRA-TF correlation during LBLOCA meet the requirement of 10 CFR 50 Appendix K.



Figure 1. Modeling using SPACE for FLECHT Test

2. Analysis

FLECHT test was carried out to obtain experimental data to evaluate ECCS performance during LOCA. Two types of heater were designed to model the commercial fuel bundle of 7x7 and 10x10. Heated length of heater is 12 feet and heated pitch is 0.563 inch as square pitch, the diameter of the heater is 0.422 inch. Test is started with being decayed heat of heater and coolant injection from lower plenum. According to result of the experiment the core experiences various two-phase flow regimes: liquid flow, nucleate boiling flow, transition boiling, film boiling flow, dispersed flow of liquid droplets, steam flow.

2.1 Model

As shown in Figure 1, FLECHT test facility is modeled with 20 thermal hydraulic nodes and heaters. FLECHT test cases are shown in Table 1. Test number of 0690, 6948, 4225 and 3541 are chosen to represent wide range of the experimental conditions. The measured mass flow data of test are given by boundary condition of low plenum. The wall temperature of fuel and coolant temperature are also considered as initial conditions of analysis. The decay heat of fuel in the FLECHT test is also simulated by modelling heated structure

Table 1. FLECHT Test Cases					
Test No.	0690	6948	4225	3541	
Initial Clad Temp. (°F)	1,531	1,615	1,596	1,598	
Flooding Rate (in/sec)	0.6	1.0	1.9	5.9	
Peak Power Density (kW/ft)	0.69	1.24	1.24	1.24	
Inlet Coolant Temp. (°F)	190	146	153	148	
System Pressure (psia)	15	58	59	57	
Bundle Size	10x10	7x7	10x10	10x10	

Table 1. FLECHT Test Cases

2.2 Sensitivity Analysis

At the beginning of the reflood period, the vapordroplet flow is dominant. As the core is being refilled, the transition boiling flow has affected heat transfer between heat wall and water. As the water level increases, film boiling HTC is changed with transition boiling correlation, critical heat flux (CHF) correlation and nucleate boiling correlation at the surface of the axial heater. When the heaters are quenched, therefore, comparatively large HTCs are applied

For the purpose of sensitivity analysis, various heat transfer correlations are evaluated for each flow regime. The COBRA-TF correlation is used to calculate the transition boiling region and film boiling look-up table (2006) is used to calculate the film boiling region. The Chen correlation is used to calculate the nucleate boiling flow region and Dittus-Boelter correlation is used to calculate the single phase flow region. CHF and condensation heat transfer correlation are also considered to predict the wall heat flux. Heat transfer correlations used to sensitivity study are shown in Table 2.

Option	#1	#2	#3
Transition boiling	COBRA-TF	TRACE	
Film boiling	2004 film boiling look-up table	COBRA-TF	
CHF corr.	2006 AECL-UO CHF lookup table	1986 AECL-UO CHF lookup table	Biasi's corr.
Single phase flow	Dittus -Boelter corr.	Gnielinski corr.	
Wall condensation	Colburn -Hougen	No-Park	Vierow -Schrock
Inverted annular film boiling	Modifed Hammouda & Groeneveld (1997)	RELAP5 /MOD3.3	
Nucleate boiling	Chen	Thom	

 Table 2. Sensitivity for Wall Heat Transfer Correlations

3. Results

Figure 2 shows analysis results of reflood heat transfer coefficient and carryout rate fraction for test number of 0690 with lowest flooding rate and lowest peak power density. The analysis result is more conservative compared with test data. According to the test result, core quenching occurs at about 700 seconds. It is found that SPACE predicts core quenching 200 seconds earlier. For CRF, analysis result is oscillated frequently. The CRF result with polynomial curve fit is compared with data of test. For all of the range, SPACE predicts more conservative CRF in point of mass release.

Figure 3 shows analysis results for test number of 3541 with highest flooding rate. The analyzed HTC is more conservative compared with test data. According to the test result, core quenching occurs at about 70 seconds. It is found that SPACE predicts core quenching 20 seconds earlier. For CRF, analysis result is oscillated. SPACE predicts more conservative CRF with respect to mass release

Figure 4 shows analysis results about test number of 4225. Mostly the analyzed HTC is more conservative compared with test data. According to the test result, core quenching occurs at about 185 seconds. It is found that SPACE predicts core quenching time similarly with test result. For CRF, analysis result is oscillated. SPACE predicts more conservative CRF in the view point of mass release.

Figure 5 shows analysis results about test number of 6948 with 7x7 bundle. Before about 200 seconds, the analyzed HTC is more conservative comparing with test data. The test result shows that core quenching occurs near at 250 seconds. However, SPACE predicts the

reflood HTC less conservatively until about 300 seconds. SPACE predicts CRF more conservatively in the aspect of mass release. Nevertheless analyzed reflood HTC is not calculated conservatively.



Figure 2. Reflood HTC and CRF for test number of 0690



Figure 3. Reflood HTC and CRF for test number of 3541



Figure 4. Reflood HTC and CRF for test number of 4225



Figure 5. Reflood HTC and CRF for test number of 6948

Sensitivity study is performed for wall heat transfer correlations to analyze test number of 6948. It is shown in Figure 6 that the results of the sensitivity analysis. The options of first column in Table 2 are used to calculate the reference case. The quenching times that are calculated with various wall heat transfer correlations except COBRA-TF correlation for film boiling option are delayed compared with the test result. By this result, it is important to apply effective film boiling option during reflood period. Figure 7 is shown that reflood HTC and CRF are conservative when using COBRA-TF correlation for film boiling. All of the test range, SPACE predicts reflood HTC and CRF conservatively.



Figure 6. Sensitivity result for wall heat transfer correlations



Figure 7. Reflood HTC and CRF for test number of 6948 with COBRA-TF film boiling correlation

4. Conclusions

In this study, the analysis results using SPACE predicts heat transfer phenomena of FLECHT test reasonably and conservatively. Reflood HTC for the test number of 0690, 3541 and 4225 are conservative in the reference case. In case of 6948 HTC using COBRA-TF is conservative to calculate film boiling region. All of analysis results for CRF have sufficient conservatism. Based on these results, it is possible to apply with COBRA-TF correlation to develop MER methodology to analyze LBLOCA using SPACE.

REFERENCES

[1] F. F. Cadek, D. P. Dominicis, R. H. Leyse, PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Final Report, WCAP-7665, April, 1971.

[2] S. I. Lee, S. Y. Lee, C. E. Park, H. R. Choi and C. J. Choi, Assessment of Reflood Heat Transfer Model of COBRA-TF Against ABB-CE Evaluation Model, Korean Nuclear Society, May, 2000.

[3] Topical Report, KOPEC Improved Mass and Energy Release Analysis Methodology (KIMERA), Rev.0, July, 2007.
[4] S. J. Ha, C. E. Park, K. D. Kim and C. H. Ban, Development of the SPACE Code for Nuclear Power Plants, Nuclear Technology, Vol. 43, No. 1, 2011.