Proceedings of the Korean Nuclear Society Autumn Meeting
Taejon, Korea, October 2000

A Preliminary Results of B9401 Multi-channel RIH Break Experiment using RELAP5/MOD3 with RD-14M Test Facility

Yong Jin Cho, Young Seok Bang, and Sukho Lee
Korea Institute of Nuclear Safety
P.O. Box 114, Yusong, Taejon, 305-600, Korea

Gyoo Dong Jeun
Nuclear Engineering Department, Hanyang University

ABSTRACT

The RELAP5 computer code has been developed for best-estimate transient simulation of pressurized water reactors and associated systems, but it has not been completely assessed for a CANDU reactor. A recent studies by S. Lee, et.al. has been initiated with an aim to identify the code applicability for all the postulated transients and accidents in CANDU reactors and suggested that the RELAP5 could be applicable to simulate the transients and accidents in the CANDU reactors. Nevertheless it was indicated that there were some works to be resolved, such as modeling of headers and multi-channel simulation for the reactor core, etc. In the present study, a Test B9401 conducted in the RD-14M test facility was simulated with RELAP5/MOD3.2.2 code. The RELAP5 results were compared with experimental data. The RD-14M provides five 7-element heated sections per pass instead of a single heated setion of the RD-14. Therefore the effect of multi-channels could be observed in the RD-14M test. The RELAP5 analyses demonstrated the code's capability to predict the major phenomena occurring in the transient, both qualitative and quantitative points of view. However, some discrepancies after emergency coolant injection, the behaviors of the ECI mass flowrate and the sheath temperatures were observed.

I. INTRODUCTION

In Korea, four CANDUs have been operated in Wolsong site. Wolsong unit 1 had been operated since 1983 and others since 1997, 1998, 1999 respectively. In Canada, the effectiveness of emergency core cooling system (ECCS) have been considered as "generic safety issues" identified by the Canadian regulatory body, CNSC, as being applicable to all or most of the CANDU nuclear power plants in Canada. To provide information on the effectiveness of ECCS in a CANDU reactor, various series of experiments has been carried out in the RD-14 pressurized water loop at the Whiteshell Nuclear Research Establishment from 1984 to 1987. As a following experimental facility, the RD-14M had been constructed and operated since 1988.

In the present study the RELAP5 code, which have been used worldwidely in the PWRs, was chosen to identify its applicability in the CANDU reactor. The RELAP5 code has been developed for best-estimate transient simulation of pressurized water reactors (PWRs) and associated systems. The model is based on a non-homogeneous and non-equilibrium model for one-dimensional, two-phase system is solved by a fast, partially implicit numerical scheme to permit economical evaluation of system transients. The RELAP5 has been used world-widely and modified continuously for PWR transient simulation. Since the RELAP5 has been developed to simulate the steady and transient behaviors of the PWRs, most of models and correlations are based on the vertical channels (i.e. PWRs). Moreover, since the RELAP5 has a one-dimensional numerical solution system, it is difficult to simulate properly the thermal-hydraulic behaviors of the horizontal multi-channels of a CANDU reactor. However, in the recent versions of the RELAP5, a horizontal volume flow-regime map has been developed, which consists of bubbly, slug, annular mist, dispersed (droplets or mist), and horizontally stratified regimes. Transition regions are included discontinuities going from one correlation to another one in drag and heat and mass transfer.

There has been an effort to identify the RELAP5 code capability for the use in the CANDU reactors, but it has not been fully assessed for CANDU reactors. A previous work performed by S. Lee [1,2] showed that the RELAP5 could be applicable to assess the transients and accidents in the CANDU reactors. However, it was indicated that there were some works to be resolved, such as modeling of headers and multi-channel simulation for the reactor core, etc.

In this study, the multi-channel experiment B9401 was analyzed preliminarily using RELAP5/MOD3.2 [7] and the analysis result was compared with the experiment results. The test B9401 was chosen to identify the RELAP5 capability to simulate the multi-channel behavior during the transient.

II. RD-14 TEST FACILITY AND EXPERIMENTAL PROCEDURES

A. Facility Description [3]

RD-14 was designed and constructed starting 1981. Due to funding limitation, the RD-14 reference design chosen was two, 5.5 MW, 37-element channels, (i.e., one channel per pass), with 1:1 scaling of vertical distances throughout the loop. This determined the sizing of piping and various components (e.g., steam generators, pumps, headers). The values for various loop parameters dictated by the choice of reference design were 5.5 MW maximum thermal power per pass, 590 kW/m maximum surface heat flux per pass and 24 kg/sec rated flow rate (one 37-element channel).

The modification of RD-14 to RD-14M provides for the study of the interaction of multiple heated channels in parallel in a full height loop. As multiple channel, five 7-element heated sections per pass were chosen to replace the single, 37-element channel. The cross sectional area of the associated below header pipe-work was scaled at 7:37 to preserve heat and mass fluxes in the multi-channel facility.

As noted in reference [3,4,5], the large number of non-dimensional groups to be considered precludes the scaling of two-phase flow dynamics with complete similarity. However, if the model is made of a similar solid material and has a similar fluid under the same system pressures as the prototype, scaling is simplified. Reference [3,5] presents an appropriate set of similarity criteria to be used under such conditions. Using 1:1 scaling of vertical elevations and axial lengths simplifies the scaling of the facility. It is appropriate to choose the piping diameters such that the flow velocities will be scaled 1:1. This ensures that the characteristic transit times will be approximately equal in both the facility and the reactor.

In RD-14M, consideration was given to the several experimental program in the design of the loop, the loop peripherals and the loop instrumentation. The experimental programs were categorized into three groups, safety-type transients, process dynamics and control-type transients, and component-type transients.

B. Experimental Procedure [3]

A series of experiments to investigate the thermal-hydraulic consequences of critical break with emergency coolant injection is in progress in the RD-14M test facility. The experiment used in this study is B9401 experiment – 30 mm inlet header break experiment with high pressure pumped emergency coolant injection.

The nominal initial conditions for the first experiment in this series, B9401, were 10.0Mpa(g) outlet header pressure, 4.0MW per pass nominal input power, 4.4 Mpa(g) steam pressure, and 186^{o}C feed water temperature.

Before the experiment, the loop was evacuated, filled and degassed, all instrument lineswere vented, and instrument readings were checked and adjusted. The loop was warmed using low power and reduced pump speed. Input power and pump speed were then increased to bring the loop to the desired steady-state single phase starting conditions. The output from all instruments was then scanned and printed as a final check. Then data gathering started. The detailed sequence of events during the experiment was described in Table II.

A programmable pump-speed controller was used in some experiments to simulate pump rundown following a loss of class-IV power. The pump began ramping down at 12s. Cold water was injected into the loop when the primary pressure fell to or below the emergency coolant injection (ECI) pressure. The isolation valves at the ECI pipes to all four headers were opened as soon as the pressure in header 7 fell below 5.5 MPa. As long as the pressure in any header was above 5.5 MPa (pressure in the ECI tank), no ECI water entered that header. When the pressure in any header was below 5.5 MPa, ECI water entered the header at a rate determined by the pressure difference between the ECI tank and the header.

The actual flow rate of ECI is determined by the size and location of the break. Orifices in the injection lines provide scaled simulation of reactor injection flow rate. The high-pressure injection may be from the ECI tank at high pressure, or from the ECI tank at low pressure via corresponding pumps. In either case, the high-pressure ECI water is delivered to the ECI system at approximately constant pressure during the transient. However, as the pressure in any header varies, so does the ECI flow rate into a particular header.

The heated sections are protected from overheating by high-temperature interlocks. if the heater sheath temperature exceeds the set point selected by the loop operator, the heated section power supplies are shut down.

III. RELAP5 SYSTEM MODEL

System model for RELAP5 calculation is shown in Figure 1 and 2, which is basically similar ones found in CATHENA model [8-10] and therefore may help reduce the effect of nodalization. The system model composes of primary heat transport system including heaters and pumps, secondary system, ECI system, accumulator, and break model.

IV. RELAP5 ANALYSIS RESULTS AND DISCUSSIONS

In this analysis, the calculation was only performed for base case. This base case means that almost all of options uses standard or recommended in RELAP5 manual and standard CATHENA nodalization was used without any modifications.

<u>Header Pressures</u>

Figure 4 shows the header pressures and the break location was located in inlet header 8 Experiment started at 10 seconds as the p14 valve opened and RCP (Reactor Coolant Pump) and reactor trip occurred at 12 seconds. After the break initiated at that time, the primary system pressure rapidly decreased as the inventory lost. Due to void generation, the slope of the depressurization rate decreased and few seconds later depressurization rate recovered as the ECI injection delivered into the HTS.

In view of break flow, B9401 experiment did not measure the break flow, and the pressure behavior was only clue to judge whether the break flow was correctly calculated or not. Generally, break flow quality could vary according to the upstream conditions and depressurization characteristic through the break piping. Initially, the break flow was liquid single phase and the inventory loss was larger than other phase. As primary heat transport system pressure reduced and the vaporization was occurred, the break flow had vapor. As the void fraction of break flow increases, the break mass flow rate decreases due to decreasing mass flux.

RELAP5 predicted header pressure slightly higher than the experiment during the period after depressurization. Before the emergency coolant injection (ECI), the pressure transient was correctly predicted during short period, but after the initiation of ECI, the pressure decrease rate was reduced. After that, the calculated was slightly higher than the experiment, as shown in figure 4. One of these differences might be the smaller break flow after the initial rapid depressurization. The sensitivity study of break modeling should be studied including the modeling of downstream of the break.

Emergency Coolant Injection

In RD-14M and CANDU NPP, the ECI coolant delivered into each headers and the coolant could cool the heater section. ECI injection in RD-14M was actuated when header 7 pressure decrease below 5.5 Mpa. After initiation of break at header 8, the header 7 pressure continuously decreased under 5.5Mpa at 26 seconds. The calculated ECI Flow well predicted, but the difference was shown during the initial high pressure emergency injection period (~116 seconds). After the injection was finished, the calculated ECI flow rate had big differences. But this kind of behavior resulted from the piping of ECI system. After the end of ECI injection, the residual coolant in ECI piping showed the oscillatory behavior. Because the ECI valves in each header connection piping were closed at 350 seconds, the behavior could not be shown in results.

Related to the heat transport system (HTS) pressure behavior, the depressurization rate recovered after the initiation of the ECI (at 26 seconds) which collapses the generated void. The RELAP5 predicts broken header pressures well during blowdown period, while it overpredicts them during ECI period. These discrepancies might be arisen from the complicated effects, such as header model itself, amount of ECI flowrate and the predictability of steam condensation, etc.

FES (Fuel-Element-Simulator) Sheath Temperatures of Heated Section

In experiment, the stratification in header did not occur, and the comparison among channels might be meaningless. The results showed the differences only depends upon the channel power. Figure 6 shows the fuel element sheath temperature in each channel. It is shown in figures of channel 8 and 13 which were the most highest power channel. In these figures, the experimental data were divided into three groups, top, middle, and bottom like figure 3.

In channel 8, upstream of the break, the calculated results show several differences. RELAP5 underestimated peak of sheath temperature near 200 seconds, but in other periods, RELAP5 can predict well. In the case of channel 13, downstream of the break, different phenomena were occurred. In experiment, two peaks were shown in figure 6, such as initial peak, and later peaks. RELAP5 extremely underestimated the initial peak. The later peak can be seen in the top sets of experimental data but there were no later peaks in the other experimental data set. These sheath temperature behaviors are resulted from the characteristics of horizontal channel. Fuel rod located in the top uncovered in early phase and the uncovered duration also relatively longer than that in the middle and bottom.

Eventually, this kind of deficiencies resulted from the lack of CANDU specific model, such as horizontal channel model, header model, etc.

V. CONCLUSIONS

Test B9401,30mm inlet header break LOCA, in the RD-14M multichannel facility have been performed with the RELAP5/MOD3.2.2 with an aim to identify its applicability to simulate the multi-channel effects in CANDU and the analysis results were in comparison with the experimental results. The RELAP5/MOD3 predicted reasonably the main phenomena occurring in the transient. The general conclusions from the present work are summarized as follows:

1) The RELAP5/MOD3 predicted reasonably well thermal-hydraulic behaviors in the inlet header break tests, particularly multi-channel. However, some discrepancies were observed after the ECI. Pressure transient in the broken header was overpredicted after the ECI. This might be arisen from the complicated effects, such as header model itself, amount of ECI flowrate and the predictability of steam condensation.
2) Pressure differences between headers govern the flow characteristics through the heated sections, particularly after the ECI. In determining header pressure, there are many uncertainties arisen from the complicated effects as mentioned above. Therefore, it would be concluded that further works are required to reduce these uncertainties, and consequently predict appropriately thermal-hydraulic behaviors in the reactor coolant system during LOCA analyses.

Besides the above assessments, the RELAP5 sensitivity study on the break model, choking model at the junctions, etc. and the analysis using RELAP5-CANDU version [11] are under study. Issues identified from the present analysis will be examined and in particular the model development of the multi-channel analysis will also be performed in progress.

REFERENCES

[1] S. Lee and I.G. Kim, "RELAP5 Simulation of Thermal-Hydraulic Behavior in a CANDU Reactor," J. Nuclear Technology, Vol. 130, pp.18-26,April(2000).

[2] S. Lee and M. Kim, "RELAP5 Simulation of the Small Inlet Header Break Test B8604 Conducted in the RD-14 Test Facility ", KNS Spring Meeting, Pohang, Korea, May(1999).

[3] R.S. Swartz, et al. "An RD-14M Experiment for the Intercomparison and Validation of Computer Codes for Thermal-hydraulic Safety Analyses of Heavy Water Reactors," RC-2491 June 2000.

[4] R.S. Swartz, "An RD-14M Experiment B9401 Data Set," CD ROM June 2000.

[5] P.J. Ingham, F.W. Barclay and J.B. Hedley, "Large Outlet Header Break Experiments in a CANDUTH-Type Heat Transport Loop," 14th Annual Nuclear Simulation Symposium, Pinawa, Manitoba, Canada, April 25-26 (1988).

[6] V.S. Krishnan and P. Gulshani, 'Thermosyphoning Behavior of a Pressurized-Water Facility with CANDU-PHTS Geometry," 2nd Intl Topical Mtg on NPP T/H and Operations, Tokyo, April (1986).

[7] V.H. Ransom, "RELAP5/MOD3 Code Manuals, Vol. I-V", EG&G Idaho, June (1990).

[8] B.N. Hanna, "CATHENA MOD-3.2n, Theoretical Manual", THB-CD-002, AECL-WL, (1989).

[9] J.P. Mallory,"CATHENA Idealization of the RD-14 Test facility", RC-54-1, AECL (1988).

[10] B.N. Hanna and T.E. MacDonald, "CATHENA Idealization Documentation of the RD-14 Test facility, Secondary Side Characterization Tests", RC-54-2, AECL (1988).

[11] B.D.Chung, et al, "Development of Best Estimate Auditing Code for CANDU Thermal Hydraulic Safety Analysis," KINS/HR/293, March, 2000

Table I Comparison of Characteristics of RD-14 and CANDU reactor

Parameters	RD-14	RD-14M	Typical Reactor
Operating Pressure (MPa)	10	10	10
Loop Volume (m^3)	0.95	1.01	60.
Heated Sections:	37-rod bundles	7-rod bundles	37-element bundle
Number per pass	1	5	95
Length (m)	6	6	12 x 0.5
Rod diameter (mm)	13.1	13.1	13.1
Flow tube Dia. (mm)	103.4	44.8	103.4
Power (kW/channel)	5500.	3x750, 2x950 per pass	5410.
Pumps:	single stage	single stage	same as RD-14
Impeller diameter(mm)	381	381	813
Rated flow (kg/s)	24.	24.	24. (max/channel)
Rated head (m)	224.	224.	215.
Specific speed	565.	565.	2000
Steam Generators:	recirculating U-tube	recirculating U-tube	recirculating U-tube
Number of tubes	44	44	37/channel
Tube diameter I.D.(mm)	13.6	13.6	14.8
Secondary heat-transfer area (m^2)	41	41	32.9/channel
Secondary Volume (m^3)	0.9	0.9	0.13
Heated Section-to-Boiler Top Elev. Difference (m)	21.9	21.9	21.9

Table II. B9401 test (30mm inlet header break) procedure

Experiment Time	Event Description
0	start data gathering
10	open break valve, p14 start
12	step input power to decay level & RCP ramped down
20.6	ECI isolation valve open
22.8	pressurizer tank (surge tank) isolated
116.2	HP ECI terminated, LP ECI start
213.2	primary pumps off
229.2	scan stopped
231	scan start
350.7	LP ECI terminated
460	scan stopped
463	scan start
692	scan stopped
695	scan start
924	scan stopped

RD-14M Experimental Facility Nodalization for RELAP5/MOD3

(Heat Structure 는 표시안함.)

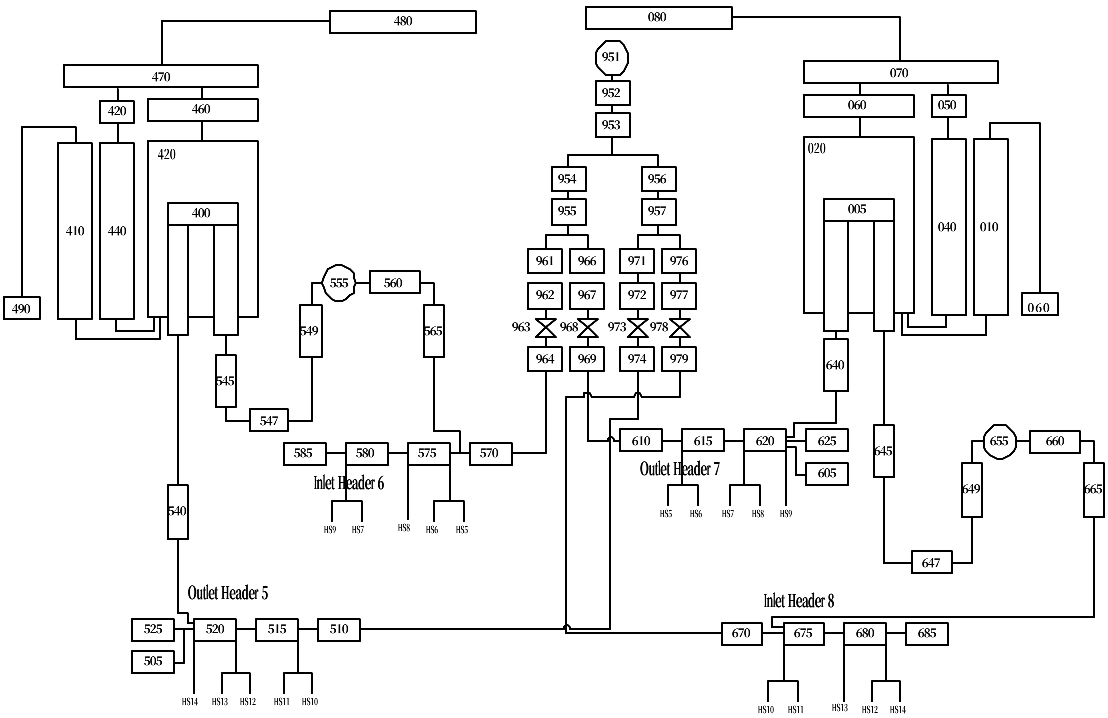

Figure 1. RD-14M Nodalization using RELAP5

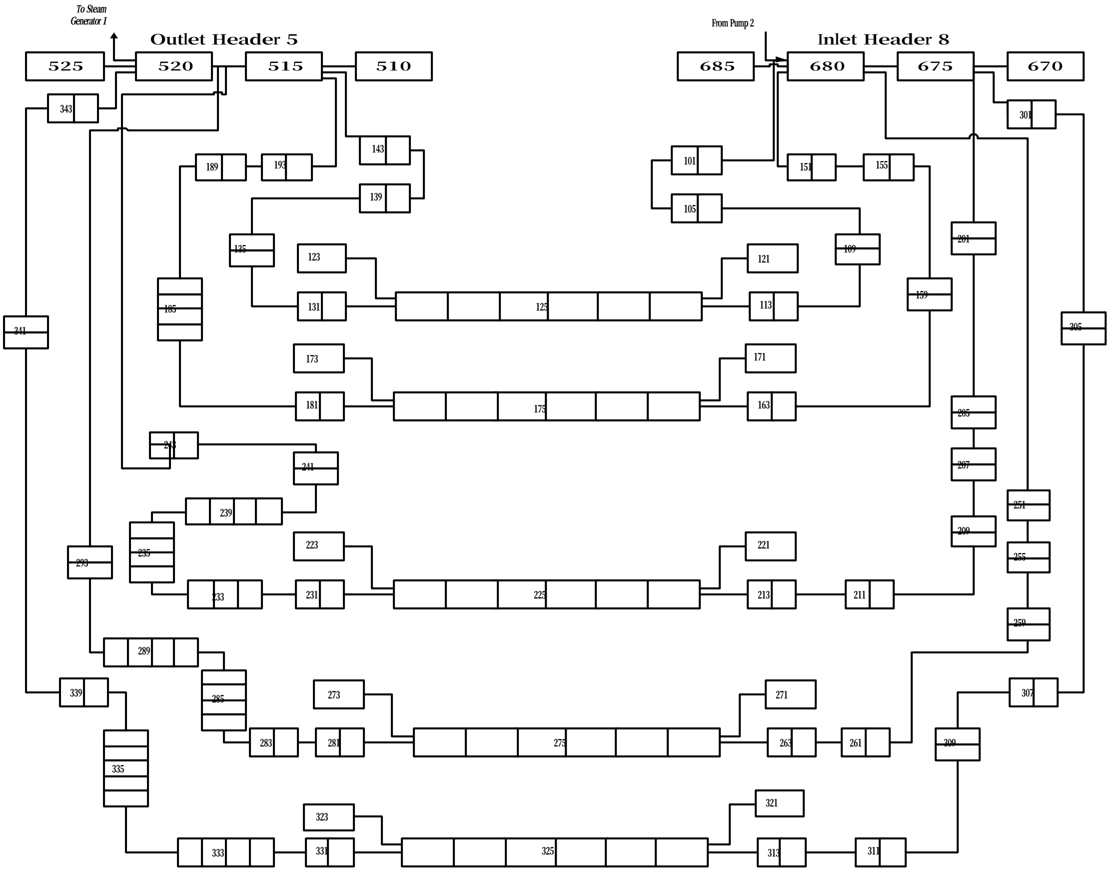

Figure 2. RD-14M Nodalization below Headers using RELAP5

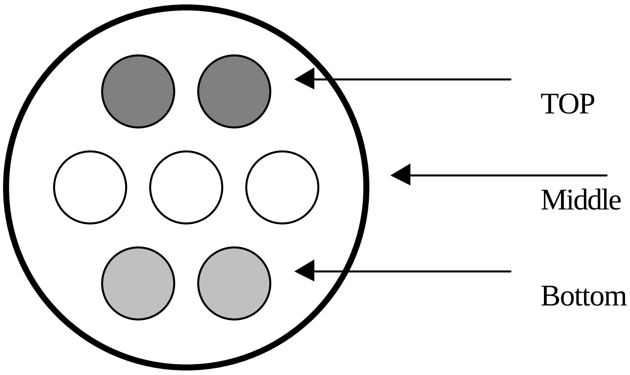

Figure 3 Shape of Fuel Channel Geometry

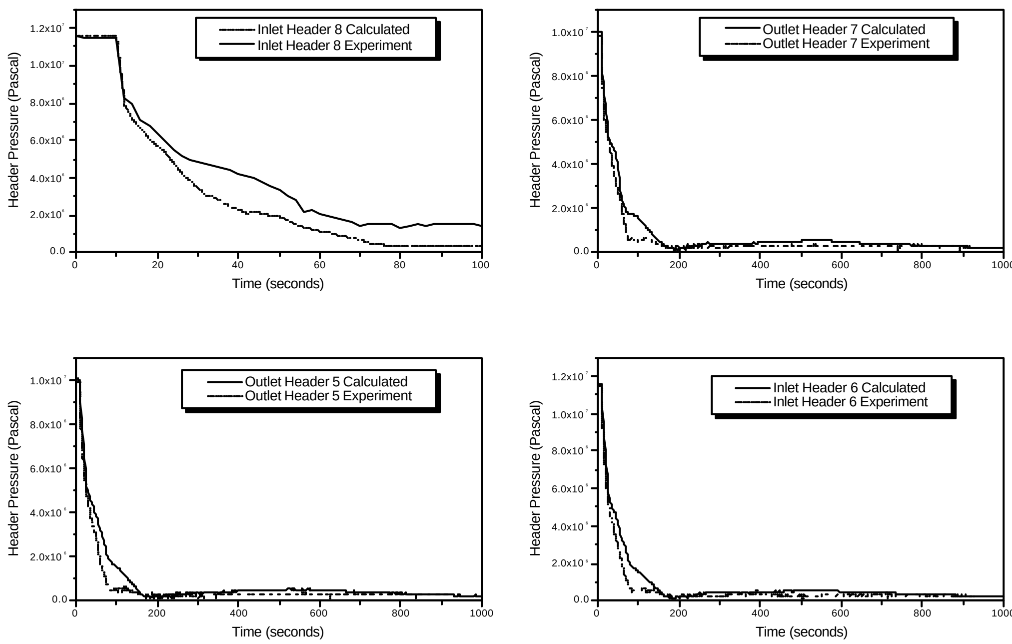

Figure 4. Header Pressure

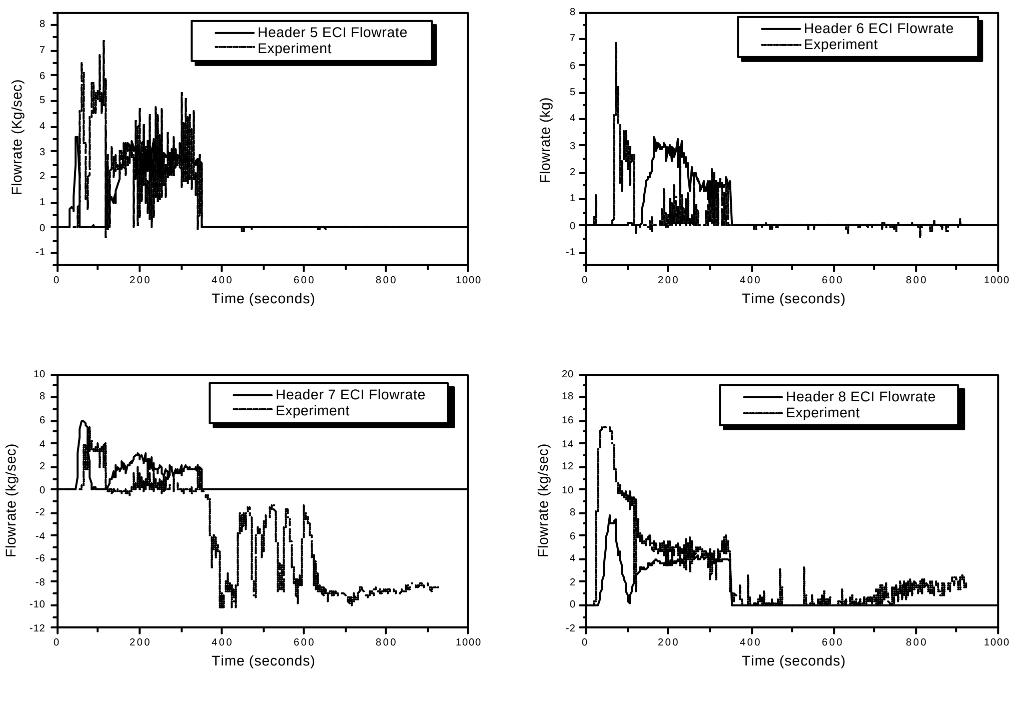

Figure 5 ECI Header Mass Flowrate

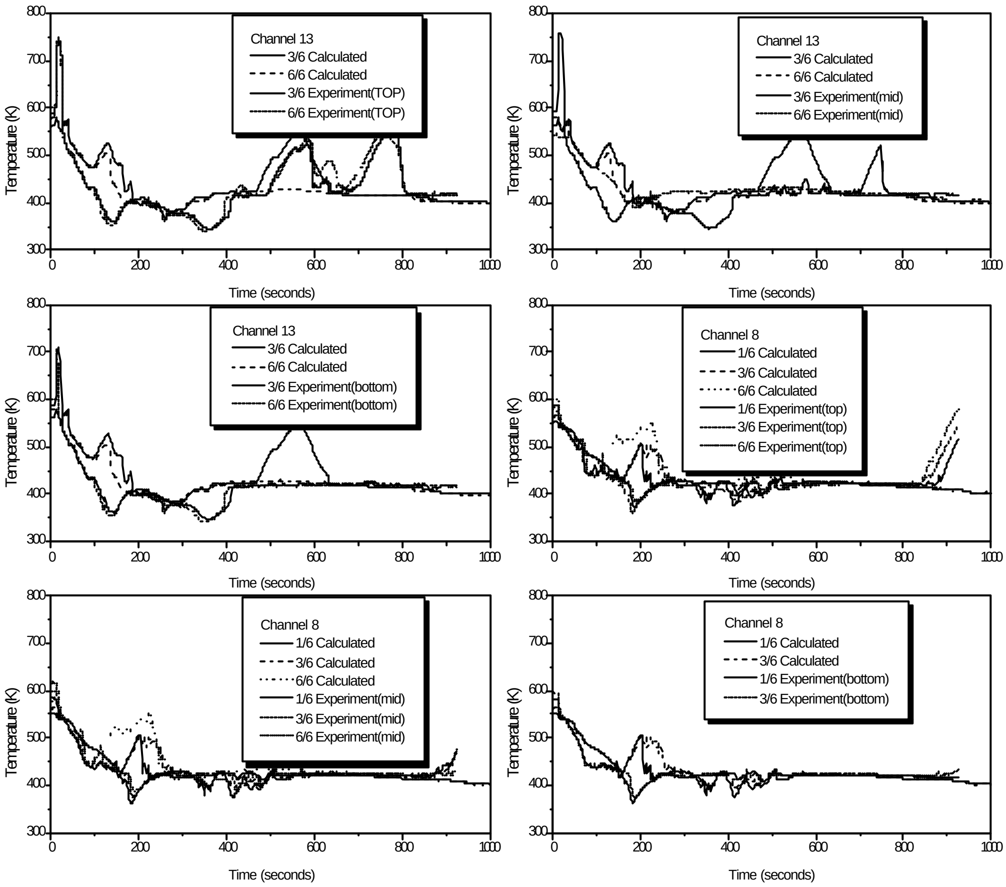

Figure 6. Test Section Fuel Element Simulator Sheath Temperature