FLOW OSCILLATION SIMULATION WITH CATHENA

L. Sun
Point Lepreau Generating Station, PO Box 600, Lepreau, NB, Canada E5J 2S6

Abstract

Flow oscillations may occur in some postulated accident events in CANDU 6 reactors, which may influence the timing of reactor trip and occurrence of fuel sheath dryout. Therefore an accurate prediction of the characteristics of the flow oscillations is necessary from a reactor trip coverage perspective. Flow oscillations in the heat transport system of CANDU 6 reactor due to a single pump trip have been predicted by different codes. However, the predicted results are not always consistent. To make certain that CATHENA code is capable of predicting flow oscillations in some postulated accidents events, a commissioning test performed at Point Lepreau Generating Station in 1983 is simulated using CATHENA Plant Model. The simulation results indicate that CATHENA code is able to capture the characteristics of the oscillations caused by initial perturbation of the heat transport system though the amplitude of the oscillation is much smaller than that cause by single pump trip. The sensitivity of the results to the maximum time step is also examined. This simulation of a reactor commissioning test may provide confidence in the capability of CATHENA in simulating flow oscillation phenomena.

1. Introduction

The technical details of the CANDU 6 reactors can be found elsewhere (Reference [1]). The following descriptions on CANDU 6 reactors are extracted from Reference [1].

In CANDU 6 reactors the system to remove nuclear reaction heat is called Heat Transport System (HTS) as shown in Figure 1. The HTS circulates pressurized D_2O coolant through the fuel channels to remove the heat produced by fission in the nuclear fuel. The coolant transports the heat to steam generators (boilers) where it is transferred to light water to produce steam to drive the turbine. Two parallel HTS coolant loops are provided in CANDU 6. The heat from half of the 380 fuel channels in the reactor core is removed by each loop. Each loop has one inlet and one outlet header at each end of the reactor core. D_2O is fed to each of the fuel channels through individual feeder pipes from the inlet headers and is returned from each channel through individual feeder pipes to the outlet headers. Each HTS loop is arranged in a ‘Figure of 8’ format, with the coolant making two passes, in opposite directions, through the core during each complete circuit, and the pumps in each loop operating in series. The coolant flow in adjacent fuel channels is in opposite directions.

The pressure in the HTS is controlled by a pressurizer connected to the outlet headers at one end of the reactor. The pressure in the outlet headers is measured and compared to a setpoint preset in a control program. The pressure can be controlled by turning on the variable heaters in the pressurizer and opening of the pressure control valves installed on the pressurizer.

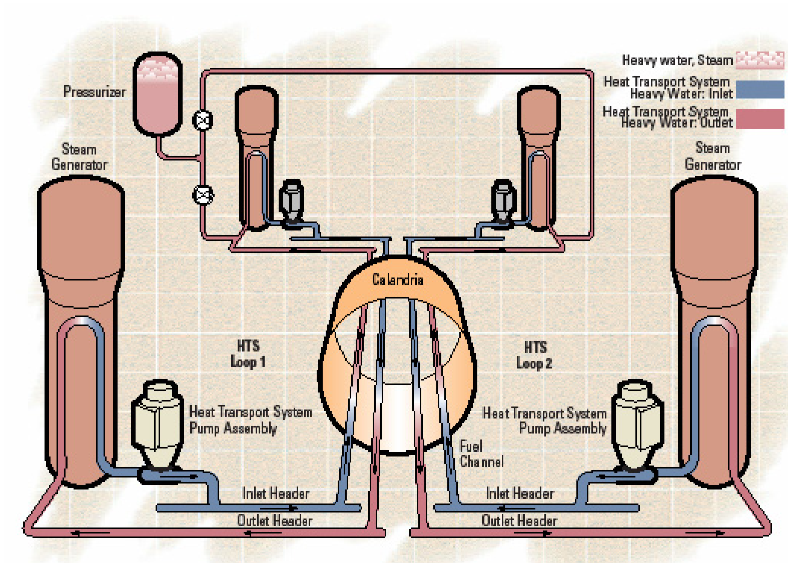

Figure 1: Demonstration of heat transport system for CANDU 6 reactor (from Reference [1])

In some postulated accident events in CANDU 6 reactors flow oscillations may occur, which may influence the timing of reactor trip and occurrence of fuel sheath dryout. Therefore an accurate prediction of the characteristics of the flow oscillations is necessary from a reactor trip coverage perspective. A schematic of major components in one of the loops in the HTS of Point Lepreau Generating Station (PLGS) in 1983 is shown in Figure 2. The other loop has same structure.

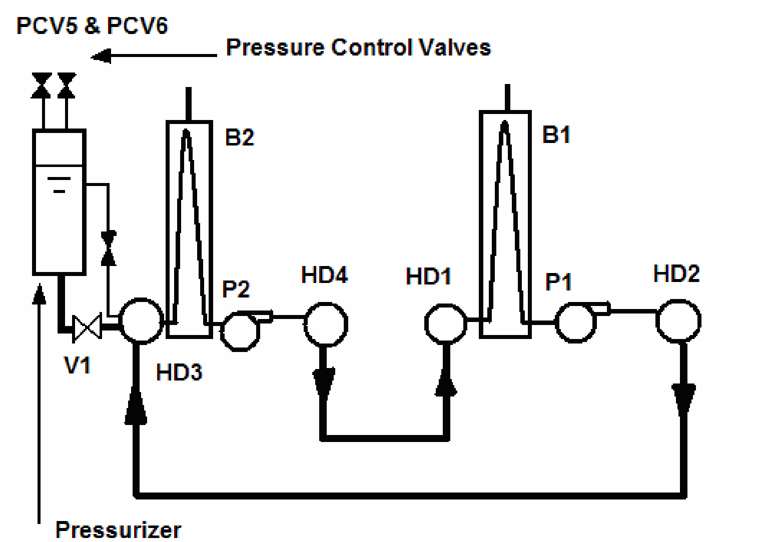

Figure 2: Schematic of Loop 1 of heat transport system in PLGS reactor in 1983

The out-of-phase oscillations of flow through each pass of the affected loop have been observed in single pump trip experiments (References [2]) and were also predicted by both homogeneous fluid codes and two-fluid codes for CANDU 6 rector in single pump trip event. However, under the same conditions, some codes predict a divergent flow oscillation, i.e., the oscillation amplitude increases with time, while other codes predicted convergent flow oscillations, i.e., the oscillation amplitude decreases with time and the oscillation periods are also different. The objective of this work is to prove that CATHENA is capable of predicting the flow oscillation phenomena appropriately which might occur in some accident conditions and to confirm the impact of the applied time step on flow oscillation simulation for CATHENA code.

2. CATHENA and LEPCON codes

CATHENA is a two-fluid thermalhydraulic computer code, developed by AECL for analyses of flow transients in reactors and piping networks (Reference [3]). Although CATHENA was developed primarily for thermalhydraulic simulations of postulated accident scenarios in CANDU reactors, the code has been used for a wide range of thermalhydraulic applications. CATHENA code uses a one-dimensional, non-equilibrium, and two-fluid thermalhydraulic representation of two-phase flow. Conservation equations for mass, momentum and energy, together with flow-regime dependent constitutive relationships describing the interfacial transfer of mass, momentum and energy are solved for each phase (liquid and vapor). The code uses a staggered-mesh, semi-implicit, finite-difference numerical solution technique to solve the conservation equations. The CATHENA code includes component models such as pumps, valves, pipes, generalized tank models, and has extensive control modelling capability. It also contains a GENeralized Heat Transfer Package (GENHTP) allowing the modelling of solid components such as pipes and fuel pins, including heat generation within these components and heat transfer between these components and the surrounding fluid. In this work, all the default heat transfer correlations were applied in the simulations.

LEPCON code (Lepreau Control Program) simulates the response of the relevant Point Lepreau plant control programs including Boiler Level Control, Boiler Pressure Control, and Pressure and Inventory Control. LEPCON Version 0.0 (Reference [4]) is used in this analysis. Compiling and linking the two codes generated the CATHENA/LEPCON executable. An interface has been built in CATHEANA working file for CATHENA code and LEPCON code to interact with each other. The CATHENA code performs thermalhydraulic simulations based on the working file to determine the plant parameters and output the control parameters to LEPCON working file. Then LEPCON working file will adjust the status of controlled instrument such as valve opening fraction or the status of the heaters in the pressurizer and etc. The changed parameters of these instruments will be fed back to CATHENA working file which will update the thermalhydraulics simulations based on the updated instrument configuration.

3. Methodology

In the single pump trip tests performed in operating commercial CANDU reactors, no flow oscillations have been observed in the primary heat transport system since the reactor will trip before any oscillation develops, e.g., in the pump rundown tests at Point Lepreau Generating Station, the reactor tripped quickly after the pump was tripped. However, there exist other tests showing flow oscillations caused by initial perturbations other than a pump trip. Since they are all density wave oscillations, it is expected that the oscillations in these tests will provide some common characteristics for oscillations caused by different initial perturbations.

On April 15, 1983, a PLGS plant commissioning test was performed at 100% full power to verify that flow instability can be caused by an initial perturbation. Before the test, the reactor was operated in a steady state at 100% full power and the pressure control setpoint is 9.64MPa(g) which is 9.89MPa(g) in a normal operation state. The outlet header pressure and temperature (such as HD1 and HD3 in Figure 2) were 9.64MPa(g) and 308ºC, respectively. The inlet header pressure and temperature (such as HD2 and HD4 in Figure 2) were around 11MPa and 262ºC, respectively. The flow through each flow pass during the steady state was around 2150kg/s. The parameters for the other loop were same.

When the test began, the two Pressure Control Valves shown in Figure 2 were opened for 10 seconds and steam was discharged from the Pressurizer (shown in Figure 2). With the steam discharge from the pressurizer, the pressure in the outlet headers adjacent to the pressurizer, e.g., HD3 in Figure 2, dropped, which decreased the flow downstream of HD3 and increased the flow upstream of HD3. This perturbation makes pressure in HD1 and HD3 imbalanced which subsequently caused divergent out-of-phase flow oscillations after the Pressure Control Valves were closed (for details see References [5] and [6]). Similar phenomenon was also observed in the other loop. The data demonstrating the oscillation in the flows through instrumented channels and the oscillations in the outlet header pressures were recorded by the plant data logging system and part of the test results are demonstrated graphically in Figure 3, Figure 4, and Figure 5.

The test results in References [5] and [6] show that the amplitude of the flow oscillations is not very large and the propagation and grow of the oscillations may be expected to be sensitive to the simulation time step. To study the influence of the time step on the temporal convergence of the simulation results, different time steps were used to simulate the flow oscillation test. The oscillation amplitude, the increasing rate of the amplitude and the oscillation frequency obtained with different time steps will be compared to the test results or the simulation results in agreement with the test results to demonstrate the impact.

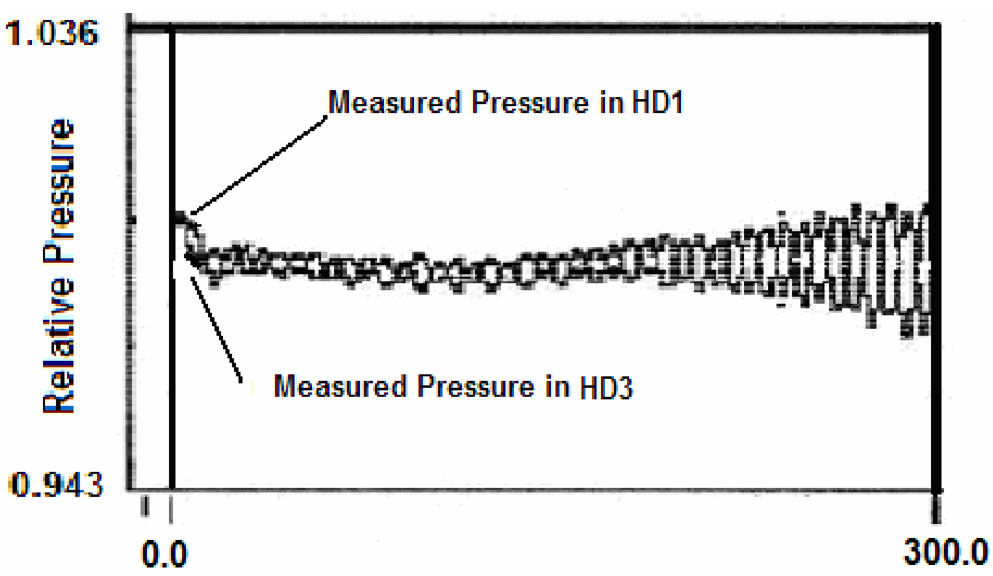

Figure 3: Ratio of Measured Transient Outlet Header Pressure to Steady Outlet Header Pressure (From Reference [6])

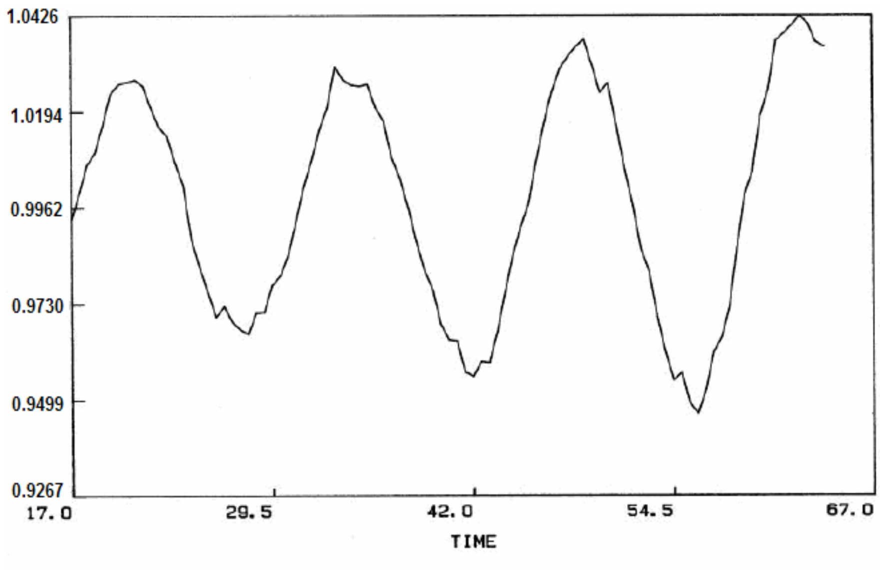

Figure 4: Ratio of Measured Transient Flow to Steady Flow through Channel B9 with 102.5%FP (Reference [6] with Time in Seconds)

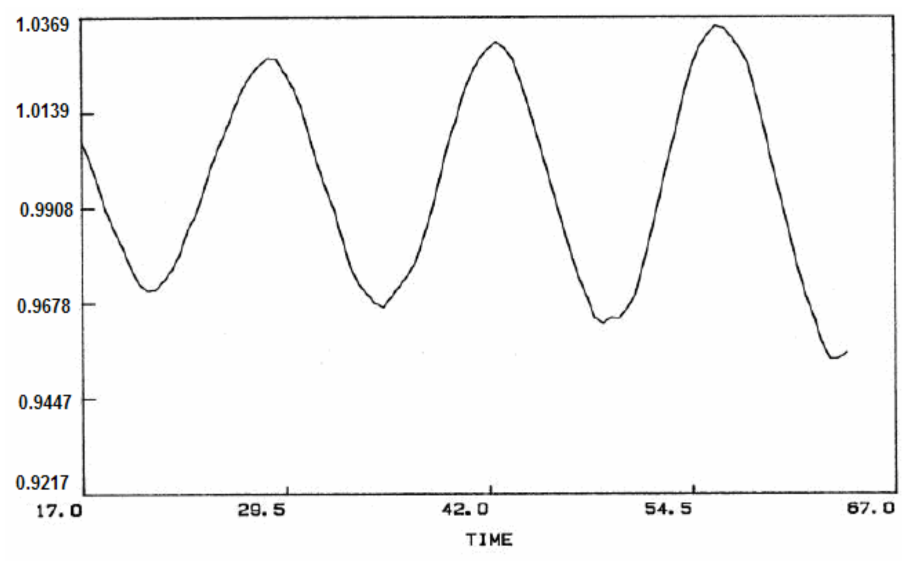

Figure 5: Ratio of Measured Transient Flow to Steady Flow through Channel B14 with 102.5%FP (Reference [6] with Time in Seconds)

The average uncertainty in the calculated inner and outer core powers is about 2.5% (Reference [7]). At same time, with the opening of the pressure control valves, the pressure of the system will decrease and the void fraction in the system will also increase in response. As a result, the reactor power may not remain at the initial value, but a bit higher than the initial value. Since there is no any data of the power increase available, it is assumed that the increase in reactor power can be enveloped by the 2.5%FP uncertainty. Considering these factors, a simulation with 102.5% full power and with the maximum time step of 0.01s is used as the base case. The simulations with 102.5% full power but with larger or smaller time steps are considered sensitivity cases. To study the influence of the reactor power on the flow oscillation frequency, amplitude and the amplitude increasing rate, the test was also simulated with 100% full power with different time steps. The analysis matrix is shown Table 1.

Table 1: Test Simulation Matrix

	Reactor Power	Maximum Time Step
Case 1 (Base Case)	102.5%FP	0.01s
Case 2	100% FP	0.01s
Case 3	100%FP	0.001s
Case 4	102.5%FP	0.1s
Case 5	102.5%FP	0.005s
Case 6	102.5%FP	0.002s
Case 7	102.5%FP	0.001s

The CATHENA integrated model at the start of PLGS plant life (Reference [8]) was used for the simulation together with LEPCON file of MOD-0.0. In the plant model, the models necessary for this work such as primary heat transport system, the pressure control system and the secondary heat transport system including the boiler feedwater system and steam lines are modeled. The primary heat transport system, the boilers and the pressurizer with the pressure control valves are shown in Figure 2 for illustration. In modeling the primary heat transport system, fuel channels with similar channel power and elevation are grouped together for each pass and seven channel groups are obtained for each pass. Since each loop has two passes, there are a total of 28 channel groups in the core. The flows through the channel group containing the instrumented channel is used to compare the flow through this instrumented channel to judge the accuracy of the simulation in flow oscillation amplitude, flow oscillation frequency and the increasing rate in the oscillation amplitude. The heat transfer in channel group and power distribution along each channel group are modeled using the CATHENA GENHTP package.

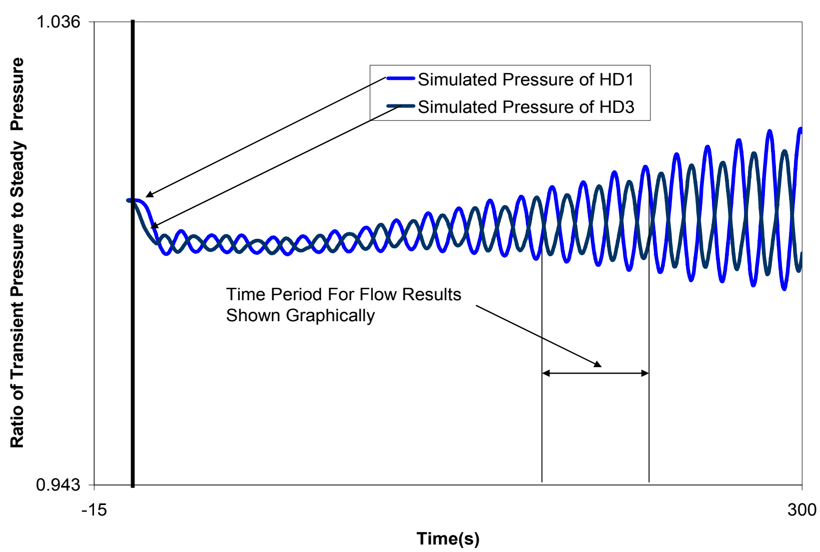

Figure 6: Ratio of Simulated Transient Outlet Header Pressure to Steady Outlet Header Pressure with Reactor Power of 102.5FP and Maximum Time Step of 0.01s

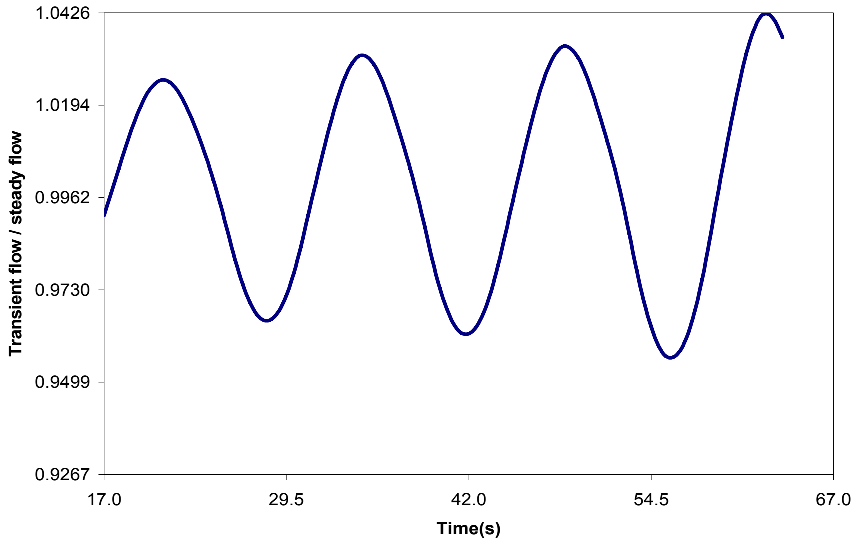

Figure 7: Ratio of Simulated Transient Flow to Steady Flow through Channel B9 with 102.5%FP and Maximum Time Step of 0.01s

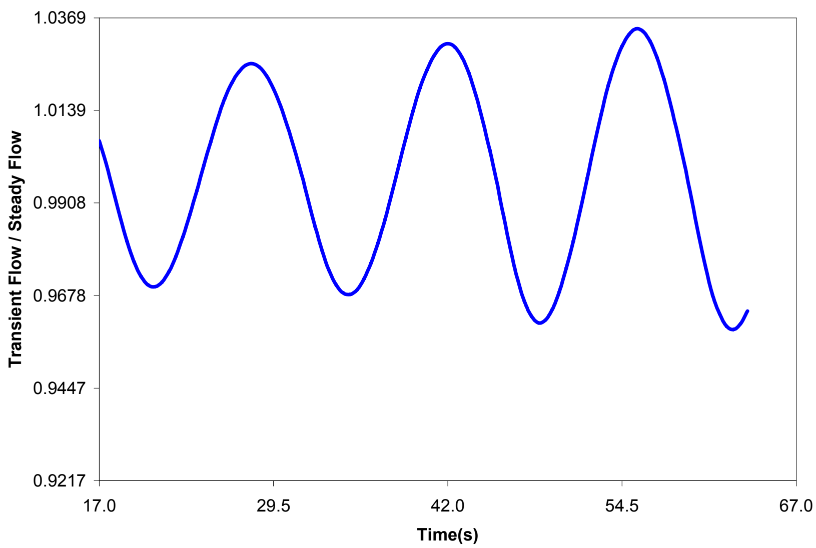

Figure 8: Ratio of simulated transient flow to steady flow through channel B14 with 102.5%FP and maximum time step of 0.01s

4. Results

The simulated responses of the pressures in outlet headers HD1 and HD3 for the base case (Case 1) are shown in Figure 6 and the corresponding measured results are shown in Figure 3 for Loop 1. The measured and simulated flows through Channels B9 and B14 for a specific time period of time are shown in Figure 4, Figure 5, Figure 7 and Figure 8, respectively. A detailed comparison indicates that the simulated oscillation frequency, amplitude, and amplitude increasing rate in both pressure and flow results agree very well with test results. Because of the good agreement between the simulated results for the case and the test results, the base case results will be used to compare with the sensitivity cases to demonstrate the sensitivity of the simulated results to reactor power and the maximum time step.

To demonstrate the influence of the reactor power on the flow oscillation results, the case with 100%FP and maximum time step of 0.01s was simulated (Case 2) and the results were compared with those with 102.5%FP with same maximum time step (Base Case). The flows through Channels B9 and B14 within same time period are shown in Figure 9 and Figure 10.

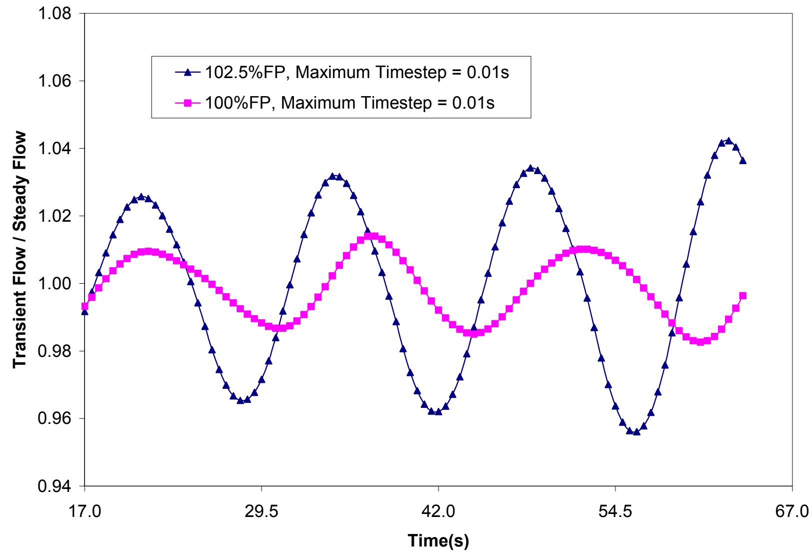

Figure 9: Ratio of Simulated Transient Flow to Steady Flow through Channel B9 with Maximum Time step of 0.01s and Different Power Levels.

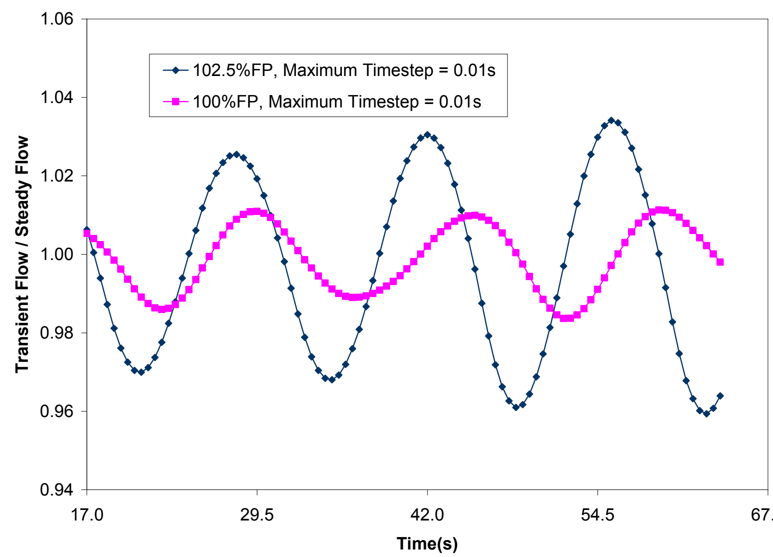

Figure 10: Ratio of Simulated Transient Flow to Steady Flow through Channel B14 with Maximum Time step of 0.01s and Different Power Levels.

Figure 9 and Figure 10 indicate that the flow oscillation amplitude and its increasing rate are very sensitive to the reactor power. These two parameters for 100%FP case are smaller than those for 102.5%FP case, which agrees very well with the test results. However, the oscillation frequencies for the two cases are almost identical which means this parameter is not as sensitive as the other two parameters (amplitude and its increasing rate) to the reactor power.

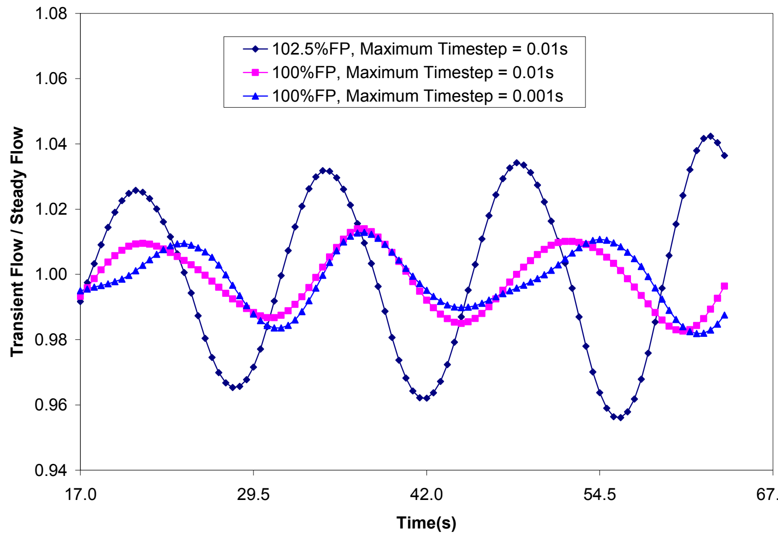

Figure 11: Ratio of Simulated Transient Flow to Steady Flow through Channel B9 with Different Reactor Powers and Different Maximum Time Steps.

The test was also simulated with a smaller time step of 0.001s with 100%FP (Case 3). The simulated flows through Channel B9 with different reactor powers and time steps within the same time period are shown in Figure 11. The results indicate that decreasing the time step to 0.001s does not change the results with 100%FP significantly. This comparison confirms that it is the difference in power, but not in the time step that causes the difference in the results of 100%FP case and 102.5%FP case.

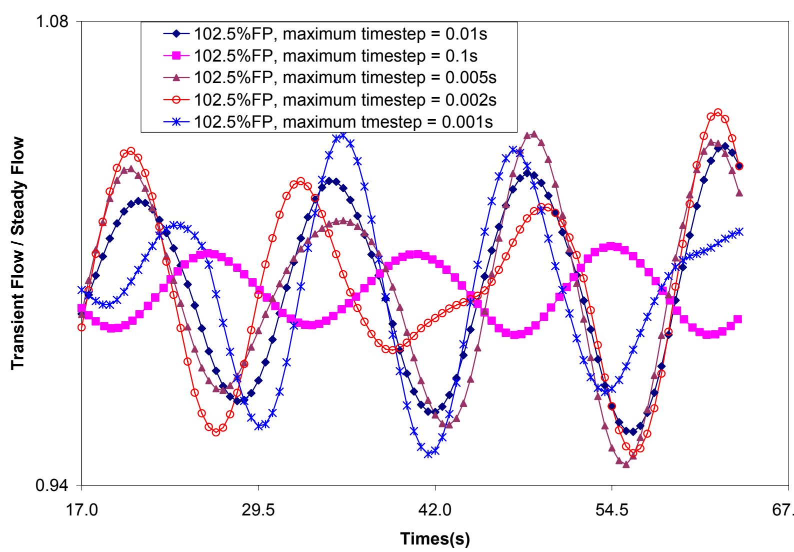

Figure 12: Ratio of Simulated Transient Flow to Steady Flow through Channel B9 with Different Maximum Time Steps at 102.5%FP

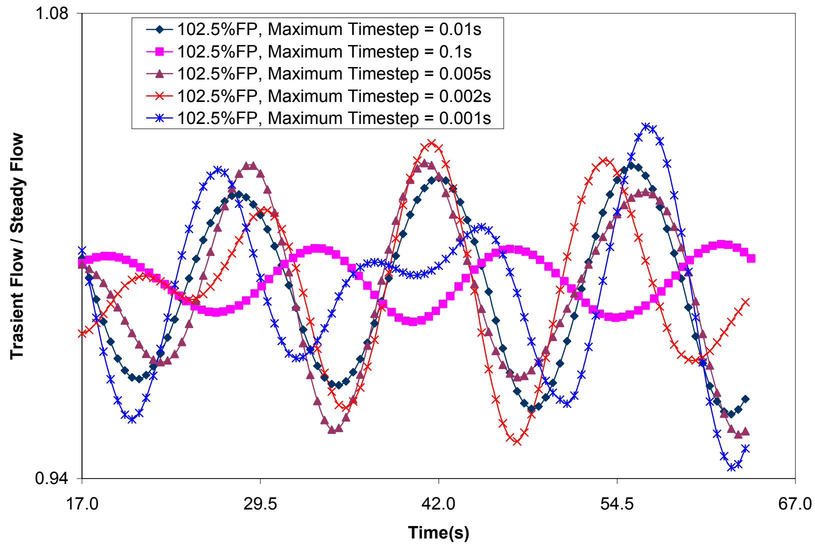

Figure 13: Ratio of Simulated Transient Flow to Steady Flow through Channel B14 with Different Maximum Time Steps at 102.5%FP

To demonstrate the sensitivity of the flow oscillation results to the maximum time step when same reactor power is applied, four more different values of the maximum time step were used to simulate the test with reactor power of 102.5FP (Cases 4 to 7 in Table 1). The flows through Channel B9 and B14 within same time period are shown in Figure 12 and Figure 13 for different values of maximum time steps. The comparison indicates that if the maximum allowed time step is as large as 0.1s, the oscillation amplitude and frequency will be much smaller than that in the case with 0.01s which implies they are much smaller than the test results. However if the maximum applied time step is smaller than 0.01s, the amplitude changes irregularly (instable) deviating from the test results. By spreading the outlet header pressure and Pass 1 flow results to the whole the time span for the cases with 0.1s, 0.01s and 0.001s (Figure 14 and Figure 15), it can be seen that if the time step is as large as 0.1s, the oscillation amplitude does not increase as fast as that with 0.01s time step, though the amplitude is increasing consistently. When the time step is lower than 0.01s, for example, 0.001s, the increasing rate of the oscillation amplitude has a similar tendency as that of 0.01s case in the simulation period, but locally, the amplitude is changing more irregularly. The reason for the irregularly changing amplitude is not clear.

The results in Figure 11 to Figure 15 indicate that if a sufficiently small time step is applied, an accurate tendency of the flow oscillation can be predicted and it is demonstrated that a maximum time step of 0.01s is sufficiently small for temporal convergence when simulating flow oscillations using the CATHENA plant model for Point Lepreau. It is recommended that for simulation of flow oscillations by any perturbations, the influence of the time step, or temporal convergence should be assessed.

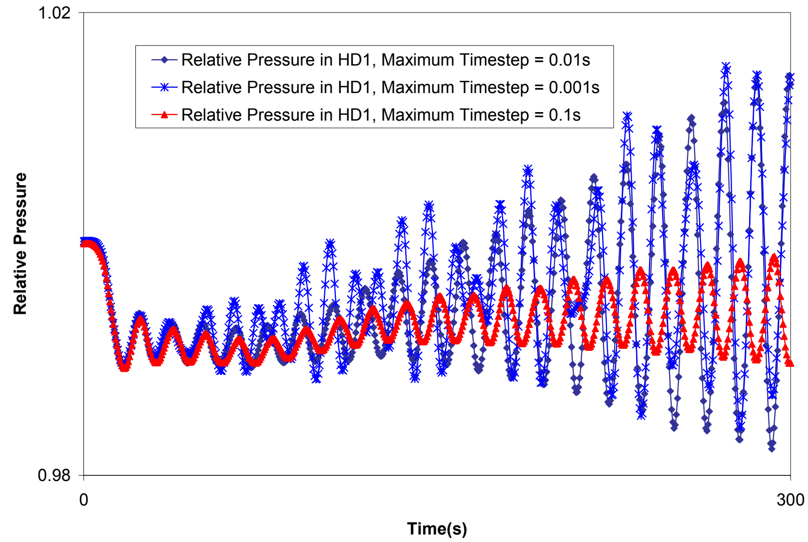

Figure 14: Ratio of Simulated Outlet Header Pressure to Steady Outlet Pressure with Different Maximum Time Steps at 102.5%FP

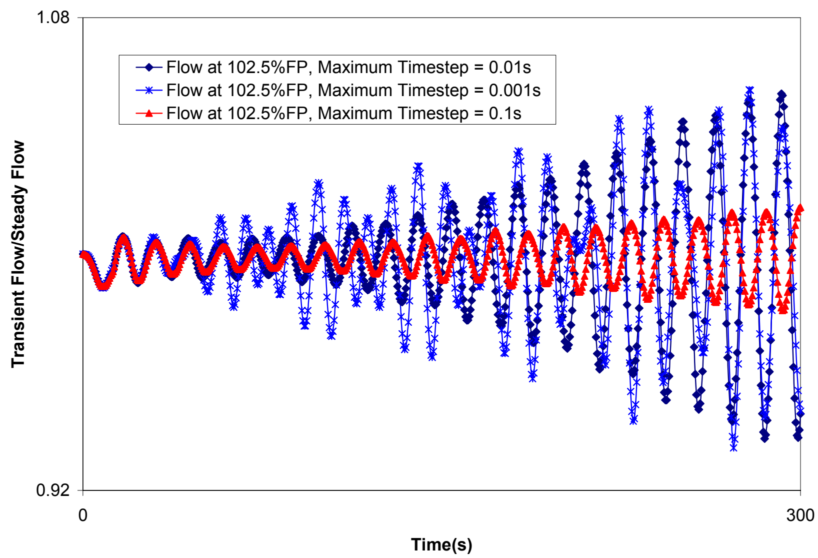

Figure 15: Ratio of Simulated Transient Flow to Steady Flow through Channel B14 with Different Maximum Time Steps at 102.5%FP

5. Conclusion

To confirm the capability of CATHENA to predict flow oscillation accurately, a PLGS commissioning test was simulated using the PLGS plant model with CATHENA code. The sensitivity of the oscillations to the power level and the time step in the code were also verified. The following conclusions can be drawn from this work:

(1) CATHENA code which has been used in numerical simulations in safety analyses is capable of capturing the major characteristics of flow oscillations and this gives the analysts more confidence in simulating oscillation phenomena caused by other initial perturbations

(2) Uncertainty in reactor power has an impact on the oscillation amplitude and its increasing rate.

(3) The time step should be sufficiently small for temporal convergence to obtain accurate simulation results when simulating oscillation phenomena.

(4) The sensitivity analysis indicates that it is always recommended that the influence of time step or temporal convergence should be assessed when simulating this phenomenon.

6. Acknowledgement

The author thanks Dr. Bruce Hanna and Dr. Bruce Hedley for technical review and discussion. The author also thanks Point Lepreau Generating Station to allow me using the plant test data for analysis.

7. References

[1] CANDU 6 Program Team of Reactor Development Business Unit, AECL, "CANDU 6 Technical Summary", http://www.aecl.ca/Reactors/CANDU6.htm, June 2005.

[2] P.J. Inghan, "A.J. Melynk and T.V. Murray, Two-Phase flow Stability Experiments in RD-14M", 12th Annual Conference of the Canadian Nuclear Society, Saskatoon, SK June 9-12, 1991

[3] B.N. Hanna, "CATHENA: A Thermalhydraulic Code for CANDU Analysis", Nuclear Engineering and Design, 180 (1998), 113-131.

[4] R. Girard, "LEPCON Mod-0.0 User's Manual", Point Lepreau Generating Station Information Report IR-03500-51, Vol. 2, Rev. 0, September 1995.

[5] S. Alikhan, W. S. Pilkington and W. J. Garland, "Plant Commissioning Test to Verify Dynamic Stability Phenomenon in CANDU 600 MWe Primary Circuit", Proceedings of the Institution of Nuclear Engineers Conference on Simulation for Nuclear Reactor Technology, Cambridge, England, April 9-11, 1984.

[6] PHT Stability, Book 4A, Point Lepreau Generating Station Vault Record, BSI 33000, February 5, 1989.

[7] C. Newman, "Update of BP/CP Uncertainty Allowances", Point Lepreau Generating Station SAI Report, SAI-03100-1994-06, Rev. 4, May 15, 2003.

[8] R. Prime, "CATHENA Integrated Model", Point Lepreau Generating Station Information Report IR-03551-14, Rev. 5, September, 2001.