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MARS-KS Assessment on B9802 SBLOCA test in RD-14M Test Facility

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Abstract

This paper presents the MARS-KS assessment results on B9802 SBLOCA test performed in RD-14M test facility. Through the simulation results, it is provided that MARS-KS well predicts the thermal hydraulic behavior in the system. However, the calculated FES temperature trend at middle region of channel, and steam condensation in steam generators show inconsistency in comparison with test data. This study suggest that a further study on the heat transfer model prediction considering bundle effect in the large diameter long horizontal pipe, and condensation model along the u tube is required to have more accurate code prediction.

1. Introduction

For an IAEA international standard problem entitled “Comparison of heavy water reactor (HWR) code predictions with SBLOCA experimental data”, two tests performed in the RD-14M facility were selected for blind calculation as a code validation exercise. The one, test B9006 was carried out for the simulation of small break LOCA, and the other test B9802 is for steam generator condensation. MARS-KS assessment results on B9802 test are presented in this paper. Korea Institute of Nuclear Safety (KINS) participated in the IAEA standard problem in order to validate the MARS-KS (Multi-dimensional Analysis for Reactor Safety-KINS Standard) code for CANDU analyses. MARS-KS has been originally developed for a realistic analysis of thermal hydraulic transients in pressurized light water reactors. For widening of MARS-KS applications to heavy water reactors, some important models for CANDU characteristics have been modified and applied to original MARS code and now still under development. The R&D project in order to utilize the MARS-KS as a thermal hydraulic system code for CANDU regulatory audit calculation is ongoing in KOREA.

2. RD-14M Experiment [1]

2.1 RD-14M Test Facility Description

RD-14M is an 11 MW, full elevation scaled thermal hydraulic test facility possessing most of the key components of a CANDU primary heat transport system. The facility is arranged in the standard CANDU two pass, figure of eight configuration. The reactor core is simulated by ten, 6 m-long horizontal channels. Each test section has simulated end-fittings and seven electrical heaters, or fuel element simulators, designed to have many of the characteristics of the CANDU fuel bundle. The RD-14M fuel element simulators are designed to model CANDU natural uranium fuel in power density or heat flux, and in heat capacity or heat up rate. Test sections are connected to headers via full-length feeders. Above header piping is also CANDU typical including two full height, U tube steam generators and two bottom suction centrifugal pumps. Steam generated in the secondary, or shell side of the steam generators is condensed in a jet condenser and returned as feedwater to the boilers. The primary side pressure is controlled by a pressurizer/surge tank using a 100 kW electric heater. The facility operates at typical CANDU primary system pressures and temperatures and is designed to produce the same fluid mass flux, transit time, pressure, and enthalpy distributions in the primary system as those in a typical CANDU reactor under both forced and natural circulation conditions.

2.2 B9802 Test Procedure Description

Test B9802 was a 3 mm inlet header break experiment, to provide data on the influence of condensation rates in the steam generators on primary loop response under conditions where such sensitivity is expected. The break was represented by a 3 mm orifice installed in the drain line from header 8 to an inventory tank. The break discharge flow was condensed and measured. Test B9802 had a smaller break than B9006, no ECC, no pump ramp, and no secondary pressure ramp. Channel power remained at full power during most of the transient, and pump speed was slightly reduced from nominal in order to achieve higher initial enthalpy in the channels and thus more boiling. This test was intended to study boiling in channels and condensation in steam generators in a slowly depressurizing loop rather than a blowdown.

3. MARS-KS Code Description

A MARS-KS has been developed for a realistic analysis of thermal hydraulics transients in pressurized water reactors. More information of MARS-KS is described in detail reference [2]. For widening of MARS-KS applications to heavy water reactors, some important models for CANDU characteristics have been modified and applied to original MARS code. The major model modifications for CANDU reactor were performed, such as Critical flow model, Nuclear kinetics model, Critical heat flux model, Valve and spray model, Improvement of horizontal flow regime map, Heat transfer model in horizontal channel. These six items were improved not only for CANDU reactor but also for generic nuclear reactor system. The CANDU fuel channel heat transfer model and the flow regime model were improved to be suitable to a CANDU specific feature. The stratification criteria proper to CANDU fuel feature and the fuel element heatup process induced by stratification were newly implemented. The digital control model was mainly deduced from CANDU plant specific feature of digital control, but it could be also applied to the special processing feature of plants. By this improvement, the digitally processed signal can be simulated. Other four items, i.e. ANS94-4 decay heat model, Moody critical model, motor operative valve model and pressurizer spray model, were developed on generic base and could be applicable to PWR also. Especially, the plutonium contribution to decay heat can be considered by the ANS94-4 decay heat model. The Moody critical flow model using a heavy water property can provide an additional capability to evaluate the conservative break flow in CANDU system. Simulation of different rate of opening and closing the motor valve is now possible when calculating the liquid relief vale behaviour during transient. And the new pressurizer spray model can be used for evaluating the droplet size effect on the condensation. All of those improvements were verified through some assessments with simple conceptual problems and Marviken critical flow test.

4. System Model and Nodalization

According to the general principle of MARS-KS code’s system model and nodalization, RD-14M facility was modeled as shown in Fig. 1. The system model composes of primary heat transport system including heated sections, feeders and headers, pumps, ECI system, and break orifice, as well secondary system.

Basically, in view of hydrodynamic model, more fine nodalization scheme is adopted for the components where two phase phenomena and system functions play important roles in transient behavior. And cross flow junctions are modeled where the flow direction is perpendicularly linked to the main flow direction such as end fitting connection.

In modeling the primary-side nodalization, volume, length, flow area and elevation change of each MARS-KS pipe component resembles the RD-14M test facility, as closely as possible. This ensures that the fluid transit time and hydrostatic pressure changes around the loop are represented accurately during the simulation. The pipe component is adopted for heated section model because the pipe horizontal stratified criterion works for low- and moderate viscosity liquids, including water, at least in small diameter pipes up to 5 cm. In the heat structure, the seven fuel rods are combined into a single fuel rod heat structure maintaining the surface area, mass and equivalent heated perimeter. These fuel pins generate heat corresponding to each channel power. The power distribution in the axial direction is assumed to be uniform.

The break is modeled by the trip valve attached at inlet header 8. And the break valve is connecting with the downstream pipe and a time dependent volume for discharge reservoir. It is anticipated that this modeling suppresses unnecessary fluctuation of header pressures during two phase stage. The improved critical flow model is adopted instead of original RELAP5 critical flow model developed by Ransom and Trapp. The system controls are modeled by trip cards, which accept logical inputs and variable inputs based on time, pressure, and other thermal hydraulic parameters. The power trip, pump coast down, and break initiation, etc. are controlled through these trip cards.

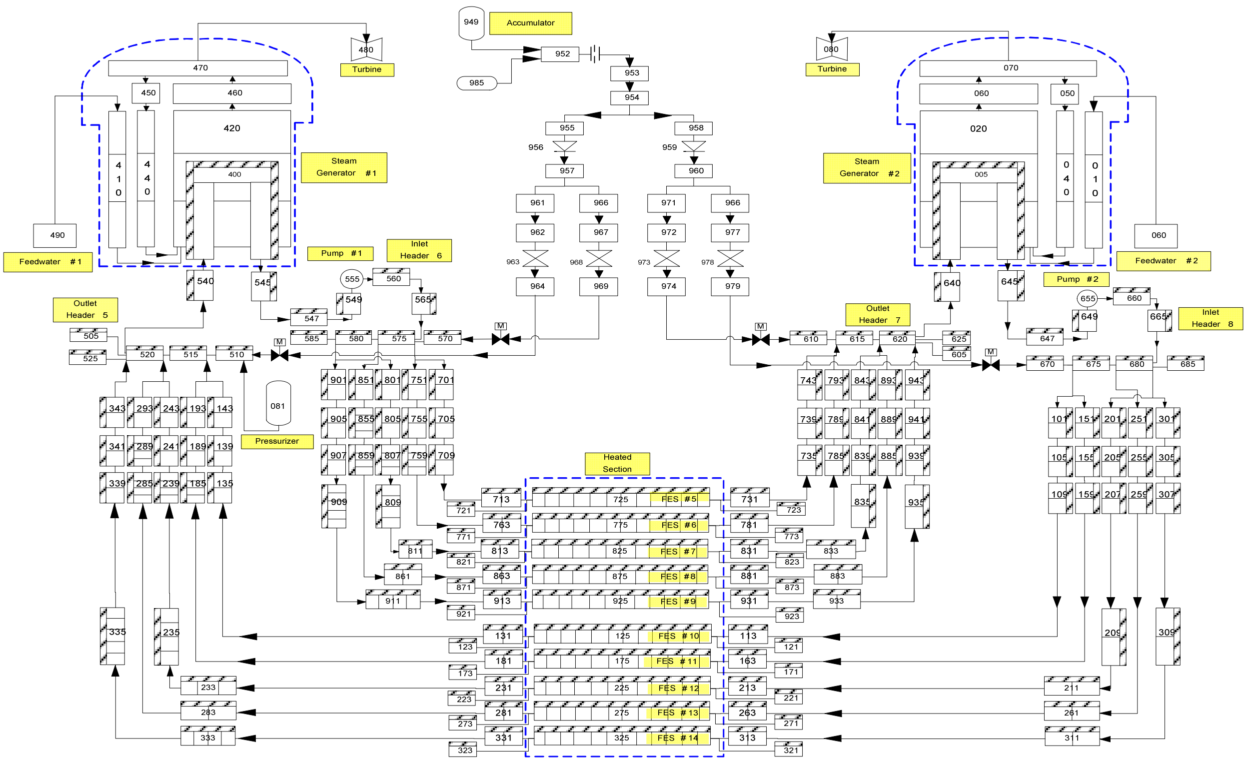

Figure 1 MARS-KS Nodalization for RD-14M Facility

5. Simulation Results

5.1 Steady State and Initial Condition

The steady state simulation is performed to obtain the initial conditions. The most important parameters for steady state are the loop mass flow rate and the differential pressures among the headers. Those parameters were adjusted using junction form loss coefficients. Secondary side initialization is performed using adjusting separator setting parameters and form loss coefficient settings. The calculated values of the major parameters are in good agreement with the experiment data, as shown in Table 1.

Table 1 Selected Variables for Comparison of Steady State

Selected Variables	Variable Description	Exp.	Cal	Var. Unit
ΔP_{P1}	Pump 1 Differential Pressure (DP)	1295.6	1279.5	kPa(a)
ΔP_{P2}	Pump 2 Differential Pressure (DP)	1283.1	1280.4	
$\Delta P_{HD8\text{-}5}$	DP from HDR8 to HDR5	1083.9	1118.3	kPa(a)
$\Delta P_{HD6\text{-}7}$	DP from HDR6 to HDR7	1121.4	1129.1	
P_{HD8}	Header 8 Pressure	11.07	11.096	MPa(a)
P_{HD6}	Header 6 Pressure	11.10	11.101	
P_{HD7}	Header 7 Pressure	9.96	9.972	
Q_{P1}	Pump 1 Discharge Flowrate	18.34	18.34	kg/s
Q_{P2}	Pump 2 Discharge Flowrate	18.2	18.34	
$T_{B1\text{-}IN}$	Boiler 1 Inlet Fluid Temp	301.8	302.69	°C
$T_{B1\text{-}OUT}$	Boiler 1 Outlet Fluid Temp.	260.88	262.93	
$T_{B2\text{-}IN}$	Boiler 1 Inlet Fluid Temp	301.8	302.46	
$T_{B2\text{-}OUT}$	Boiler 2 Outlet Fluid Temp.	262.63	262.93	
T_1	FES Temp.@ middle HS13	322.98	313.99	
T_2	FES Temp.@ inlet HS13	313.61	300.41	
T_3	FES Temp.@ outlet HS13	337.98	326.75	
T_5	FES Temp.@ middle HS8	326.56	315.10	
ΔPHS13	DP Across HS13	824.22	838.06	kPa(a)
Q7	HS5 inlet mass flow	3.39	3.3984	kg/s
Q8	HS8 inlet mass flow	4.26	4.2617	
Q9	HS10 inlet mass flow	3.29	3.3099	
Q10	HS13 inlet mass flow	4.14	4.1476	
Q11	Boiler 1 steam flow	1.93	1.9909	kg/s
Q12	Boiler 2 steam flow	1.91	2.1226	

5.2 Transient

The break was simulated by opening a trip valve to the containment at 11.2 seconds after initial steady state conditions were reached. After the break valve initiation, no power shut down, no ECC were actuated because discharge of inventory is insufficient to make the rapid transient. Primary pump 1 tripped at 1191.8 seconds due to the over voltage process protection trip, but the forced flow was not lost completely. The reduction of the primary flow resulted in rapid temperature excursion in the system. Then, a process protection trip due to high FES temperature interrupts the power supplies at 1336 seconds and it is predicted at 1418 seconds in the simulation. Table 2 shows the sequence of events in B9802 test comparing the simulation results.

Table 2 Major Sequence of Events

Experiment	Simulation	Procedure / Significant Events
t = 0.0 s	0.0 s	Start Scans/Calculation
t = 11.2 s	11.2 s	Break valve opens
t = 1191.8 s	1191.8 s	Pump 1 trip due to over voltage process protection trip is treated as boundary condition.
t = 1336.13 s	1418.0 s	Power supplies tripped on high FES sheath temperature process protection trip

The break occurs at inlet header 8 at 11.2 s, but the primary power and pumps are not ramped down early stage. After the break opened, the primary system pressure rapidly decreased due to the sudden discharge of the inventory mass. After then, the pressure relatively and slowly decreased since the power supply and reactor coolant pumps are still operated. Typically, the calculated header pressure shows a good prediction in all headers, as shown in Fig. 2.

The pump differential pressure and pump discharge flow rate trends are shown in Fig. 3 and Fig. 4. The pump differential pressure shows positive and flow direction is also positive during the transient. The initial differential pump pressures are maintained until the boiler outlet void fraction appears in experiment and calculated results. The differential pump pressure variations are largely depends on the boiler outlet void fraction. Approximately, when the boiler outlet void fraction exceeds 0.1 values in both results, the differential pump pressure began to oscillate. Fig. 5 shows the boiler 2 void fraction time history at inlet and outlet location. In overall, the differential pump pressure, pump discharge flowrates, and boiler void fraction variations are well predicted during the single phase maintained, but the simulated oscillation during the two phase flow exist in the pump is not as big as experiment. The less steam condensation in the steam generators is predicted from 600 seconds to 800 seconds, and the larger steam condensation is predicted after 800 seconds. This disagreement might be arisen from discrepancies of two phase pump modeling and heat removal rate along the steam generator u tube. So the steam condensation and two phase pump modeling should be studied more for accurate code prediction. Large differential pump pressure transient may initiate the over voltage pump process protection trip signal. Pump 1 over voltage trip occurred at 1192 seconds in the experiment and it is simulated as a boundary condition in MARS-KS. The pump 2 differential pressure begins to decrease at 1336 and 1418 seconds in each experiment and simulation by high FES temperature process protection signal.

Fig. 6 shows the comparison between calculated and measured void fraction at inlet and outlet location of heated section 13. The void fraction at outlet locations increased as transient time passed because the system pressure reached to saturation and steadily decreased while the power is still operated. Inlet void fraction is maintained almost zero whole the transient by steam condensation in the boilers and steam separation in the headers. In the experiment, some measured void fraction reaches under zero which is contributed by instrument error. All minus void fraction values considered as zero, calculated void fraction is good agreement with test data. Fig. 7 provides the channel flow rate comparison in channel 10 and 13. The channel flow rate is slowly decreased and it is also maintained normal flow direction during the transient. As explained earlier, the oscillation between 1050 seconds and 1192 seconds is not predicted.

Fig 8 implies the inlet, middle and outlet FES temperature comparison between calculated and measured. In the experiment data, the FES temperature excursions in channel 13 have no change until 1000 seconds when the channel flow rate began to oscillate. The full removal of the FES bundle heat makes the FES temperature maintain since the primary loop flow rate has almost normal circulation. In this range, single phase liquid convection, subcooled and saturated nucleate boiling heat transfer are predicted as shown in Fig 9. After 1000 seconds, the FES temperature began to be fluctuated due to coolant flow oscillation. Although the oscillation of outlet FES temperature is not captured in the simulation, overall transient is well predicted. The sudden increase of FES temperature appeared around 880 seconds in the simulation, at the same time the heat transfer model is changed from nucleate boiling mode to film boiling mode. The FES temperature at middle location reached to outlet FES temperature level around 1160 seconds due to same reason as outlet FES temperature increase, but inlet FES temperature still maintained as initial values. On the other hand, the measured inlet and middle FES temperature shows same transient with similar amplitude which much lower than outlet FES temperature. The bundle effects such as parallel flow, cross flow, and mixing in the horizontal channel could be caused of different heat transfer model prediction. Therefore, further study on the heat transfer model prediction considering bundle effect of a large diameter long horizontal pipe is required.

For a sensitivity study on the channel node number, the simulation results with 12 nodes and 6 nodes are compared. The FES temperature of channel 13 resulted from 12 nodes is more closed to measured data as shown Fig. 10. The FES temperature transient after the post dryout is not predicted in a small number of nodes but it is captured in a large number of nodes case.

Because the break size is considerably small, two phase chocked flow at the break is generated about 1000 seconds, and it is maintained until power trip initiation. So, the sensitivity study on two phase discharge coefficient effect carried out for MARS-KS calculation sensitivity. Fig. 11 implies the calculation results in case four different two phase discharge coefficient entries, 0.5, 1.0, 1.4, 1.9. The outlet FES temperature comparison shows the same trend during single water discharged because two phase discharge coefficient applies only to two phase choked flow calculations. The larger two phase discharge coefficient predicts the rapid increase of outlet FES temperature and the faster power trip initiation due to more void generation.

The break flow rate comparison is shown in Fig. 12. Once the break valve was opened, the liquid was discharged until the header 8 pressure reached to saturation, after then it was changing to two phase flow. The measured trend shows the continuously decreased until when the reactor trip is actuated. The prediction of the break flow rate is very close to the experiment whole the transient.

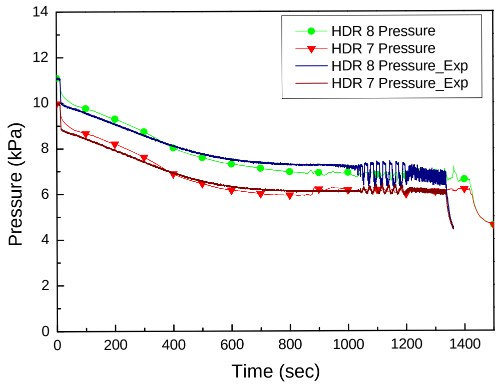

Figure 2 Header Pressure

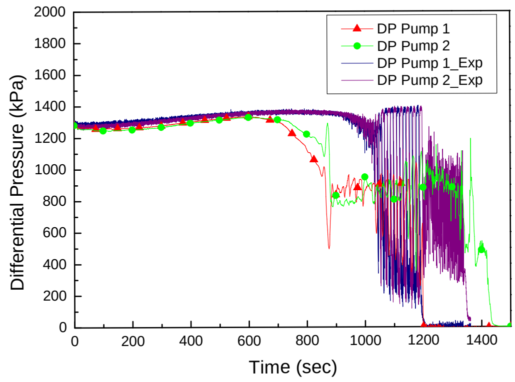

Figure 3 Pump Differential Pressure

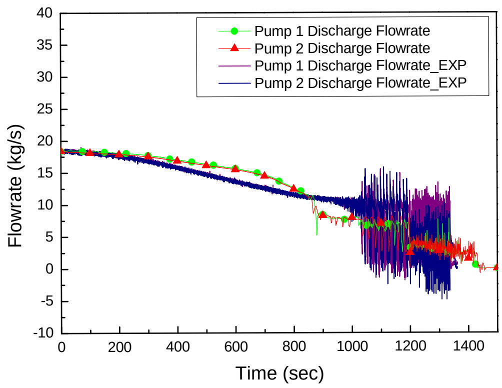

Figure 4 Pump Discharge Flowrate

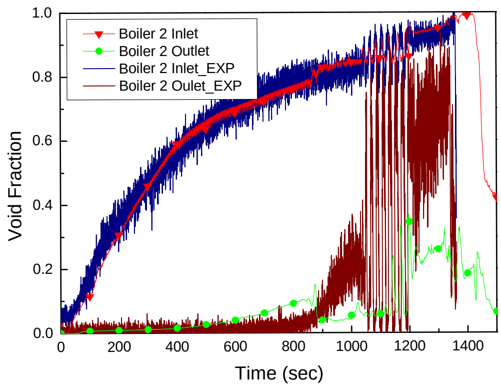

Figure 5 Boiler 2 Void Fraction

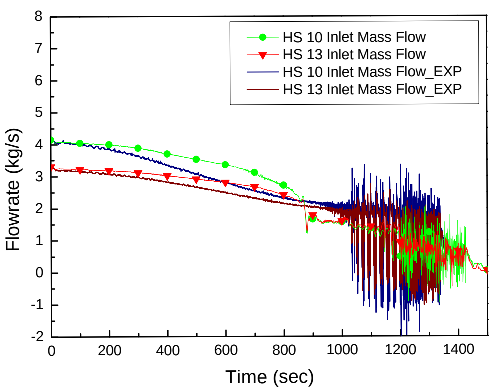

Figure 6 Channel 10 & 13 Flowrate

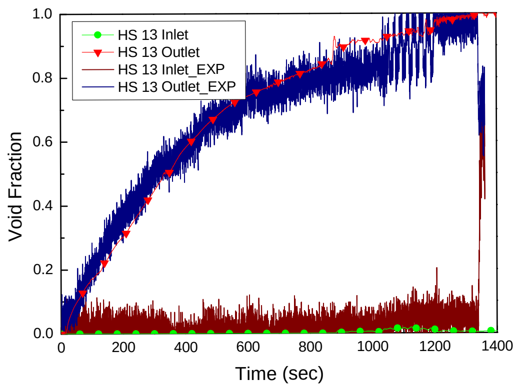

Figure 7 Channel 13 Void Fraction

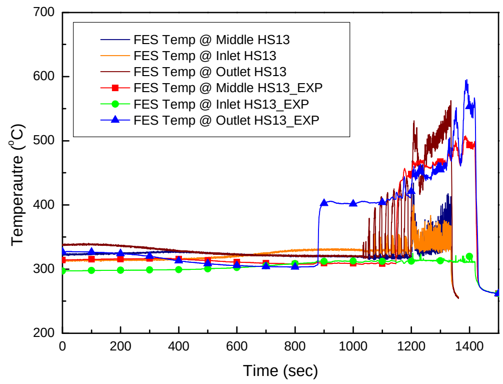

Figure 8 Channel 13 FES Temperature

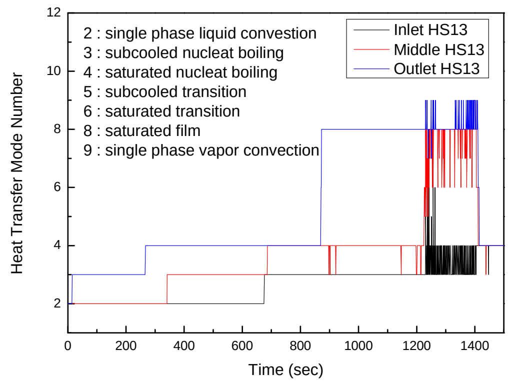

Figure 9 Channel 13 Heat Transfer Mode

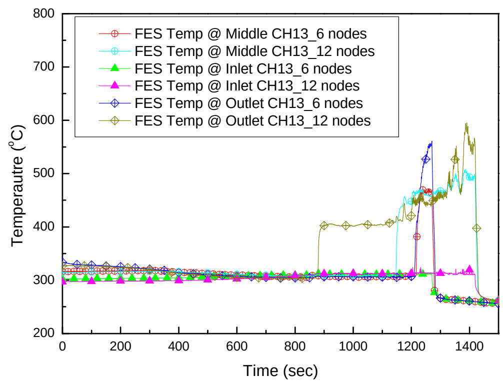

Figure 10 Channel 13 FES Temperature Comparison vs. Node Number

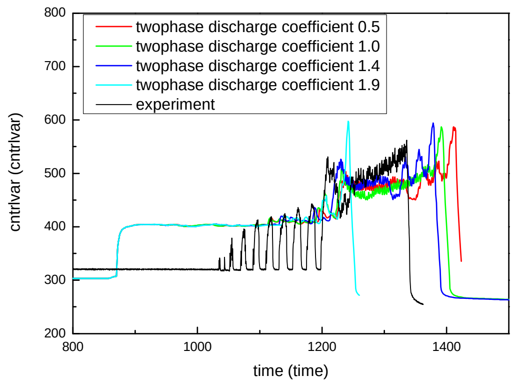

Figure 11 Channel 13 Outlet FES Temperature Comparison vs. Two Phase Discharge Coefficient

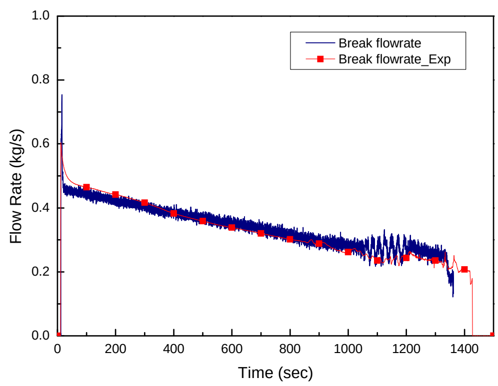

Figure 12 Break Discharge Flowrate

6. Conclusion

The capability of MARS-KS code to simulate the important phenomena of CANDU reactors during the operation and accident condition was assessed. Through the simulation results of B9802 SBLOCA test, MARS-KS code reasonably predicts main thermal hydraulic behaviors such as the primary pressure, temperature, mass flow rates, void fraction, outlet FES temperature, etc. during the steady state condition and overall transient. The calculated FES temperature trend at middle region of the channel, and steam condensation in the steam generators show inconsistency in comparison with test data. Therefore, a further study on the heat transfer model prediction considering bundle effect in the large diameter long horizontal pipe, and condensation model along the u tube is required to have more accurate code prediction.

7. ACKNOWLEDGMENTS

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8. REFERENCES

[1] " RD-14M Small-Break LOCA Experiments for IAEA International Collaborative Standard Problem ", 153-108210-440-001 Revision 0, Atomic Energy of Canada Limited(2007)

[2] B.D.Chung, et al, " Development of Best Estimate Auditing Code for CANDU Thermal Hydraulic Safety Analysis ", KINS/HR-293, Korea Institute of Nuclear Safety(2003)

[3] Thermal Hydraulic Safety Research Department, " MARS Code Manual Volume II, Input Requirement ", KAERI/TR-2811, KAERI(2004)

[4] Y.J.Cho, et al, "Assessments of RELAP5/MOD3.2 and RELAP/CANDU in a Reactor Inlet Break Experiment B9401 of RD-14M", Journal of the Korean Nuclear Safety, Volume 35, Number 5, pp. 426~441(2003)