A study on accident analysis for Wolsong 2/3/4 using MARS-KS/CANDU

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1. Introduction

The purpose of this study is to assess the simulation results MARS-KS/CANDU to evaluate the reliability of that as one of the regulatory safety analysis codes and to compare with CATHENA which has been used by utilities. The accident studied in this paper is one of the Wolsong units 2/3/4 LBLOCA (Large Break Loss of Coolant Accidents) described in the FSAR (Final Safety Analysis Report). 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. 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. Modeling Information

Figure 1 shows the nodalization diagram of Wolsong 2/3/4 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), and two RCPs (Reactor Coolant Pump); and the two loops are connected with the pressurizer. The 380 nuclear fuel channels simulated by CANCHAN were presented by

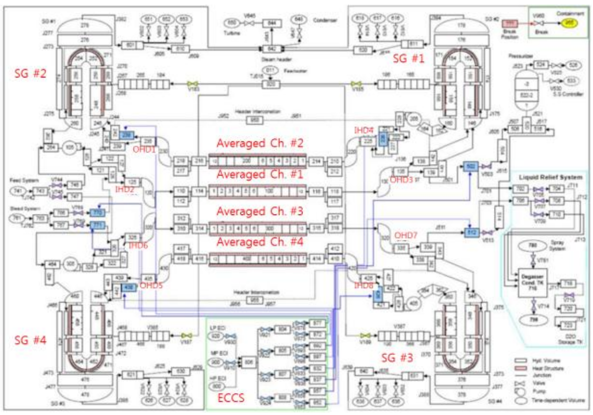

Figure 1 Nodalization diagram for CANDU type reactor (Wolsong 2/3/4).

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 LOCA break was modeled on IHD8. Table 1 shows the control logics for the simulation. The power distributions of the nuclear channels were input by the table for the analysis. The critical flow model of Henry-Fauske was employed for the analysis of the critical flow of the break.

Table 1 Control logic for MARS-KS/CANDU

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

3. Sensitivity study according to break size

The sensitivity analysis was done 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 middle of the nuclear fuel channel of the broken loop at 6th node on averaged channel 4. As shown in the figure, the stagnation of flow was most noticeable around from 5 to 25 seconds in the 35% accident.

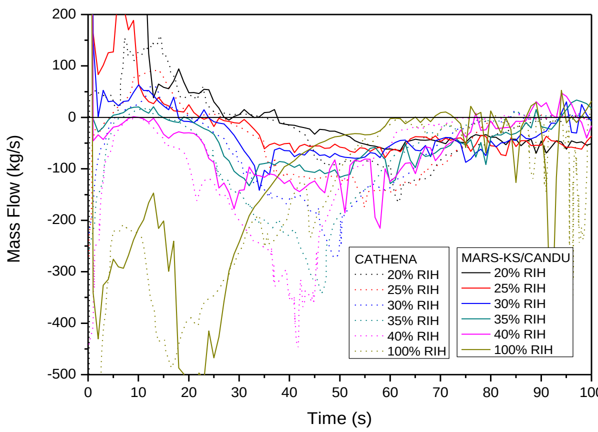

Figure 2 Mass flow rates in the middle of fuel channel.

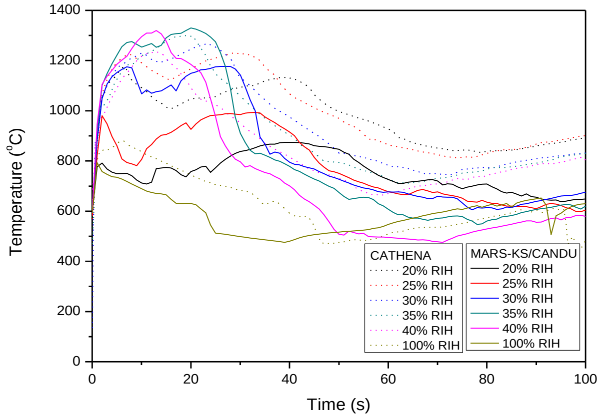

Figure 3 Fuel sheath temperature in broken loop.

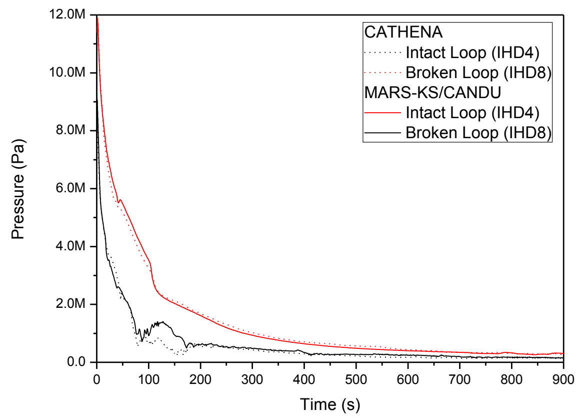

Figure 5 Pressure in inlet headers.

4. 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 2 provides the accident scenario of the system.
The fuel sheath temperature rapidly increased as shown in figure 4 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 loop isolation.

Table 2 Sequence of events (unit: s)

Event	Time
LOCA Signal	9.3
Loop Isolation Start	9.3
HP ECI Start	27.8
MSSV Open	39.3
RCP # 4 Coastdown start	53.2
RCP # 3 Coastdown start	53.9
RCP # 2 Coastdown start	102.3
RCP # 1 Coastdown start	102.4
HP ECI Stop	311.2
MP ECI Start	292.8
LP ECI Start	698.1

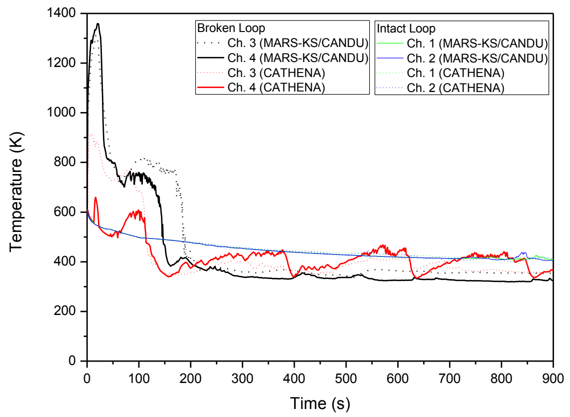

Figure 4 Fuel sheath temperature of 7th node.

The pressure of inlet is shown in figure 5. 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 continuous injection of safety water from ECCS leaded the pressure into stabilized region.

5. 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.

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