An analytic study on LBLOCA for CANDU type reactor using MARS-KS/CANDU

Jin-Hyuck Kim, Joo-Sung Kim, Kwang-Won Seul and Chae-Yong Yang
Korea Institute of Nuclear Safety, Korea
(jhkim@kins.re.kr)

Abstract

This study provides the simulation results using MARS-KS/CANDU code for the Large Break LOCA of CANDU type reactor. The purpose of the study is to evaluate the capability of MARS-KS/CANDU for simulating the actual plants (Wolsong 2/3/4). The steady state and the transient analysis results were provided. After the sensitivity study depend on break size, the case that 35% of the inlet header known as the accident that has the most limiting effect on the temperature of the fuel sheath was calculated. In order to evaluate the results, the results were compared with those of CATHENA simulation.

1. Introduction

MARS-KS (Multi-dimensional Analysis of Reactor Safety-KINS Standard) code has been developed for the realistic multi-dimensional system to estimate and evaluate the thermal hydraulic behavior of nuclear power plants. [1,2] Even through MARS-KS code has usually been made use of analyzing the light water reactor transients, it has been the thermal hydraulic code which can be employed the heavy water reactors with the adoption of the CANDU oriented models. There are some models and correlation equations for characteristics that reflect the CANDU reactors such as fuel channel model (CANDHAN), components models (Wolsong pump model, header-feeder model, etc.), improved off take model at arbitrary-angled branches, and so forth. [3-5]

The purpose of this study is to assess the simulation results MARS-KS/CANDU for the thermal hydraulic behaviors of the system during the accident to evaluate the reliability of MARS-KS/CANDU as one of the regulatory safety analysis codes and to compare with CATHENA which has been used by utilities. [6] The accident studied in this paper is one of the Wolsong units 2/3/4 LBLOCA (Large Break Loss of Coolant Accidents) transient situation described in the FSAR (Final Safety Analysis Report). [7] With the accident scenario for the evaluation, analysis was performed for the LBLOCA of heavy water type nuclear reactors with all the safety system working; and calculation was done for the case that 35% of the inlet header was broken as it is a kind of accident that has the most limiting effect on the fuel sheath temperature according to existing literatures. [7] And the sensitivity study was done to see if the 35% break size was the most limiting case of the accident as suggested. After obtaining the initial conditions from the steady state option, the transient condition was calculated with setting time of 900 sec to see the early time transient after break occurrence at the inlet of the header.

2. System modeling

2.1 Nodalization

For the analysis of the accident, systems inside of the heavy water reactor were considered in the modeling including the nuclear fuel channel; the heat transport system; the main steam system; and the ECCS (Emergency Core Cooling System). Figure 1 shows the nodalization diagram for the accident analysis of Wolsong 2/3/4 used for MARS-KS/CANDU in this study. As seen in the figure, the primary side was composed in two loops. Each loop comprises of two sets of averaged fuel channels, two SGs (Steam Generator), the feeder pipes, the IHDs (inlet header) and OHDs (outlet header), two RCPs (Reactor Coolant Pump), and the safety equipment; and the two loops are connected with the pressurizer. The 380 nuclear fuel channels which were simulated by CANCHAN were presented by four averaged channels, and each averaged channel was used in the modeling on the average of ninety five nuclear fuel channels. The averaged channel simulated by 12 nodes and that was the same as the number of the nodes of CATHENA modeling in FSAR. The header components were modeled as BRANCH component and that was connected to feeders by junctions of BRANCH. The LOCA break was modeled on IHD8.

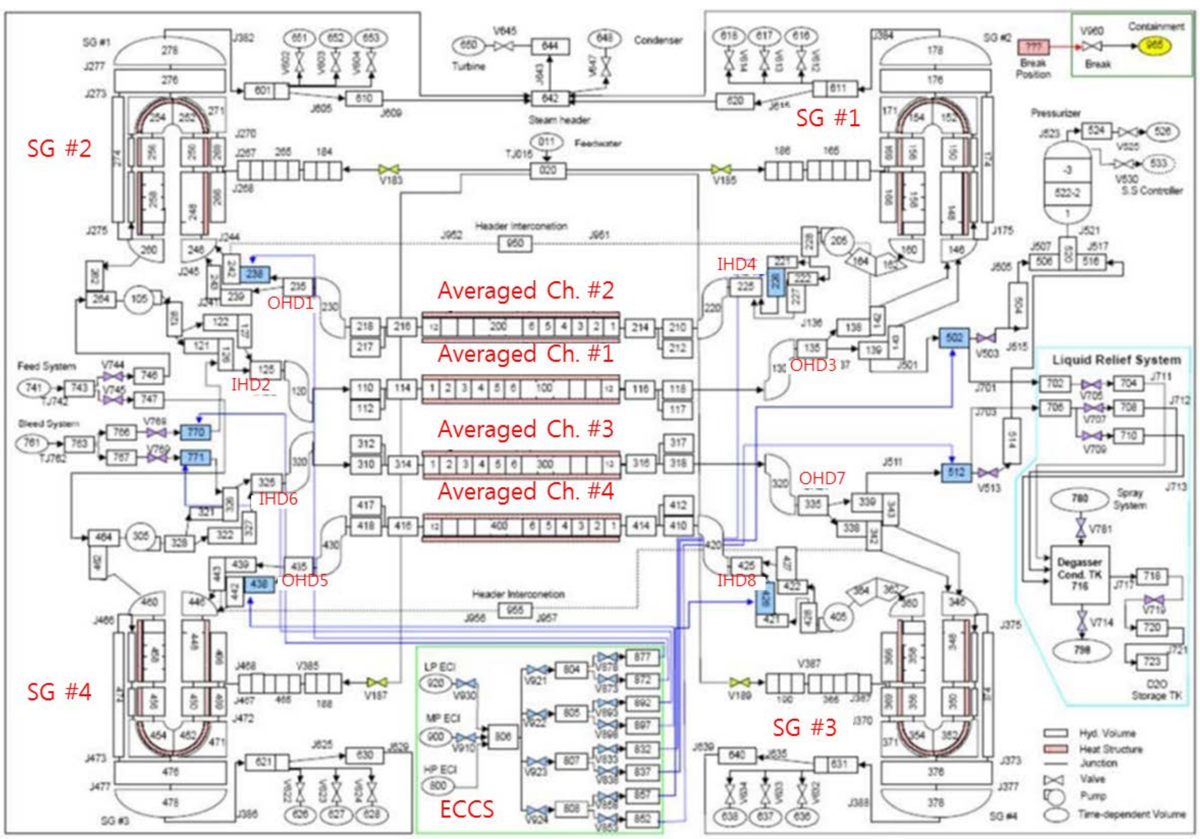

Figure 1 Nodalization diagram for MARS-KS/CANDU (Wolsong-2/3/4).

The secondary system was modeled as four SGs which were supplied water by the feed water system. In order to simulate the SG level control, the feed water supply and the condenser for SG were modeled as time dependent volume with the input of temperatures and pressures as boundary conditions. The MSSV (Main Stream Safety Valve) was presented as a valve component for simulating the crash cooling down of the system.

The ECCS was comprised of HP, MP and LP (High, Middle and Low Pressure, respectively) ECIs (Emergency Coolant Injection). The HP ECI was modeled as accumulator component and MP and LP ECI were modeled as a time dependent volume. The ECCS was connected to inlet and outlet headers.

2.2 Simulation methodology

As mentioned in previous, the study modeled LBLOCA in Wolsong 2/3/4 to assess the code capability for the CANDU type reactor. The LBLOCA in primary circuit was considered where all safety systems (ECCS, etc.) were assumed to be available. The break size was decided after sensitivity study to assess the limiting case for fuel sheath temperature. According to the simulation assumption, the RCPs are operated until automatic pump trip from the system low pressure signal. Furthermore, the MSSV (Main Steam Safety Valve) is operated for the crash cooling-down the system using SG. And the feed water valve is normally operated based on the water-level logic of steam generator. Table 1 shows the major control logics for the simulation.

Table 1 Control logic for MARS-KS/CANDU

Control parameter	Control logic
Reactor trip	LOCA + 0.0s
LOCA signal	Pressure in Max(Min(IHD),Min(OHD)) < 5.25 MPa
Loop isolation	LOCA signal + 0.0s
Governor valve close	LOCA signal + 5.0s
MSSV open	LOCA signal + 30.0s
HPECI start	Pressure in Max(Min(IHD),Min(OHD)) < 3.62 MPa
MPECI start	HPECI start+90.0s
LPECI start	MPECI stop (= amount of MPECI>200m^3)
RCP stop	Pressure in OHD < 2.5 MPa

According to FSAR, the trip logic of RCPs in the CATEHNA simulation employed that all 4 RCPs were to trip if one of the pressure of the header reached under 2.5MPa. In contrast, the CATHENA calculation in this study was modified to employ same logic with the MARS-KS/CANDU as shown in the table 1 to compare their results reasonably. But beyond that, the other simulation conditions of FSAR were applied to the CATHENA calculation in this study to compare to the results from the MARS-KS/CANDU. The initial conditions of the transient state were determined by the steady state analysis. The power distributions of the nuclear channels were input by the table for the analysis; and the power pulse of broken loop and intact loop were input to the transient state analysis by different table. The critical flow model of Henry-Fauske was employed for the analysis of the critical flow of the break. [8]

3. Steady state analysis

According to the procedures stated earlier, computations were performed for the steady state and the transient state respectively, the results of which were compared and reviewed with the analysis results by CATHENA from FSAR. First, for the steady state analysis of the Wolsong 2/3/4 model, computation was done with 103% nuclear reactor power which was decided to consider the fuel aging effect depend on FSAR. All variables were found to reach the steady state value by reviewing the variations in the steady state computation. Table 2 shows the comparison of the steady state variables from between the computation of MARS-KS/CANDU and the CATHENA. As indicated in the table, the computations of MARS-KS conform generally similar to the value of CATHENA.

Table 2 Steady state condition (103% Power)

Parameter	CATHENA	MARS-KS/CANDU
RIH Pressure [MPa]	11.4	11.4
RIH Temperature [K]	541	541.1
ROH Pressure [MPa]	10	10
ROH Temperature [K]	583.5	583.78
Channel Flow Rate [kg/s]	1897	1900.2
Power for each Pass [MW]	527.875	527.875
Heat Load to S/G [MW]	532	531
Pressurizer Level [m]	12.48	11.08
S/G Drum Pressure [MPa]	4.69	4.80
S/G Drum Temperature [K]	533	534.55
Steam Flow Rate [kg/s]	1061.37	1060.38
Feed Water Flow Rate [kg/s]	1060.71	1064.02
Feed Water Temperature [K]	459	459.6

4. Sensitivity study according to break size

Transient (accident) analysis was performed after the occurrence of the large break in the inlet header under 103% power operation conditions of the nuclear reactor. Above all, sensitivity analysis was done with MARS-KS/CANDU to see if the 35% break size was the most limiting case of the accident as suggested both in the existing literature. Figure 2 shows the mass flow rate at the broken loop at 6th node on averaged channel 4. As shown in the figure, MARS-KS/CANDU results showed similar trend to the CATHENA results in general. In addition, the stagnation of flow was most noticeable around from 5 to 25 seconds in the 35% accident. These flow stagnations indicated more remarkable in the flow for 35% break accident than 100% break accident (guillotine break). That is because of that the guillotine break accident shows the reverse flow along the nuclear fuel channel since the flow throughout the break is substantially

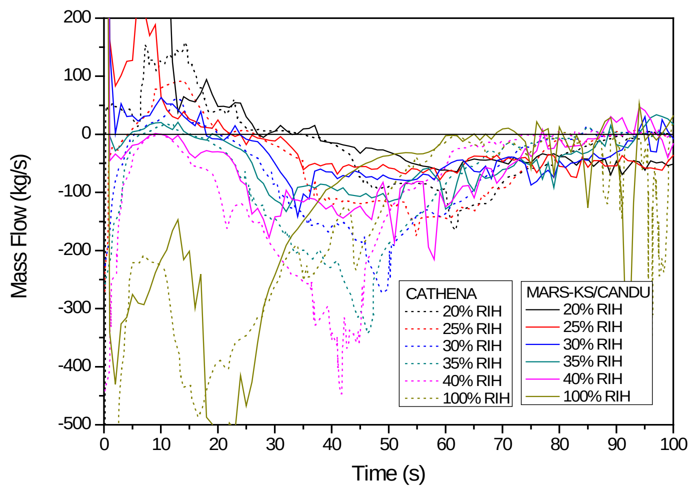

Figure 2 Mass flow rates in the middle of fuel channel. (RIH20~100%)

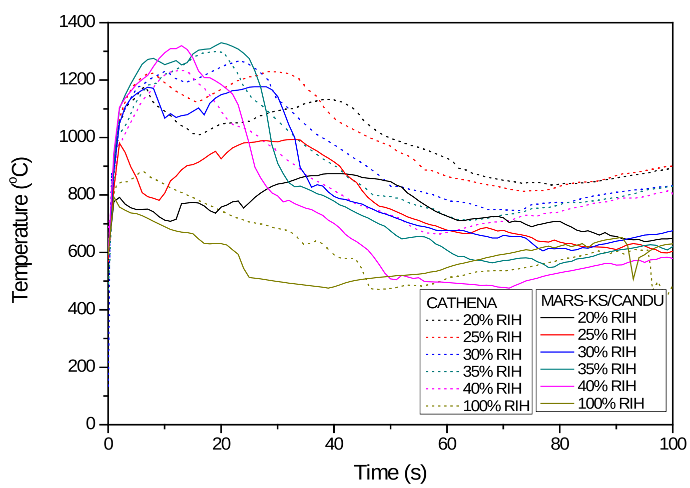

Figure 3 Fuel sheath temperature (7th node) in broken loop.

bigger than the head of pump. However, the 35% break accident indicates the stagnation flow because of the balance between the head of pump and the flow throughout the break. According to these results, figure 3 indicates that the temperature of the nuclear fuel sheath was highest for the 35% break than any other break size due to the effect of flow stagnation. Generally the analysis result by CATHENA showed similarity to MARS-KS/CANDU.

5. Safety injection modification

Figure 4 shows the comparison of the ECI mass flow rate between MARS-KS/CANDU and CATHENA. As seen in the figure, there are big differences in the mass flow behavior especially in HP ECI. The HP ECI`s flow rate simulated from MARS-KS/CANDU showed relatively

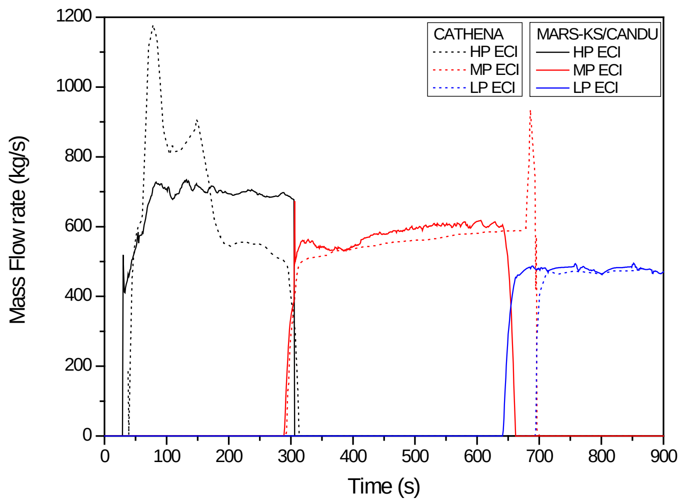

Figure 4 Comparison of mass flow rates of ECI (original).

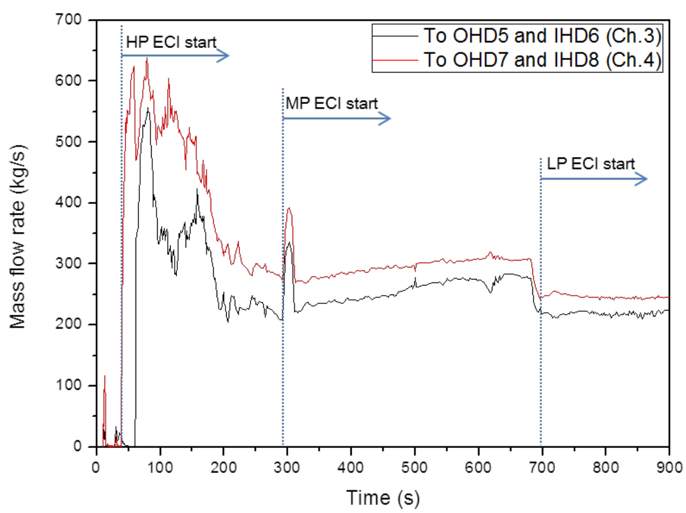

Figure 5 Mass flow rate of ECI in modified calculation.

maintaining the certain level during the injection period. On the other hand, the results from CATHENA showed considerable transition from start to end of the injection as shown in the figure. If the amount of the water injection to the system is different, the calculation of the thermal hydraulic behaviors should be different. Therefore, in order to obtain proper comparison results between two different codes, the amount of the safety injection for MARS-KS/CANDU had to be modified. Therefore, the mass flow rates of each ECI from CATHENA simulation was employed as the boundary condition of MARS-KS/CANDU calculation using a time dependent volumes and junctions. Figure 5 shows the mass flow rates of ECI to the broken loop, which were employed as the boundary condition for the MARS-KS/CANDU calculation.

6. Accident analysis

As the result of the sensitivity analysis showed that 35% inlet header break accident is the most limiting accident as in the existing literatures, it was decide to proceed with the accident analysis using that break size. Table 3 provides the accident scenario of the system. As shown in the table, general process of the accident was similar to the result by CATHENA. Since the safety injection time of MARS-KS/CANDU reflected the result of CATHENA, the starting and ending of ECI of them showed same. As shown in figure 6, the coolant loss caused sharp increase in bubble fraction, and accordingly reduces the nuclear fuel cooling capacity, so the fuel sheath temperature rapidly increased as shown in figure 7 representing the temperature of middle (7^{th} node) of the fuel sheath. And the increase in reactor power due to the positive void coefficient of reactivity also influenced the increase in temperature. This temperature rise was more remarkable in the broken loop than the intact loop. The temperature rise in intact loop was limited due to the separation of the loop by the loop isolation signal.

Table 3 Sequence of events

Event	CATHENA (sec)	MARS-KS/CANDU (sec)
LOCA Signal	8.6	9.3
Loop Isolation Start	8.6	9.3
HP ECI Start	27.8	27.8
MSSV Open	38.6	39.3
RCP # 4 Coastdown start	57.1	53.2
RCP # 3 Coastdown start	58.0	53.9
RCP # 2 Coastdown start	105.0	102.3
RCP # 1 Coastdown start	105.1	102.4
HP ECI Stop	311.2	311.2
MP ECI Start	292.8	292.8
LP ECI Start	698.1	698.1

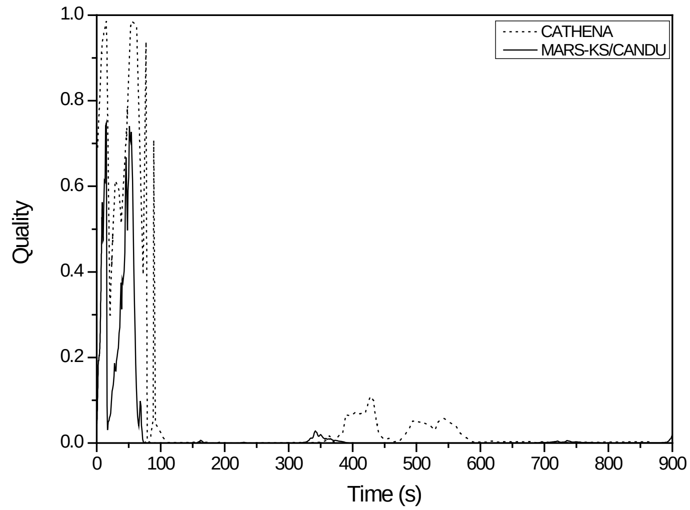

Figure 6 Quality of outlet header in broken loop.

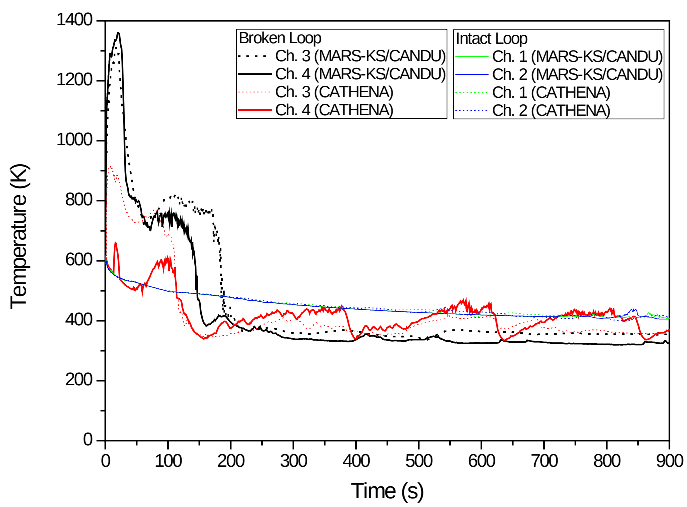

Figure 7 Fuel sheath temperature of 7th node.

After the fuel sheath temperature of the broken loop reached the peak in around 20 seconds, the rapid temperature drop represented as shown in the figure due to the decrease in the power after the nuclear reactor trip. And then, the fuel was cooled down and stabilized by the ECCS. This transition of temperature showed more obviously in averaged channel 4 than channel 3 due to the short distance from channel 4 to the break. In contrast, the temperature of the intact loop slowly decreased without a big increase due to the loop isolation. As shown in the figure, the results of MARS-KS showed similar with the results by CATHENA in general. Though the temperature of 7&9th node in averaged channel 3 showed different trend to the results by CATHENA due to the difference in the occurrence of the critical heat flux as seen in the figure 8, the temperature of other nodes were presented similar with CATHENA.

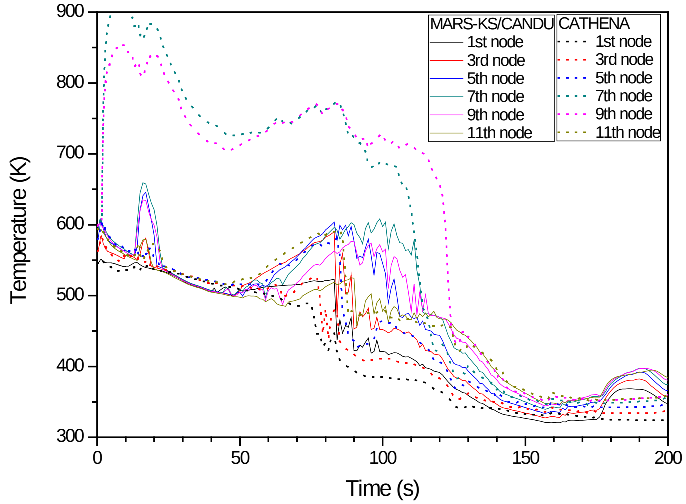

Figure 8 Fuel sheath temperature of averaged channel #3.

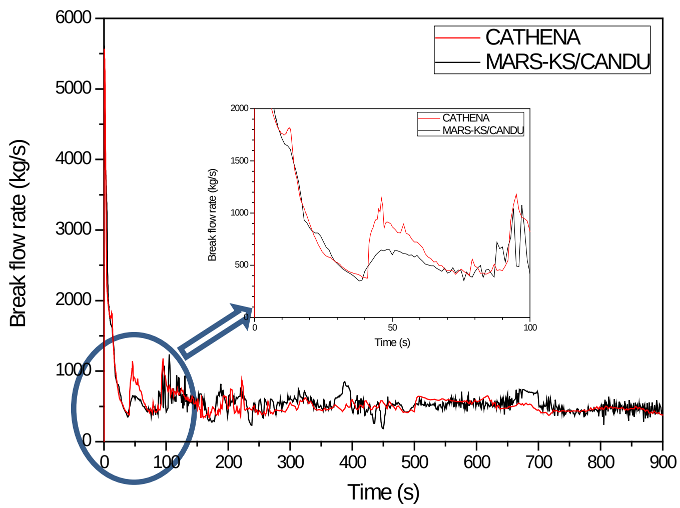

Figure 9 Break mass flow rate.

Figure 9 shows the mass flow rate at the break. After the safety injection started, the break flow showed significant increase. That increase in break flow leaded to the increase in the temperature as seen in the figure. Since this tendency strongly appeared in CATHENA results, the temperature of CATHENA after approximately around 100 seconds higher than that of MARS-KS/CANDU. Therefore, the quenching time of CATHENA could be lengthened as seen in figure 7.

Figure 10 shows the coolant mass flow rates in the middle of the averaged fuel channels of the broken loop indicating reverse flow in averaged channel 4 which is downstream of the broken header. As seen in the figure 11 representing the mass flow rate at the fuel channel of the

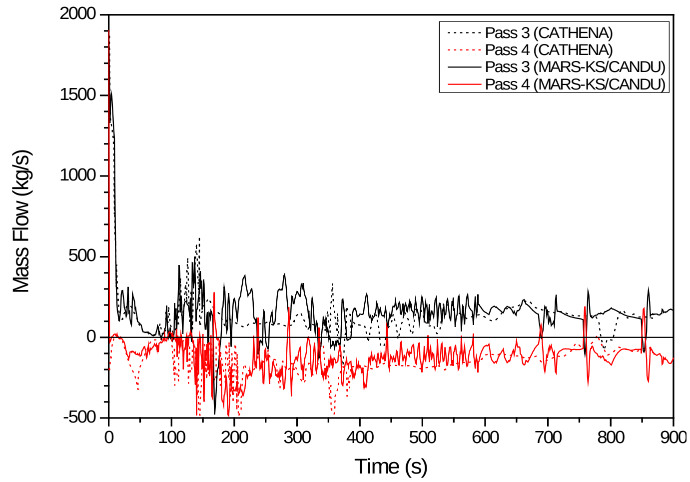

Figure 10 Coolant mass flow rates of broken loop.

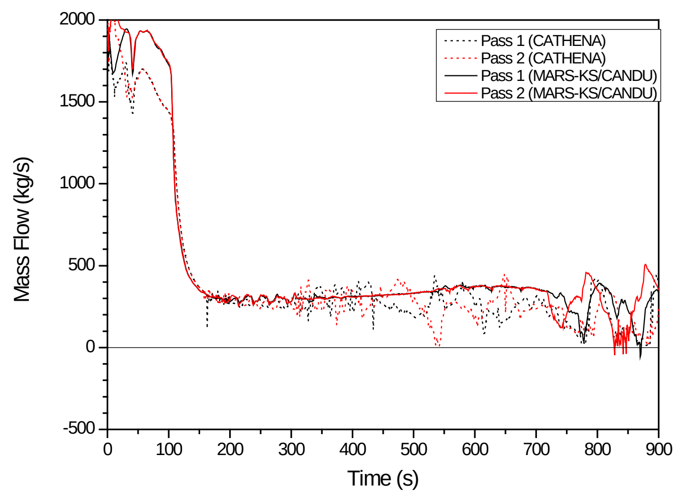

Figure 11 Coolant mass flow rates of unbroken loop.

intact loop, the flow rate decreased sharply after the RCP stop. And the decrease of flow in the early stage of the accident was affected by the reactor stop.

The pressure of inlet and outlet headers is shown in figure 12 and 13, respectively. After the break occurred, the pressure in the header decreased continuously. The pressure in the broken loop more rapidly decreases than intact loop. After rapid pressure drop the pressure reduction rate limited by the safety injection. The pressure showed slightly recovery when high pressure safety injection started as shown in the figure. This trend represented more obviously in MARS-KS/CANDU calculation due to the lower break flow at that time. The continuous injection of safety water from ECCS leaded the pressure into stabilized region.

According to the transient analysis results of the MARS-KS/CANDU and comparison with the CATHENA, the MARS-KS/CANDU showed appropriate prediction for the thermal-hydraulic behavior and the safety parameters.

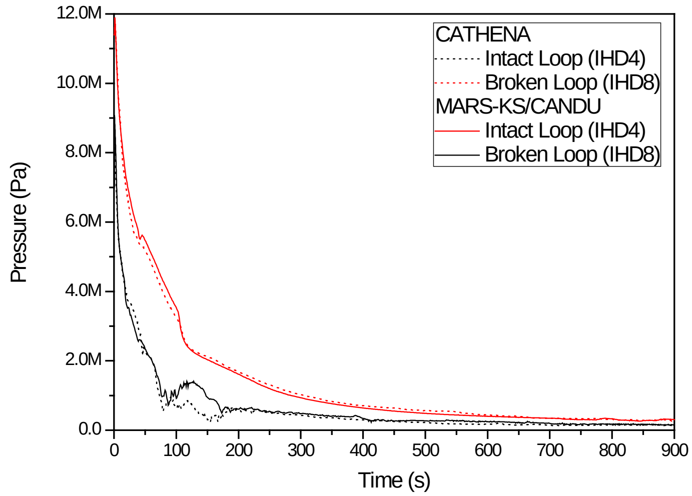

Figure 12 Pressure in inlet headers.

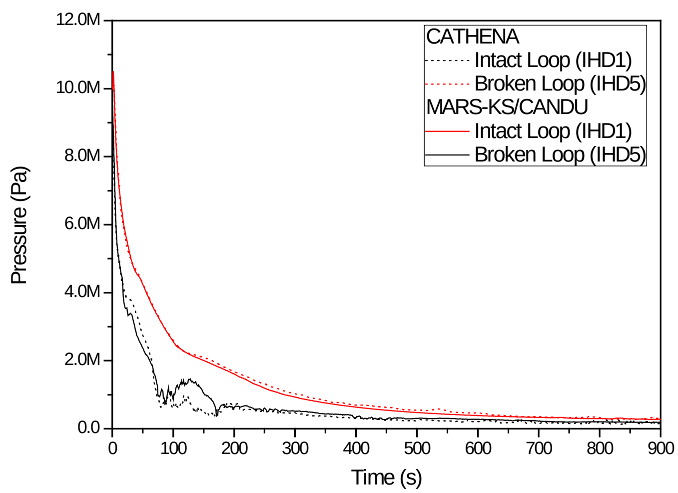

Figure 13 Pressure in outlet headers.

7. Conclusion

This study was performed to verify the capability of MARS-KS/CANDU to actual power plants by accident analysis of the LBLOCA in the inlet header of the Wolsong 2/3/4. Accident analysis was done for the 35% break of the inlet header after sensitivity study according to the break size to decide the limiting case, which causes biggest rise in the nuclear fuel sheath temperature owing to the flow stagnation due to the balance with the discharge flow through the break and the pump head as concluded in the FSAR. The initial conditions derived from the steady state analysis were input to the transient state analysis, which generally showed the similar result with the CATHENA results. On the base of the steady state analysis, the transient state analysis was accomplished. Accident scenario and the thermal-hydraulic behaviors of the system simulated by MARS-KS/CANDU showed reasonable prediction. And those also reflected the CATHENA analysis results well in general. Base on the above-mentioned results, the MARS-KS/CANDU could be said that it plays a good role in the accident analysis of the heavy water reactor and that it may be widely used as a regulatory code if interfaced appropriately with other reactor core analysis codes.

8. References

[1] Korea Atomic Energy Research Institute, "MARS code manual, Vol. I: Code structure, System models, and Solution methods", KAERI/TR-2812/2004, December 2009.

[2] B.D. Chung, et al., "Development and assessment of multi-dimensional flow models in the thermal-hydraulic system analysis code MARS", KAERI/TR-3011/2005.

[3] B.D. Chung, et al., "Quality improvement of MARS code and establishment of code coupling", KAERI/RR-3104/2009.

[4] US NRC, "Improvements of RELAP5/MOD3.2.2 models for the CANDU plant analysis", NUREG/IA-0189, 2000.

[5] Hwang S. H., "A study on the horizontal stratified entrainment characteristics at arbitrary-angled branch pipes", Ph. D. Thesis, Seoul National University, 2005.

[6] Atomic Energy of CANADA Limited, "CATHENA Mod-3.5d/Rev 2, Input Reference", 2005.

[7] Final safety analysis report of Wolsong unit 2, Chapter 15, 1993.

[8] R. E. Henry and H. K. Fauske, "The Two-phase critical flow of one-component mixtures in nozzles, orifices, and short tubes", Transactions of ASME, Journal of Heat Transfer, Vol. 93, pp. 179-187, 1971.