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Prediction of Reflood Behavior under Flow Blockage Condition in a SCTF Experiment

Young Seok Bang [a*], Joosuk Lee [a], Andong Shin [a], Il Suk Lee [a]
[a]*Korea Institute of Nuclear Safety, 62 Kwahak-ro, Yuseong, Daejeon, Korea*
[*]*Corresponding author: k164bys@kins.re.kr*

1. Introduction

Consideration of flow blockage due to swelling and rupture of the fuel cladding has been emphasized and requested in the thermal-hydraulic safety analysis of large break loss-of-coolant accident (LBLOCA), especially during the reflood phase of LBLOCA [1, 2]. The flow blockage was known to be significant at the reflood phase of LOCA and to have an impact on core cooling. Also, a need to evaluate the effect of the flow blockage under a certain burnup condition has been raised at the recent regulatory research to revise the Emergency Core Cooling System (ECCS) rule to implement the effect of fuel burnup [3]. In the context, the validation of the system thermal-hydraulic codes for the experiments simulating flow blockage condition was one of the important issues of the recent study [4].

Several tests with flow blockage were conducted in the Slab Core Test Facility (SCTF) in Japan [5]. Major findings of the SCTF Core-I test program were reported as a Research Information Letter (RIL) 157 [6] of US Nuclear Regulatory Commission (NRC). Blockage up to about 60% of the flow area of a fuel assembly was considered in those tests. Accordingly, it is meaningful to select a SCTF experiment and validate the MARS-KS code [7]. In the present paper, predictability of the MARS-KS 1.5 code for the selected test was discussed. Also the performance of the modeling scheme specific to flow blockage proposed by the authors [4] was examined.

2. Description of SCTF Experiment

Slab Core Test Facility (SCTF) was a two-dimensional representation of a reactor vessel (RV) of Westinghouse 4-loop pressurized water reactors (PWR) with the simplified intact loop and broken loop. The reactor vessel has a configuration in full-height and full-radius (8 fuel assemblies) and in one-bundle-width. The volume scale is 1:21. A sketch of the facility was shown in Fig. 1. Approximately 2000 electrically heated rods were installed in the facility (234 heater rods and 22 non-heated rods per bundle). Eight bundles were placed in a row as shown in Fig. 2. For bundles 3 and 4, the sleeve was placed in the middle position axially on the outside of all rods to simulate the flow blockage. A chopped cosine shape of power in an axial direction was simulated. It led to 60% blockage of the bundle flow area.

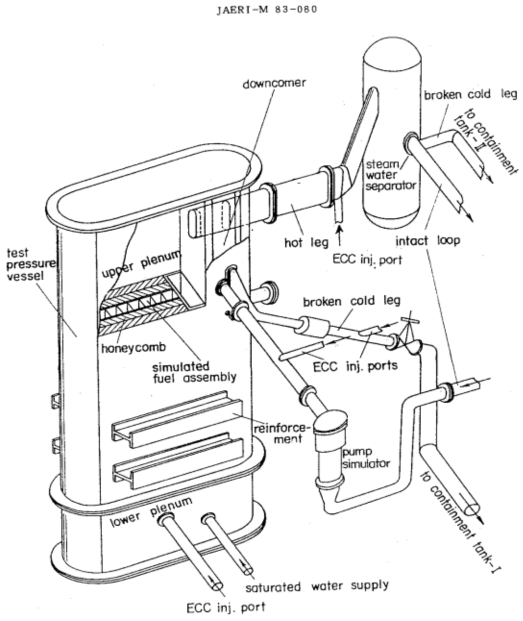

Fig. 1. Sketch of SCTF

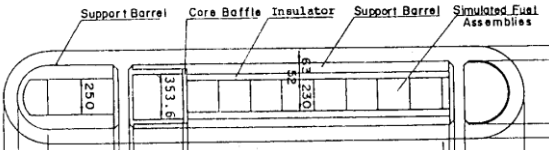

Fig. 2. Cross-section of reactor vessel

The experiment S1-01 (Run 507) [8] was selected in the present study. The test was a base case for the all the other SCTF Core-I tests. The test condition was scaled down from the actual plant condition. In the test, ECCS injection to lower plenum (LP) in order to represent the forced-feed flooding was simulated, therefore, the downcomer (DC) was isolated from the LP. The radial power shape as shown in Table 1 was used.

Table 1. Profile of bundle power

Bundle	1, 2	3, 4	5, 6	7, 8
Power, kW	887	944	900	815

Prior to the test, the RV, steam/water separator, pump simulator, containment tanks 1&2, and all the piping were heated up to near saturation condition at 0.2 MPa. Then, core temperature was adjusted such that the maximum rod temperature reaches 523 K. Then, the LP was filled with the near saturated water to 1.3 m from the bottom of the RV. The test was started by initiating the core heating. When the cladding temperatures reach 926 K, the heating was stopped and the simulation of the core power decay was started, which was to approximate the ANS decay heat curve. Simultaneously, the accumulator injection (ACC) was initiated. The maximum flow rate was 22 kg/sec. The ECCS injection was switched-over to the low pressure core injection (LPCI) after 20 seconds of ACC injection. The flooding rate was approximately 2.5 cm/sec. The specific information of the SCTF tests including the data were a part of 2D/3D program [9], and limited to its member countries. Thus all the information presented in this paper were from the google search and the data was digitized manually.

3. Code and Modeling

A system thermal-hydraulic code, MARS-KS 1.5 [7] was used, which has been extensively used for regulatory auditing calculations. The input for MARS-KS code was prepared from the TRACE code input in the reference [10]. The components '3DVESSEL' of the TRACE input were changed to 'PIPE's and multiple crossflow junctions. Several junctions which were implicitly defined in TRACE input were explicitly defined. The MARS-KS nodalization of RV was shown in Fig. 3 and one of the loop in Fig.4, respectively. The core was modeled by 8 PIPE components in parallel, each having 11 axial volumes and crossflow junctions connecting the bundles. The LP, the upper plenum (UP), the upper head (UH), the core bypass region, and the DC were each modeled by single PIPE.

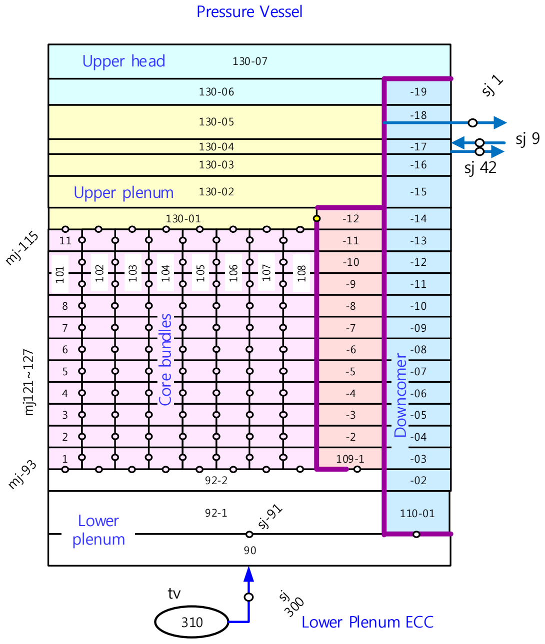

Fig. 3. MARS nodalization of SCTF RV

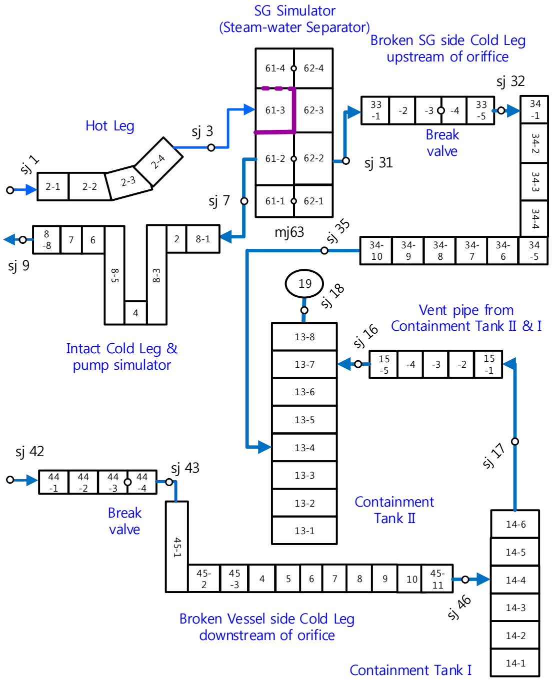

Fig. 4. MARS nodalization of the SCTF loop

4. Results and Discussions

Two cases of calculations were conducted:

(1) Base case in which any modeling of flow blockage was not applied to the bundles 3 and 4, and

(2) Blockage case in which the blockage modeling was applied

The blockage modeling scheme was from the author's previous study [4] and composed of (a) reduction of downstream junction area, (b) change of hydraulic diameter, (c) additional k-factors at the upstream and downstream junctions, and (d) reduction of flow volume of the blocked node.

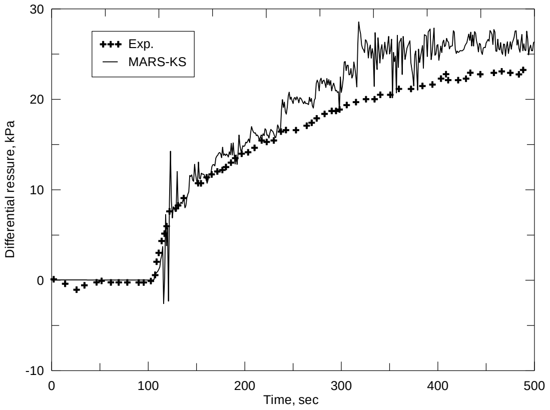

Fig. 5. Comparison of core differential pressure

4.1 Base case

Fig. 5 shows a comparison of differential pressure between inlet and exit of the core. As shown in the figure, MARS-KS calculation agreed well with the test data. In the later phase, a little over-prediction was found, which may be related to the interphase friction in the core under the low reflooding condition of the experiment.

Fig. 6 shows a comparison of rod surface temperatures of the bundle 4. Among the data of the 10 thermocouples mounted in axial direction, the data of 8 positions were compared. As shown in the figure, the predicted rod surface temperature during the reflood phase are significantly different from the experimental data. The temperatures were a little over-predicted at the mid elevations, however, the predicted quenching was earlier than the experiment for all the elevations. As discussed above, the problem may be related to the interphase friction of the core.

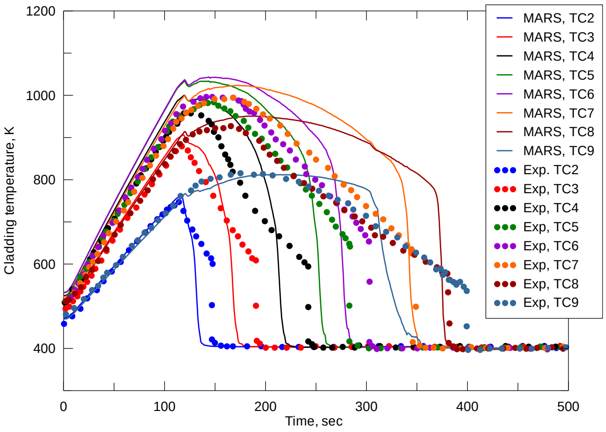

Fig. 6. Comparison of rod surface temperatures

Fig. 7 shows a comparison of rod surface temperatures of the sixth thermocouple of the bundle 1 and one of the bundle 4 with the calculation. Due to higher power density of the bundle 4, its temperature was higher than the one of the bundle 1. MARS-KS code clearly indicated the effect of the difference of the power density.

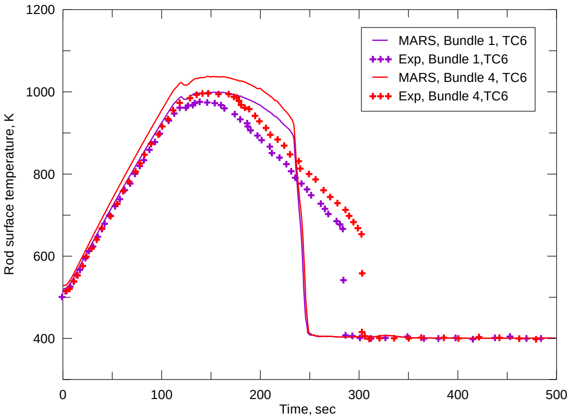

Fig. 7. Comparison of rod surface temperatures for bundles 1 and 4

4.2 Blockage case

Fig. 8 shows comparisons of rod surface temperatures at the nodes of 4th through 9th of the bundle 4, which were predicted with and without the flow blockage model discussed previously. It was found that rod surface temperatures at the nodes lower than the blockage node (6th node) were decreased a little when applying the blockage model. On the contrary, the flow blockage model resulted in higher rod surface temperatures for the node in higher elevation. It can be clearly understood the flow blockage is to delay the reflooding for the downstream nodes while it allows to keep the water for a longer time for the upstream nodes. The maximum impact from blockage was estimated 18.5K. Also it was found that the effect of the volume reduction in the present blockage model was not significant. However, effect of blockage should be further investigated using the different tests of the SCTF.

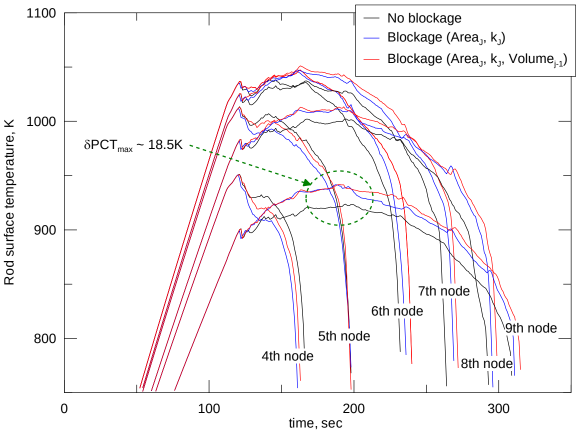

Fig. 8. Comparison of rod surface temperatures with and without using blockage model

5. Conclusions

A reflood experiment with flow blockage of the Slab Core Test Facility (SCTF) was calculated using MARS-KS 1.5 code. Additionally, a simple model describing flow blockage effect was also examined. The followings can be concluded:

(1) The present code and modeling have provided some conservative results in predicting the rod surface temperatures. To resolve the difference in reflood behavior, improvement on hydraulic behavior including the interfacial drag in low flooding rate is needed.

(2) The impact of the maximum rod surface temperature predicted by the proposed flow blockage model was 18.5 K at the downstream of the blockage, which supported the conservatism of the present flow blockage model.

Acknowledgements

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REFERENCES

[1] USNRC, Compendium of ECCS Research for Realistic LOCA Analysis, NUREG-1230, 1988.
[2] USNRC, Best Estimate Calculation of Emergency Core Cooling System Performance, Regulatory Guide 1.157, 1988.
[3] Joosuk Lee, et al, Development of Regulatory Audit Methodology and Criteria on Core Coolability for Design Basis Accidents, KINS/RR-1856, 2018.
[4] Young S. Bang, et al, Modeling of flow blockage due to swell and rupture for analysis of LBLOCA using a multiple fuel rods scheme under high burnup condition, Annals of Nuclear Energy, Vol. 134, p. 350, 2019.
[5] Hiromichi Adachi, et al, Design of Slab Core Test Facility (SCTF) in Large Scale Reflood Test Program, Part 1 (Core-I), JAERI-M-83-080, 1983.
[6] USNRC, Report on the Results of the Core I Test Series at the Japan Atomic Energy Research Institute Slab Core Test Facility, Research Information Letter No. 157, 1988.
[7] KINS, MARS-KS Code Manaul, Volume II: Input Requirements, KINS/RR-1282, Rev.1, 2018.
[8] K. Hirano, et al, Quick-Look Report in Large Sclae Reflood Test-21. Test S1-01 (Run 507), JAERI memo 9733, 1981.
[9] USNRC, 2D/3D Program Work Summary Report, NUREG/IA-0126, 1993.
[10] B.E. Boyack et al, TRAC-M/F77, Version 5.5 Developmental Assessment Manual, Vol. II: Appendices, LA-UR-01-2105, 2001.