CATHENA SIMULATIONS OF STEAM GENERATOR TUBE RUPTURE

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ABSTRACT

The CATHENA thermalhydraulic computer code was used to simulate various scenarios following a CANDU® 9 steam generator tube rupture (SGTR) event. The analysis included cases with class IV power and emergency core cooling system (ECCS) available and other cases with subsequent loss of class IV power (LCIVP) or impairment of ECCS injection. Two main approaches were followed in the analysis of each case. In the first approach, D_2O feed was credited to provide conservative data for input to radionuclide release and dose calculations. Also operator actions are credited. The other approach is designed to give conservative predictions with respect to the acceptance criteria of fuel and fuel channel integrity and to prove that in case of such event, the operator will have enough time to mitigate the consequences. This is done by not crediting makeup for the inventory loss and relying on the automatic operation of safety systems.

The analysis of the cases of the first approach provided the required data for radionuclide release and dose calculations and gave a good insight into the required sequence of operator timely actions to mitigate the consequences of such event. On the other hand, the cases of the second approach confirmed compliance with regulatory requirements for pressure tube and fuel integrity. The runs with ECCS available, showed that ECCS injection is effective in filling and cooling the core and that regulatory requirements for fuel and channel integrity are met. In the event of ECCS impairment, the earliest indication of late fuel heat-up is late enough to provide the operator with an adequate time to act in mitigating the consequences of this event.

INTRODUCTION

In CANDU reactors, steam generator tubes provide a physical boundary between the D_2O coolant of the primary heat transport system (HTS) and the H_2O inventory in the secondary side. Thus a steam generator tube rupture (SGTR) would lead to hydraulic coupling of the primary and the secondary sides, resulting in discharge of D_2O and radioactivity into the shell side of the affected steam generator. Leakage or blowdown from the secondary circuit to the atmosphere would present a direct path for radioactivity to reach the public.

The analysis of SGTR event considers three scenarios, namely, SGTR without additional impairment, SGTR with consequential LCIVP and SGTR with loss of the emergency core cooling (LOECC) system. The main objectives of these analyses are:

i. To produce the transient thermalhydraulic values, including break discharge and enthalpy, needed for radionuclide release and dose calculation. In this analysis, the expected operator actions are credited. However, to maximize the discharge, the D_2O feed is credited.

ii. To confirm compliance with the regulatory requirements for pressure tube and fuel integrity. In this analysis, D_2O feed and operator actions are conservatively not credited.

MODELLING

The CATHENA thermalhydraulic computer code is used in the analysis. CATHENA is a one-dimensional, two-fluid thermalhydraulic computer code developed by AECL primarily for the analysis of postulated loss of coolant accident events in a CANDU reactor (Reference 1). The code has a general network capability and is capable of modelling, in detail, the heat transfer phenomena in a CANDU 9 type fuel channel. The wall heat transfer package provides many heat transfer correlations to be used/selected and includes radial and circumferential conduction, thermal radiation, and the Zr-H_2O reaction heat source. The heat transfer package is general and allows the connection of multiple wall surfaces to a single thermalhydraulic node.

The CANDU 9 primary heat transport system (HTS) is arranged in four core passes, each between a unique inlet header and an outlet header that is shared by two core passes. Each core pass consists of 120 fuel channels. In the circuit model, the 120 channels in each core pass are represented by a single average channel. The CATHENA network model includes the heat transport system, secondary circuit (the steam and feedwater system) and the emergency core cooling system. CATHENA primary and secondary circuit nodalization is shown in Figure 1 while that of the ECC is shown in Figure 2.

For the SGTR analysis, a guillotine break is assumed to occur just above the outlet tube sheet of a U-tube in steam generator 4 (SG4). This location, i.e. at the bottom of the cold leg of a steam generator tube, results in higher break discharge rates. The 4663 steam generator tubes are represented by two parallel lines; one represents the broken tube and the other simulates the remaining 4662 intact tubes. Since the break is a double-ended break, there are two links, each connecting one break end to the secondary side of the steam generator (i.e., PRE4B to PREH4 and BOP4 to PREH4, Figure 3). For break discharge, the CATHENA valve model with the choked flow option is chosen to improve the running efficiency. The schematic diagram of the break discharge model is shown in Figure 3.

ANALYSIS SCOPE

Following a SGTR event, the expected event sequences are: operator actions of tripping the reactor, transferring D_2O from D_2O supply system or valving in light water from the reserve water tank to the D_2O storage tank, performing controlled cooldown of heat transport system and valving in the shutdown cooling system (SDCS). The normal procedure for the controlled cooldown of HTS should be performed by using the condenser steam discharge valves (CSDVs). These actions provide adequate core cooling and minimize doses. However, the main steam safety valves (MSSVs) are conservatively used to cool down the HTS in the present analysis. The present analysis includes two analysis procedures. One is to maximize the dose calculation and the other is to maximize the fuel and pressure tube temperature. Figure 4 shows the various cases covered in the analysis.

In the manual trip cases, the operator is credited to shut down the reactor at 15 minutes after the first unambiguous alarm which identifies a SGTR event. A conservative estimate for the time needed to confirm a SGTR event of 25 minutes is used in the analysis. Accordingly, the operator is credited to shut down the reactor at 40 minutes after the initiation of the break. If continuous makeup is available, the operator's next main action is to isolate the affected steam generator and then initiate manual crash cooling. It is assumed that the operator needs 30 minutes after the confirmation of a SGTR event to close the MSIV and MSSVs for the affected steam generator. Therefore, the MSIV and MSSVs for the affected steam generator are assumed to be closed after 55 minutes from the time the break occurs. If continuous makeup is not available, the operator initiates manual crash cooling 5 minutes after shutting down the reactor while the isolation of the affected steam generator is kept at 30 minutes after the confirmation of a SGTR event. This procedure of early cooling, prior to steam generator isolation, is found to be needed in the case of LCIVP.

The automatic trip cases are designed to confirm that there is enough time for the operator to act and mitigate the consequences of this event and to determine whether the requirements related to the acceptance criteria of the fuel and fuel channel integrity are met. This is done by not crediting D_2O feed makeup for HTS inventory loss and waiting for the automatic operation of safety systems.

RESULTS AND DISCUSSION

A brief summary of the event sequence and results of the various cases are shown in Table 1, while results from selected cases are discussed below.

No Additional Impairment, Continuous D_2O Makeup and Operator Actions

This analysis is considered as the reference case for the set of analyses which produce the thermalhydraulic data required for radionuclide release and dose calculations. The initial break discharge is 8 kg/s from tube sheet side and 2.1 kg/s from the tube side as shown in in Figure 5. Since HTS pressure is kept constant by makeup flow, the break discharge continues to be about 10.1 kg/s from both ends until reactor shutdown. After reactor shutdown at 2400 s, break discharge from the tube sheet side drops to about 7.5 kg/s while that from the tube side increases to about 2.6 kg/s. The decrease in the discharge from the tube sheet side is caused by HTS depressurization while the increase in the discharge from the tube side is caused by the change of fluid from two phase to liquid phase which outweighs the effect of the depressurization. At 3300 s, the affected steam generator is isolated and manual crash cooling is started by opening two MSSVs on each of the three unaffected steam lines. MSSVs opening results in fast depressurization of the secondary side and accordingly higher discharges from both ends of the break. However, only break discharge prior to the isolation of the affected steam generator is needed in calculating the radioactivity release to the atmosphere through the opened MSSVs. The sharp drop in break discharge at 14465 s is due to filling the isolated steam generator and the associated steam line by HTS break discharge.

Figure 6 shows the reactor inlet and outlet header pressures. Prior to reactor shutdown at 2400 s, header pressures remain constant at their steady state values since continuous makeup is assumed available.

The operator can limit the radioactive releases to the atmosphere by closing the opened MSSVs and valving in the shutdown cooling system when reactor header temperatures fall to 177 °C. After the HTS coolant is cooled down to 54 °C, the leak to the secondary side can be stopped by draining the HTS below the steam generator level. Figure 7 shows reactor header temperatures. The coolant temperature in the outlet headers falls below 177 °C at 4100 s. This means that at this time the shutdown cooling system can be valved in for long term cooling.

No Additional Impairment, No D_2O Makeup and No Operator Actions

The initial break discharge is about 8 kg/s from tube sheet side and 2.1 kg/s from the tube side as shown in Figure 8. This discharge rate gradually drops due to the continuous inventory loss until reactor trip. After reactor trip on low pressurizer level at 2864 s, break discharge from the tube sheet side drops to about 6.2 kg/s while that from the tube side increases to about 2.2 kg/s. These changes in discharge rates are due to change in pressure and flow regime. Break discharges continue to drop due to the continuous depressurization of HTS. At 5290 s, break discharges drop sharply due to the depletion of the pressurizer. At 5642 s, the ECC signal, of low outlet header pressure, comes in. This is followed by the conditioning signal of sustained HTS low pressure at 6242 s. The conditioning LOCA signal results in initiating crash cooling and results in ECCS injection at 6253 s. Shortly after the start of ECCS injection, break discharges increase sharply due to the HTS refill. Then break flows resume a gradual decline due to the continued depressurization of the HTS. The last sharp increase in break flows, at 9605 s, is caused by the increase in the pressure of the primary side of the steam generators due to automatic HTS pump trip when a pressure of 2.1 MPa(a) in either reactor outlet header is sustained for 10 minutes.

Figure 9 shows the reactor inlet and outlet header pressures. After the initiation of the event, the pressure drops gradually due to the HTS depressurization. Reactor trip at 2864 s on low pressurizer level, results in a sharp drop in pressure. The sharp drop in header pressures at 5290 s is due to pressurizer depletion which lowers the HTS pressure to the saturation pressure corresponding to the coolant temperature. After the initiation of crash cooling and ECCS injection, the depressurization rate becomes faster. The main pumps trip, at 9605 s, results in more uniform system pressure.

ECC flow to the headers is shown in Figure 10. All six rupture disks are ruptured almost simultaneously at around 6253 s. The injection is mainly to the outlet headers since they are at lower pressure than the inlet headers. Prior to main pumps trip, flow is recirculated from inlet headers to the outlet headers.

Figures 11 shows the flow rate of pass 4. Reactor trip and initiation of ECC injection results in sharp increase in core flow rate. After the main pumps trip, at 9605 s, flow through the core drops sharply since forced circulation is lost and flow is governed by natural circulation.

Figure 12 shows fuel sheath temperatures of a high power single channel in pass 4. Prior to reactor trip, the fuel sheath temperatures remain close to the initial steady state value. After reactor trip and prior to ECCS injection, sheath temperatures remain around 260 °C and drop to lower than 200 °C after ECCS injection. Accordingly, fuel integrity is ensured.

With Loss of Class IV Power, Continuous D_2O Makeup and Operator Actions

Up to the time at which the LCIVP occurs, the results are the same as in the case with class IV power. About 15 s after reactor trip, class IV power is lost and thus the HTS pumps are tripped. The initial break discharge is 8 kg/s from tube sheet side and 2.1 kg/s from the tube side. Figure 13 shows that shortly after reactor shutdown at 2400 s, break discharge form the tube sheet side regains its initial value of about 8 kg/s while that from the tube side increases to about 2.6 kg/s. After 3300 s, the pressure in the isolated steam generator (SG4) increases temporarily and accordingly break discharge decreases. However, when HTS coolant temperature drops to lower than SG4 saturation temperature, SG4 pressure, shown in Figure 14, starts to decrease due to steam condensation.

Figure 15 shows reactor header temperatures. As can be seen from this figure, the three unaffected steam generators are effective in cooling the HTS in this mode of single phase thermosphoning. The coolant temperature of inlet header 4 (RIH4) and outlet header 1 (ROH1) are close to each other due to the isolation of SG4. At 6545 s, the coolant temperature in both outlet headers falls to lower than 177 °C. This means that the shutdown cooling system can be valved in at this time for long term cooling.

With Loss of ECC, No D_2O Makeup and No Operator Actions

The initial break discharge is about 8 kg/s from tube sheet side and 2.1 kg/s from the tube side as shown in Figure 16. This discharge rate gradually drops due to the continuous inventory loss until reactor trip. After reactor trip on low pressurizer level at 2864 s, break discharge from the tube sheet side drops to about 6.2 kg/s while that from the tube side increases to about 2.2 kg/s. These changes in discharge rates are due to change in pressure and flow regime. At 5290 s, break discharges drop sharply due to the depletion of the pressurizer. The ECC signal, of low outlet header pressure, comes in at 5642 followed by the conditioning signal of sustained HTS low pressure at 6242 s. The conditioning LOCA signal results in initiating crash cooling. Since ECCS injection is impaired, the HTS depressurizes quickly as shown in Figure 17. After the main pumps trip at 7179 s, the HTS pressure becomes more uniform.

Figures 18 and 19 show channel flow and fuel sheath temperatures of a high power single channel in pass 4. The channel flow increases on reactor trip as long as the pressurizer kept the HTS refilled. After pressurizer depletion, HTS void starts to increase and accordingly channel flow decreases. The first indication of fuel heatup occurs later than 8500 s. This ensures that the operator has enough time to act to mitigate the consequences of this event such us providing makeup for the inventory loss and valving in the shutdown cooling system.

CONCLUSION

Two main approaches for a single steam generator tube rupture event are followed. In the first approach, D_2O feed is credited to provide conservative data for input to separate radionuclide release and dose calculations. Also operator actions are credited. The other approach is designed to give conservative predictions with respect to the fuel and fuel channel integrity and to prove that in case of such event, the operator will have enough time to mitigate the consequences. This is done by not crediting makeup for the inventory loss and relying on the automatic operation of safety systems.

The main operator actions for a SGTR event include shutting down the reactor, providing continuous makeup, isolating the affected steam generator, conducting controlled cooldown of HTS and ultimately valving in the SDCS. The normal procedure for the controlled cooldown of HTS should be performed by using CSDVs. These actions provide adequate core cooling and minimize doses. However, the MSSVs are conservatively used in cooling the HTS in the present analysis.

The analysis provided the thermalhydraulic data required for radionuclide and dose calculations and identified the time at which the operator can valve in the SDCS. The SDCS can be valved in to cool down the HTS from the zero power hot temperature of 260 °C under abnormal conditions. However, the normal operation of SDCS to cool down the HTS from 177 °C is assumed in the present analysis.

If continuous makeup is not available, manual crash cooling should be initiated prior to isolating the affected steam generator. This is found to be a necessary procedure in the case with subsequent LCIVP. A sensitivity test showed that the combined effect of isolating the affected steam generator and losing forced circulation resulted in heating up the pass downstream of the isolated steam generator when continuous makeup is not available.

Several cases were conducted to confirm compliance with regulatory requirements for pressure tube and fuel integrity. In these runs, coolant makeup was conservatively not credited. The manual reactor trip was ignored and the next trip signal, low pressurizer level, was credited to trip the reactor. Single channel runs for these cases were conducted. The runs with ECCS available, showed that ECCS injection is effective in filling and cooling the core and that regulatory requirements for fuel and channel integrity are met. In the event of ECCS impairment, the earliest indication of late fuel heat-up is later than 8500 s from the time the break occurs. Thus, the operator has enough time to act in mitigating the consequences of this event.

REFERENCES

1– Richards, D.J., Hanna,, B.N., Hobson, N. and Ardron K.H., "ATHENA Two–Fluid Code for CANDU LOCA Analysis", Presented at the Third International Topical Meeting on Reactor Thermalhydraulics, Newport RI, USA ,1985, Oct. 15–18, pp. 7.E.1 – 7.E.14. (ATHENA was subsequently renamed CATHENA).

Table 1
Summary of Main Events and Results for the Analyzed Cases

Case	Reactor Trip Time (s)	LCIVP Time (s)	MSSVs Opening Time (s)	SG4 MSSVs & MSIV Closing Time (s)	ECC Injection Time (s)	SDCS Time (s) (reactor header temp. reach 177 °C)	Late Fuel Heat–up
No Additional Impairment							
Manual Trip Continuous D_2O Feed	2400	N/A	3300 (manual)	3300	not reached	4100	no
Manual Trip Limited D_2O Feed	2400	N/A	2700 (manual)	3300	4287	3507	no
Automatic Trip	2864 (low pressurizer level)	N/A	6242 (automatic)	no isolation of SG4 and automatic opening of MSSVs	6253	N/A	no
LCIVP							
Manual Trip Continuous D_2O	2400	2415	3300 (manual)	3300	not reached	6545	no
Manual Trip Limited D_2O Feed	2400	2415	2700 (manual)	3300	4440	4500	no
Automatic Trip	2864 (low pressurizer level)	2878	6834 (automatic)	no isolation of SG4 and automatic opening of MSSVs	6917	N/A	no
LOECC							
Manual Trip Limited D_2O Feed	2400	N/A	2700 (manual)	3300	N/A (ECCS impaired)	3507	no
Automatic Trip	2864 (low pressurizer level)	N/A	6242 (automatic)	no isolation of SG4 and automatic opening of MSSVs	N/A (ECCS impaired)	N/A	yes after 8500 s

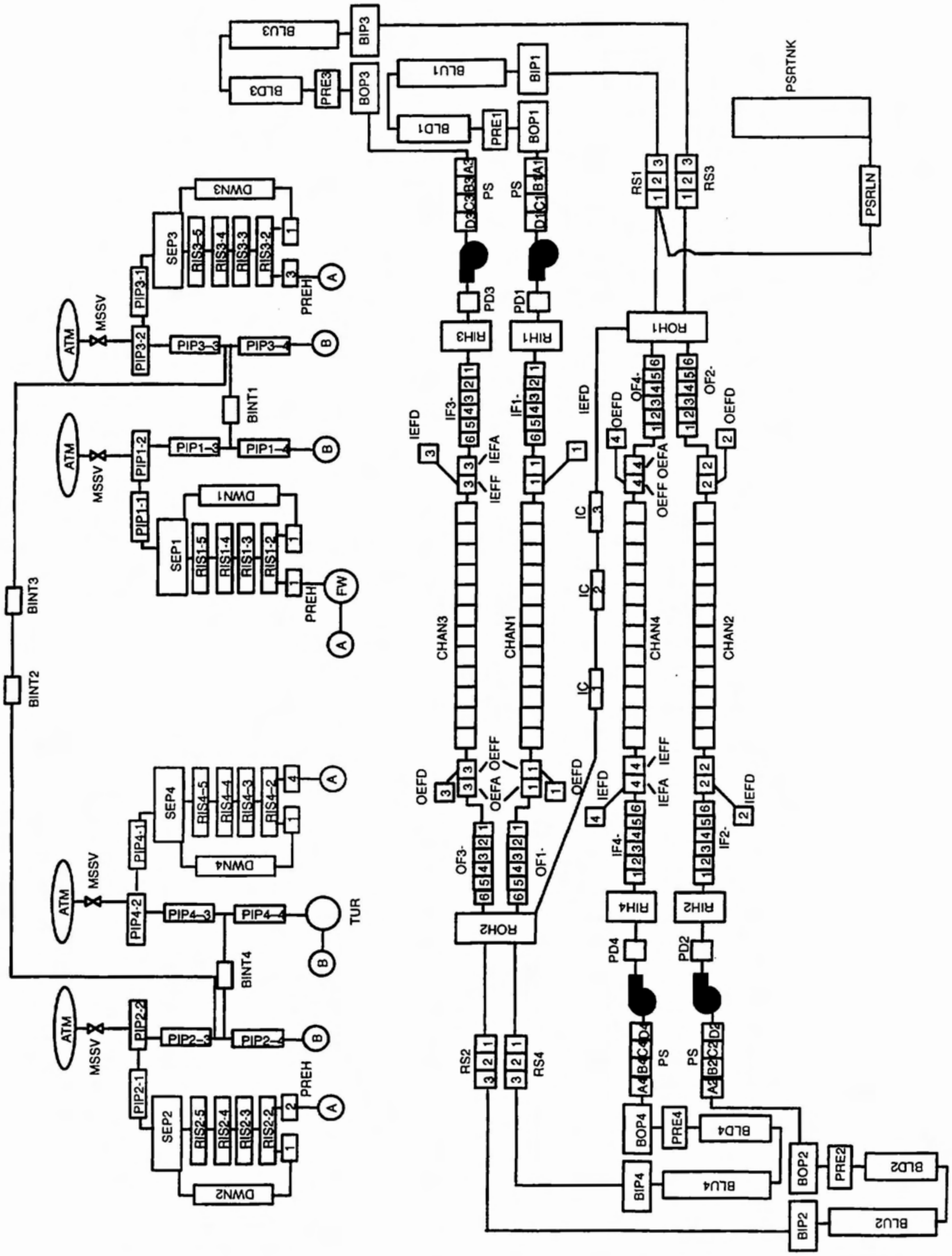


Figure 1 CANDU 9 480/NU CATHENA Nodalization

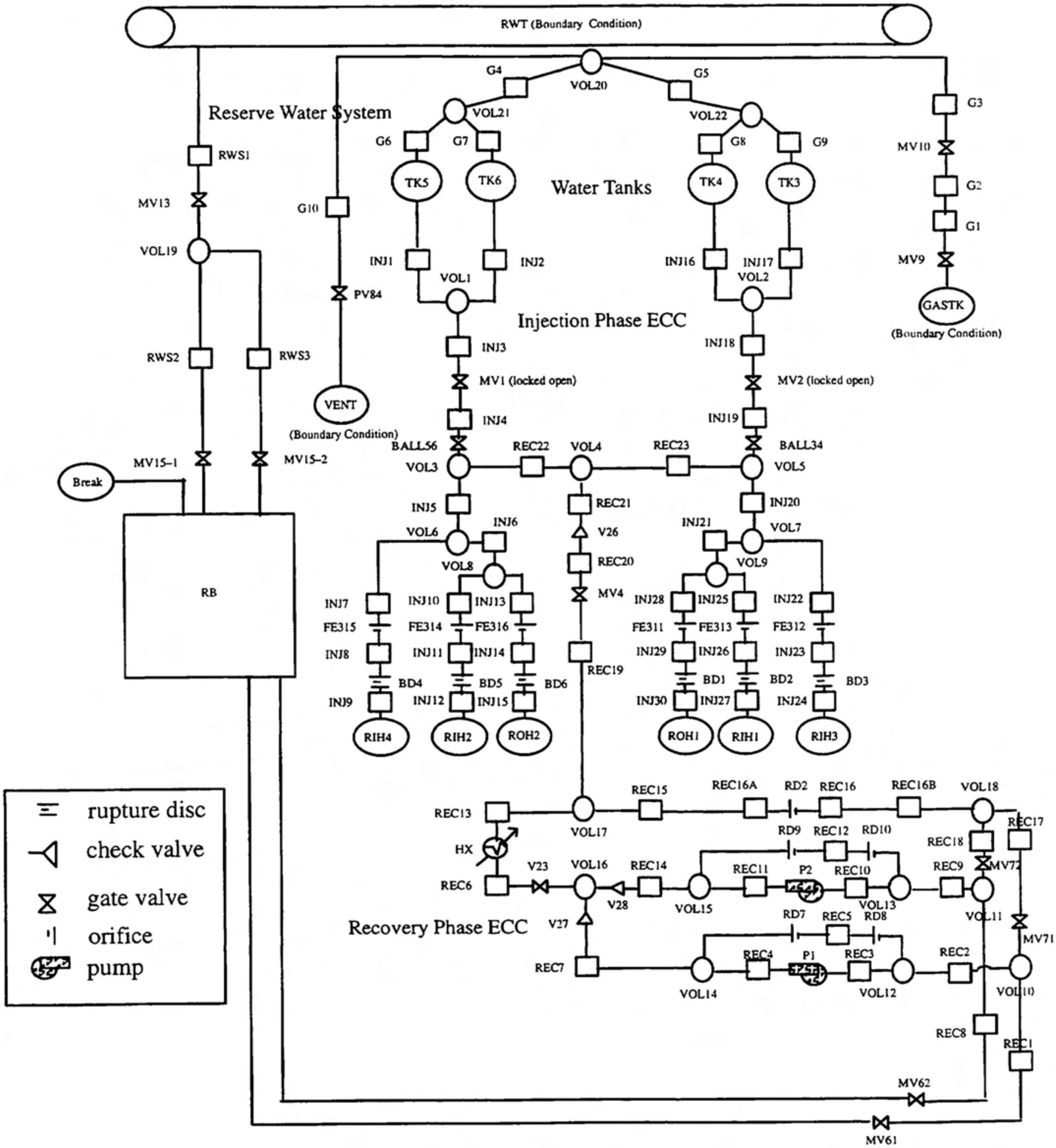


Figure 2 CATHENA Idealization of ECCS

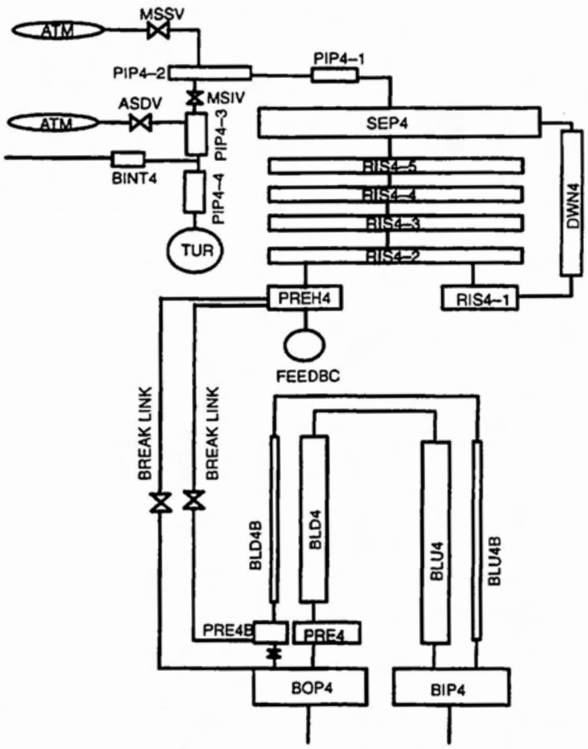


Figure 3 Schematic Diagram of the Break Discharge Model

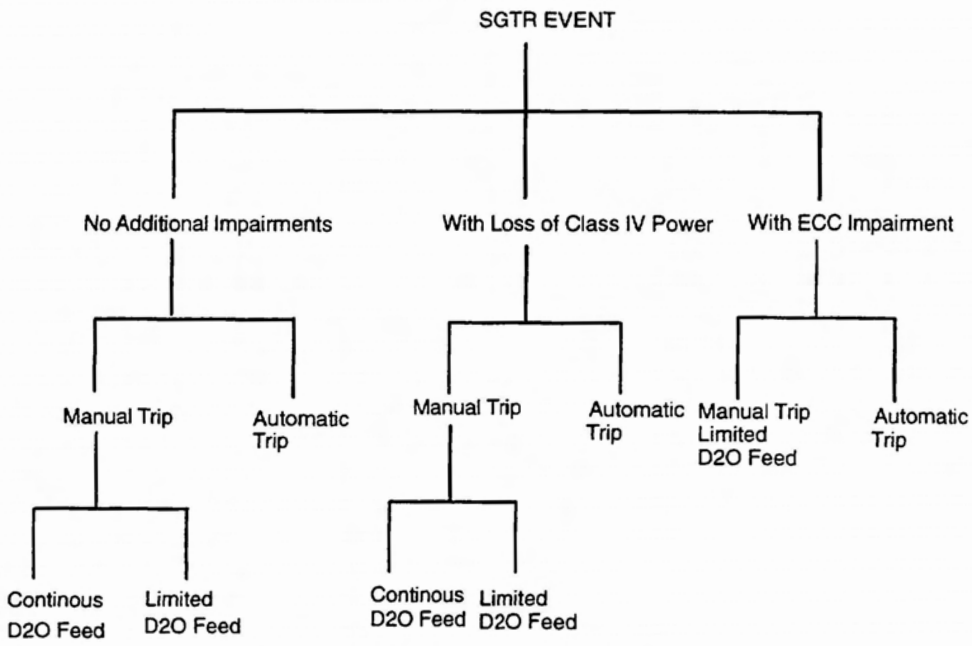


Figure 4 Analysis Cases

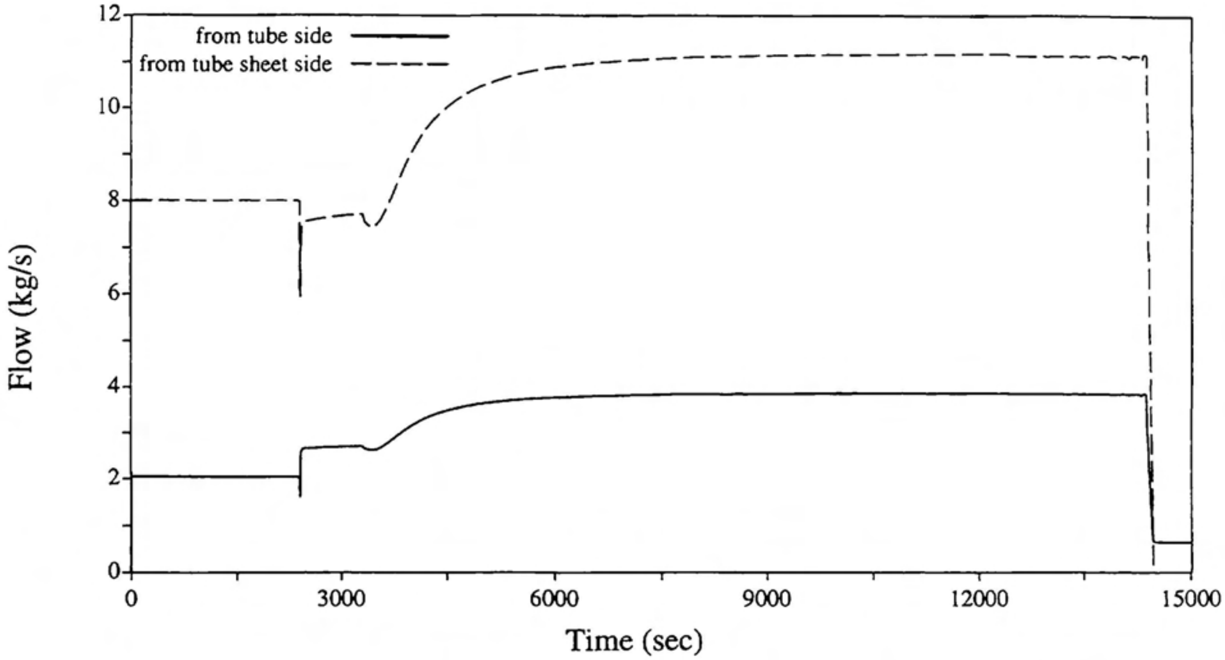


Figure 5 Break Discharge Rates For SGTR With Continous D_2O Feed and No Additional Impairments

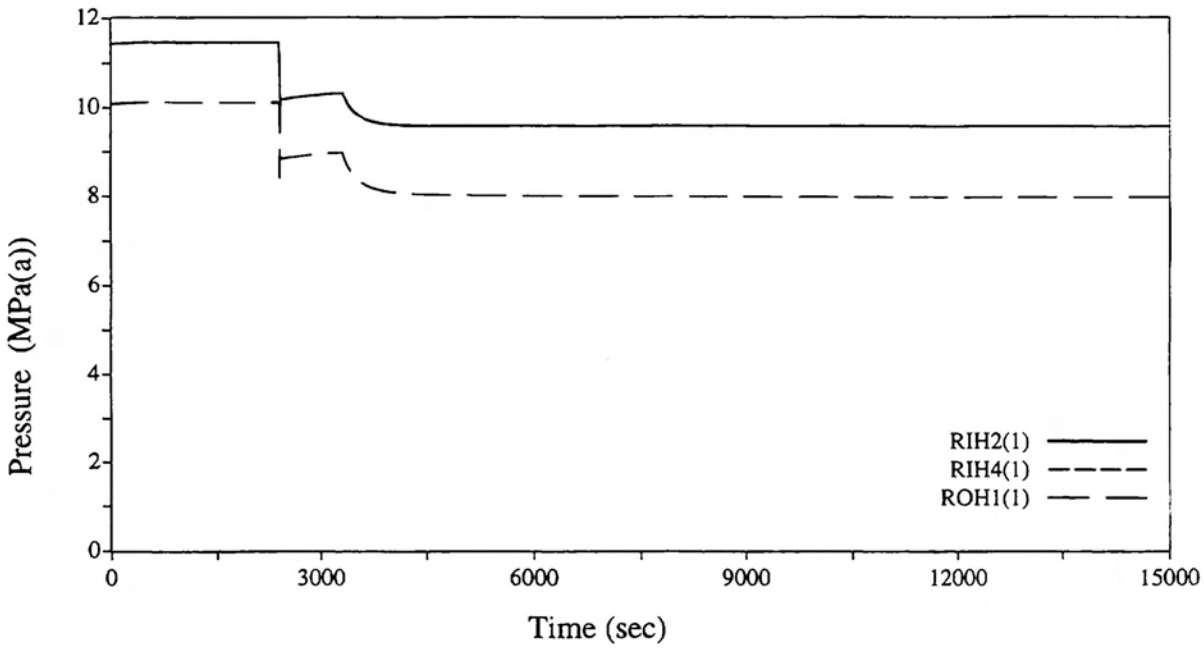
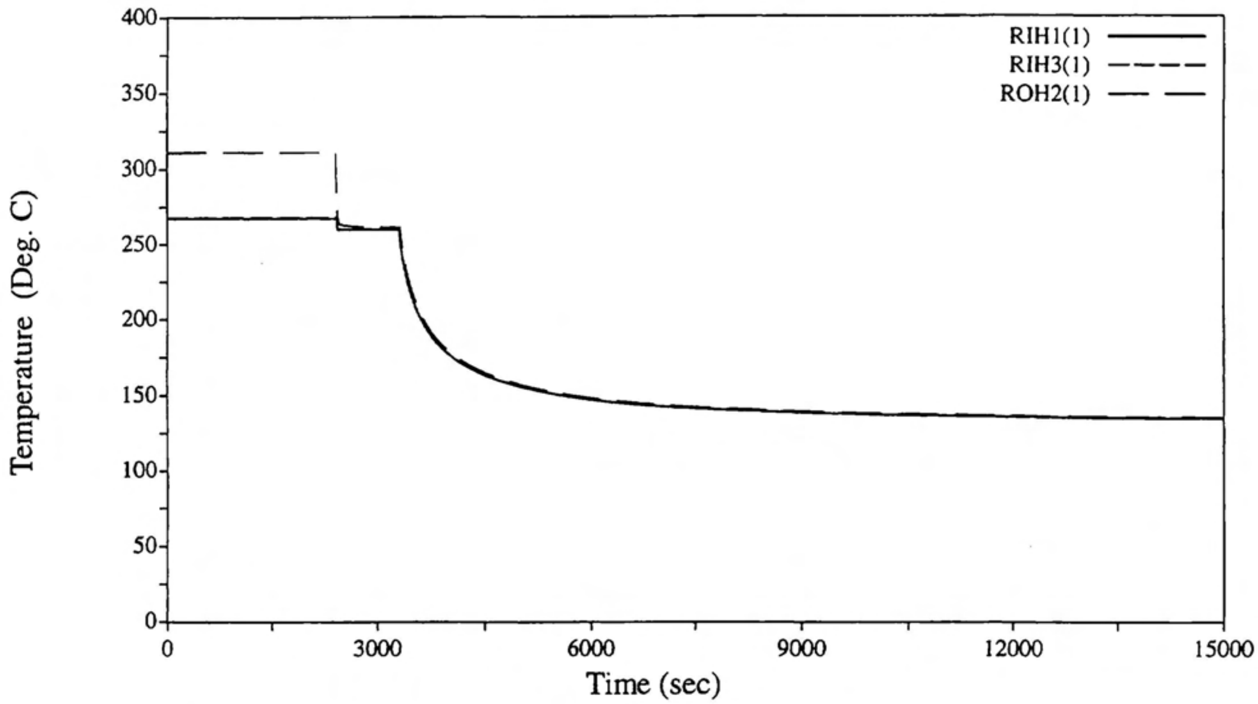


Figure 6 Reactor Header Pressures For SGTR With Continous D_2O Feed and No Additional Impairments



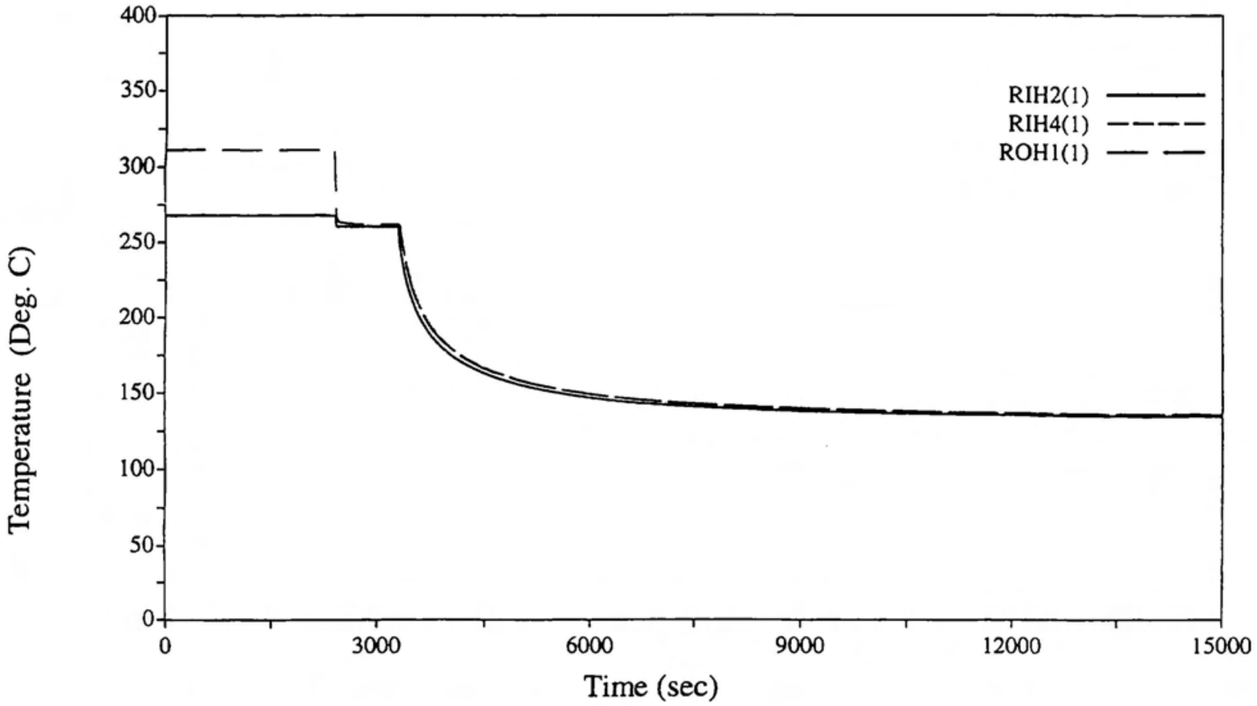


Figure 7 Reactor Header Coolant Temperature With Continuous D_2O Feed and No Additional Impairments

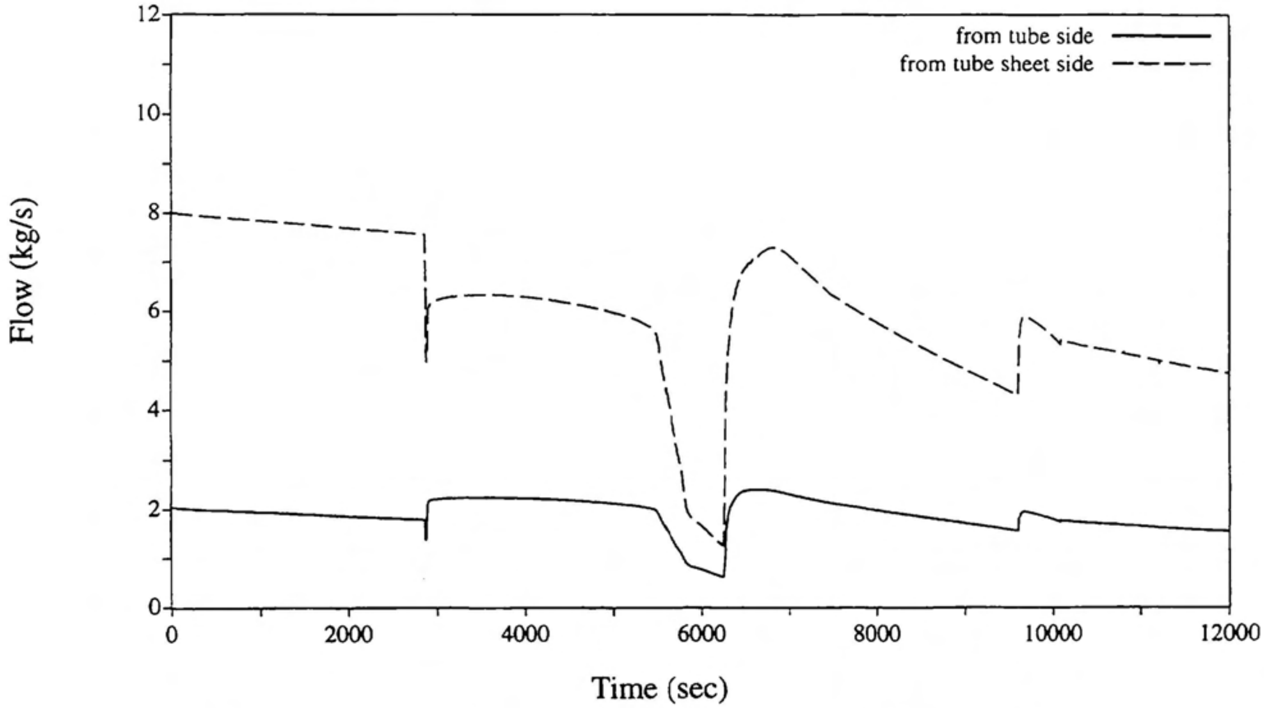


Figure 8 Break Discharge Rates For SGTR With Automatic Reactor Trip and No Additional Impairments

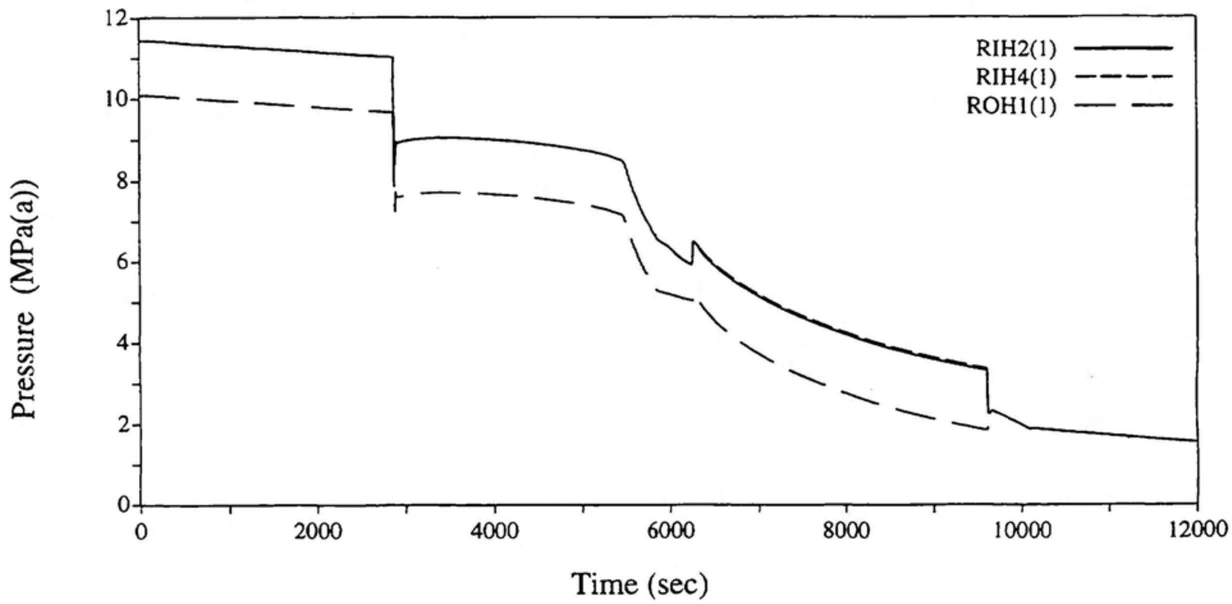
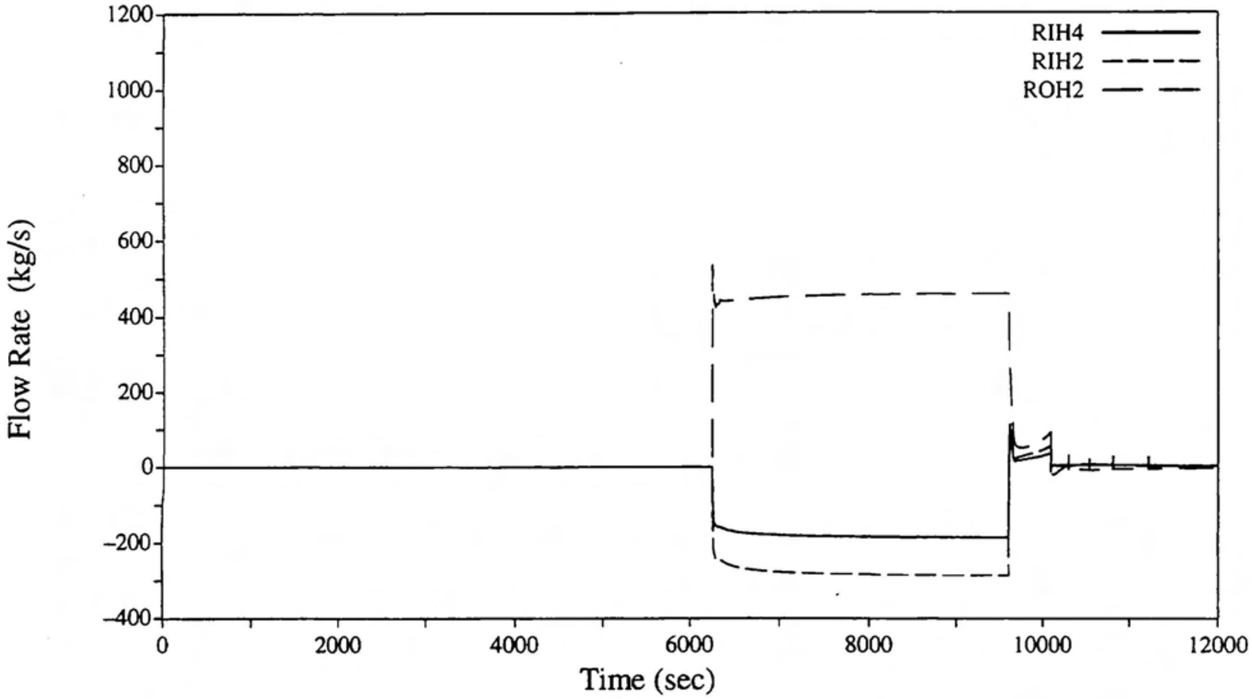


Figure 9 Reactor Header Pressures For SGTR With Automatic Reactor Trip and No Additional Impairments



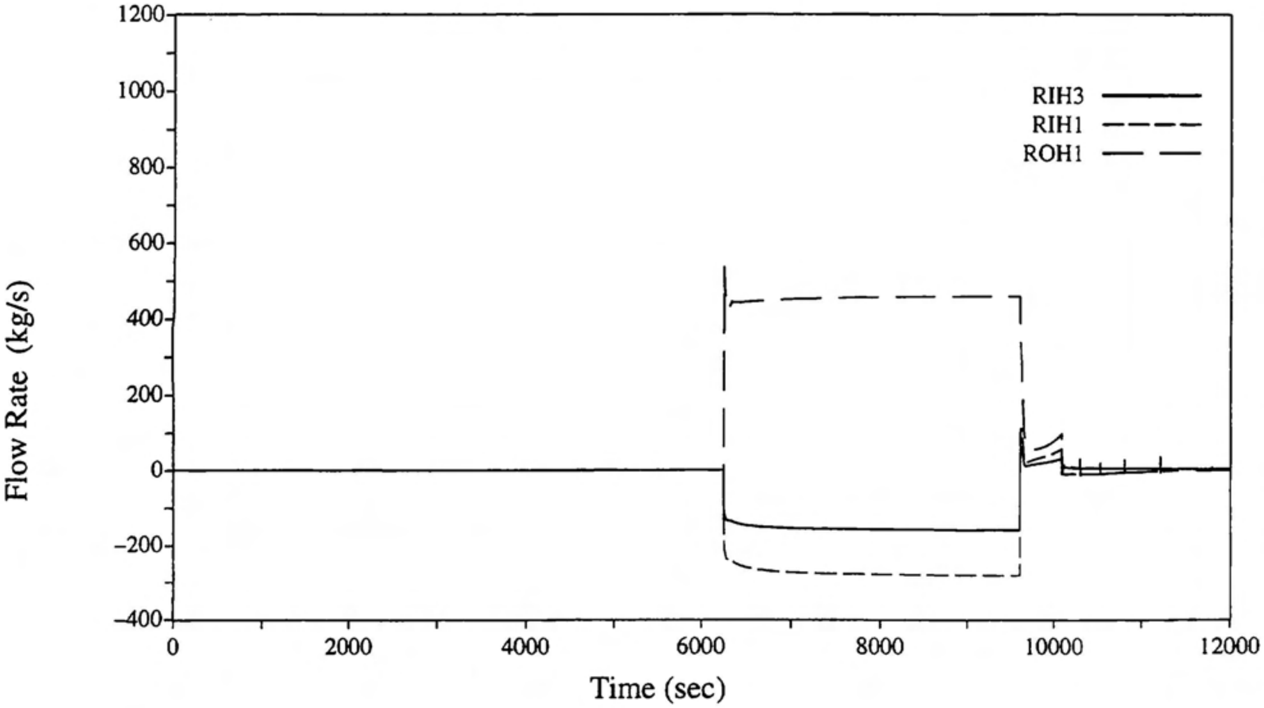


Figure 10 ECCS Injection to the Headers For SGTR With Automatic Reactor Trip and No Additional Impairments

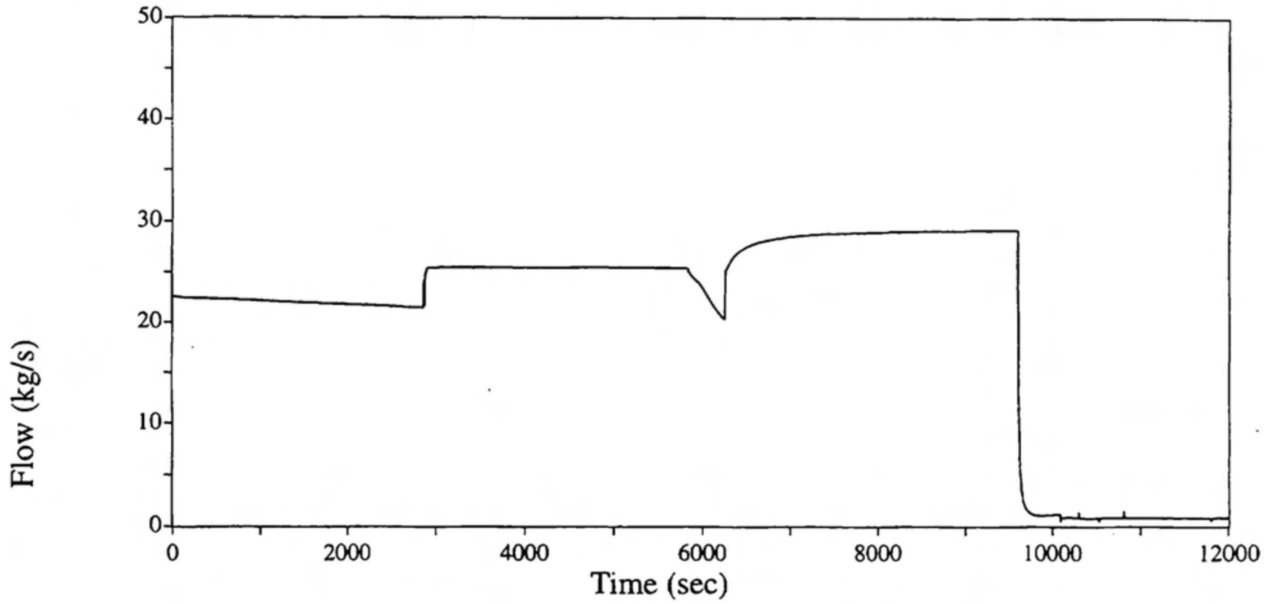


Figure 11 Pass Coolant Flows For SGTR With Automatic Reactor Trip and No Additional Impairment

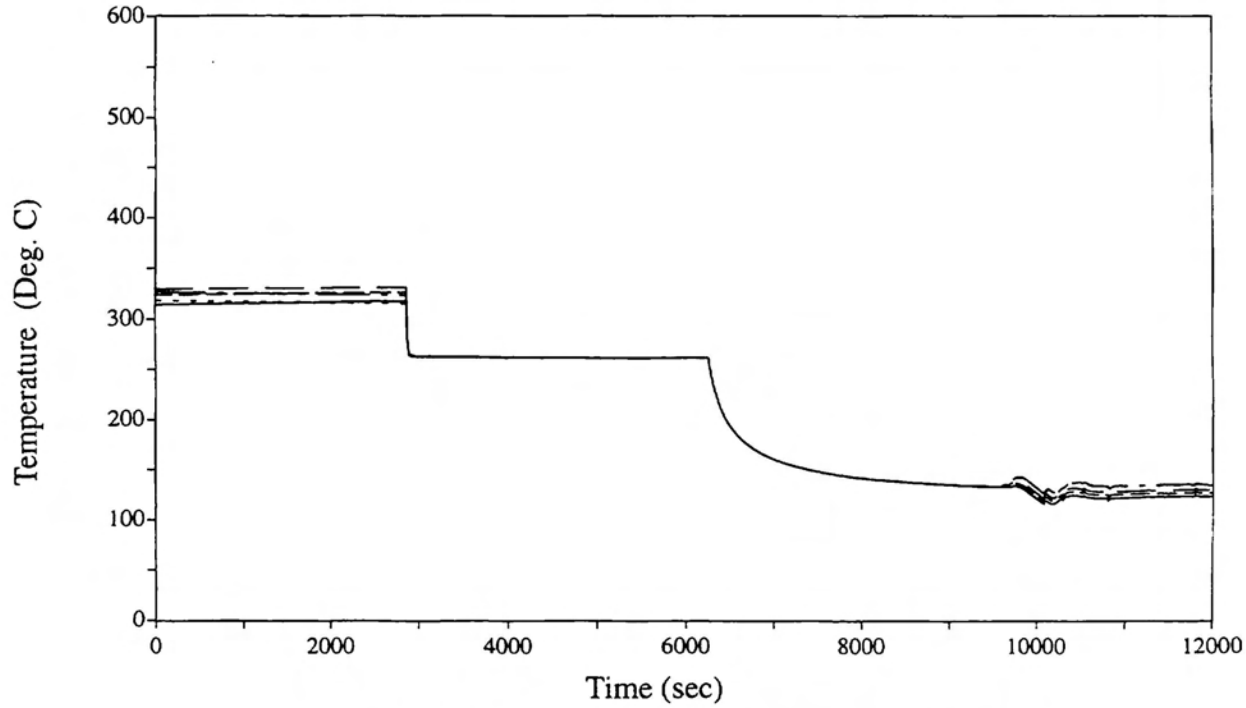


Figure 12 Fuel Sheath Temperatures – The Single Channel Model – Pass 4 Header Conditions – SGTR With Automatic Reactor Trip and No Additional Impairments

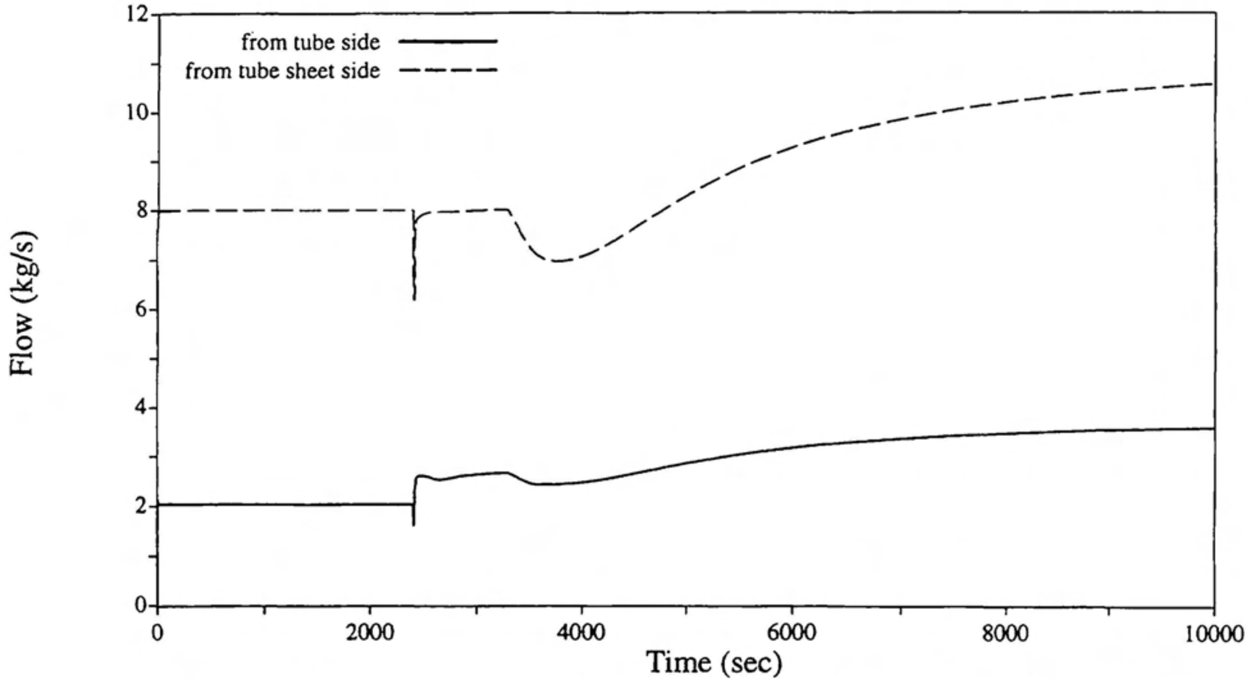


Figure 13 Break Discharge For a SGTR With LCIVP And Continuous D_2O Feed

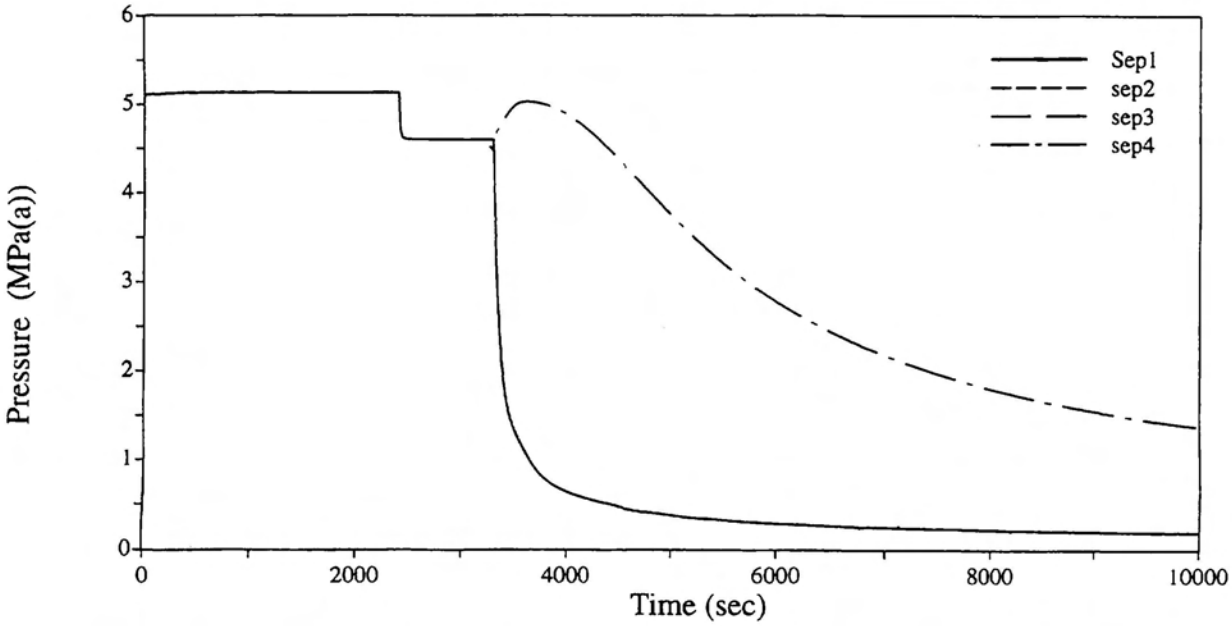


Figure 14 Pressure Transients Of Boiler Separators For a SGTR With LCIVP And Continuous D_2O Feed

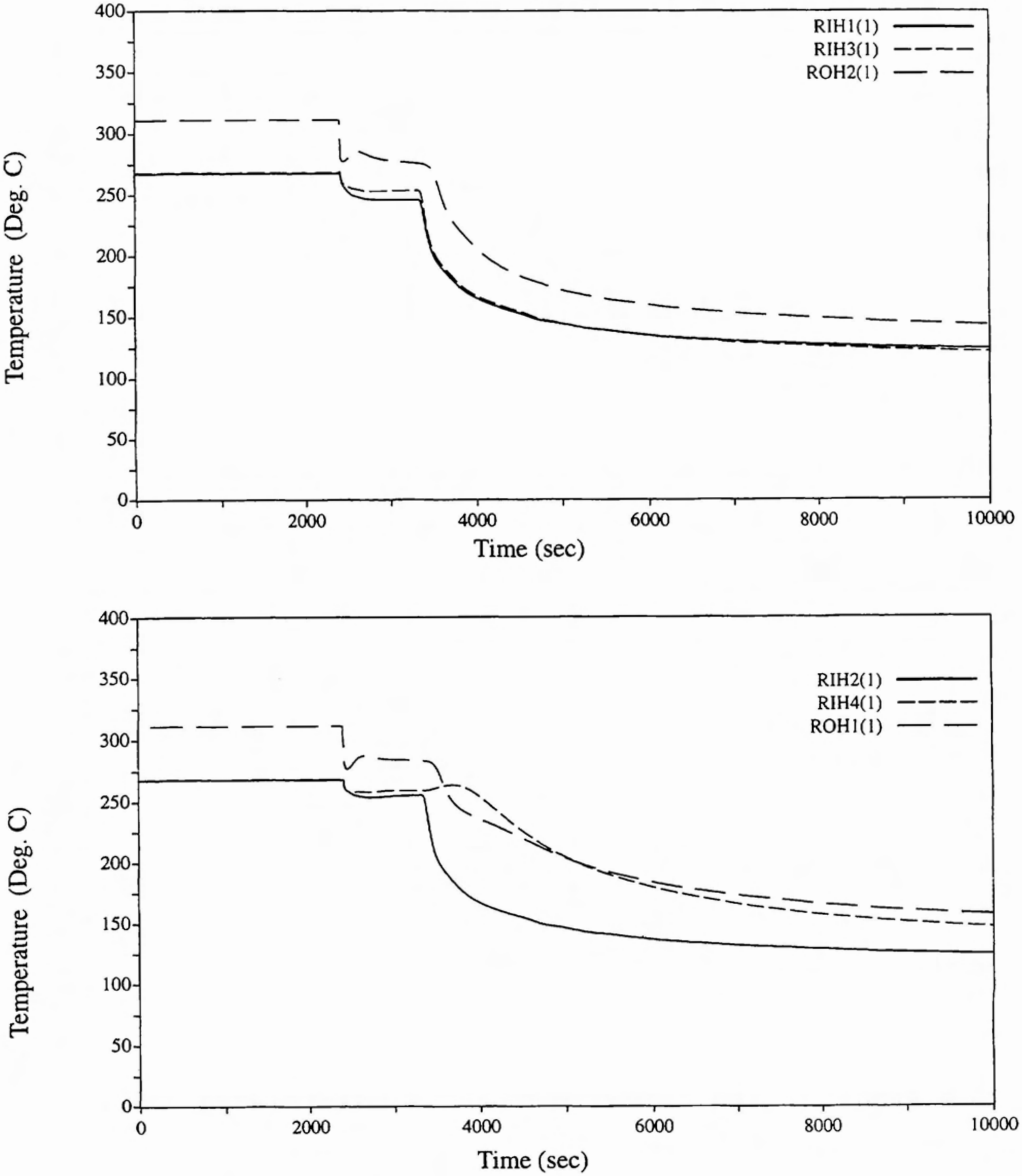


Figure 15 Reactor Header Coolant Temperature For a SGTR With LCIVP And Continuous D_2O Feed

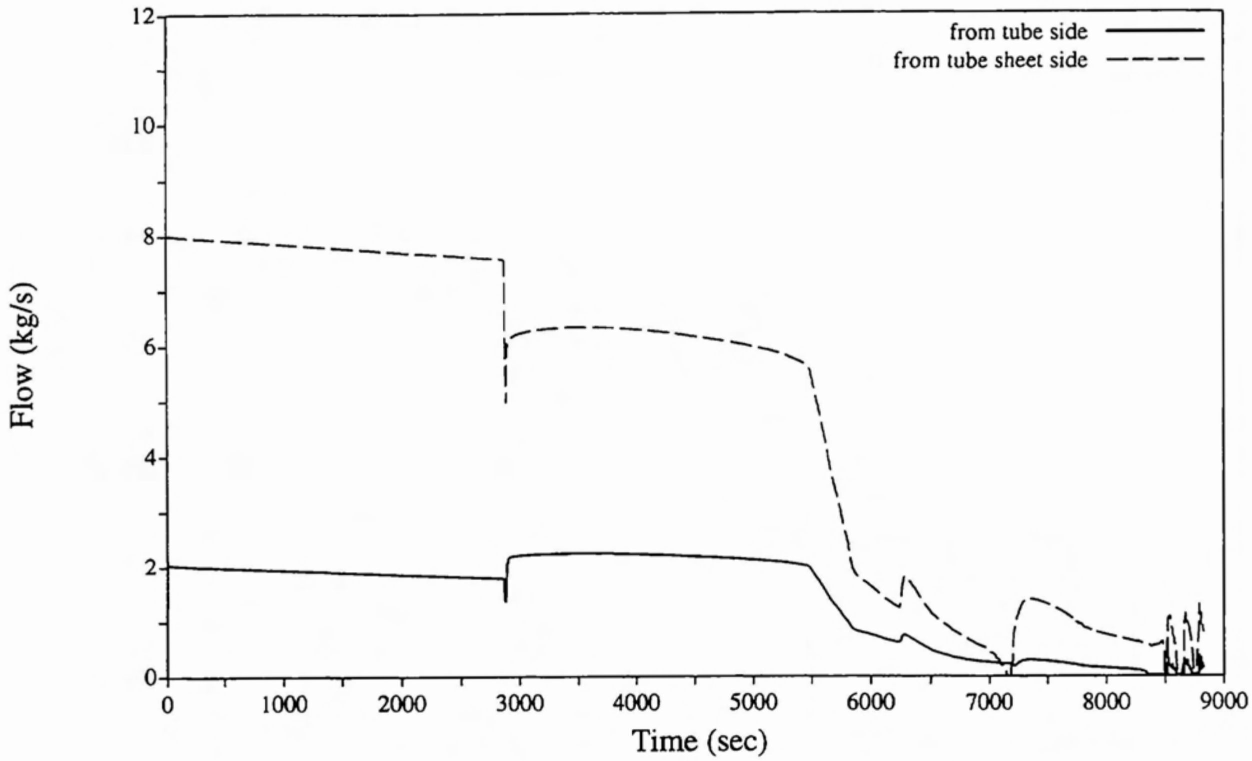


Figure 16 Break Discharge For SGTR With LOECC and No Operator Intervention

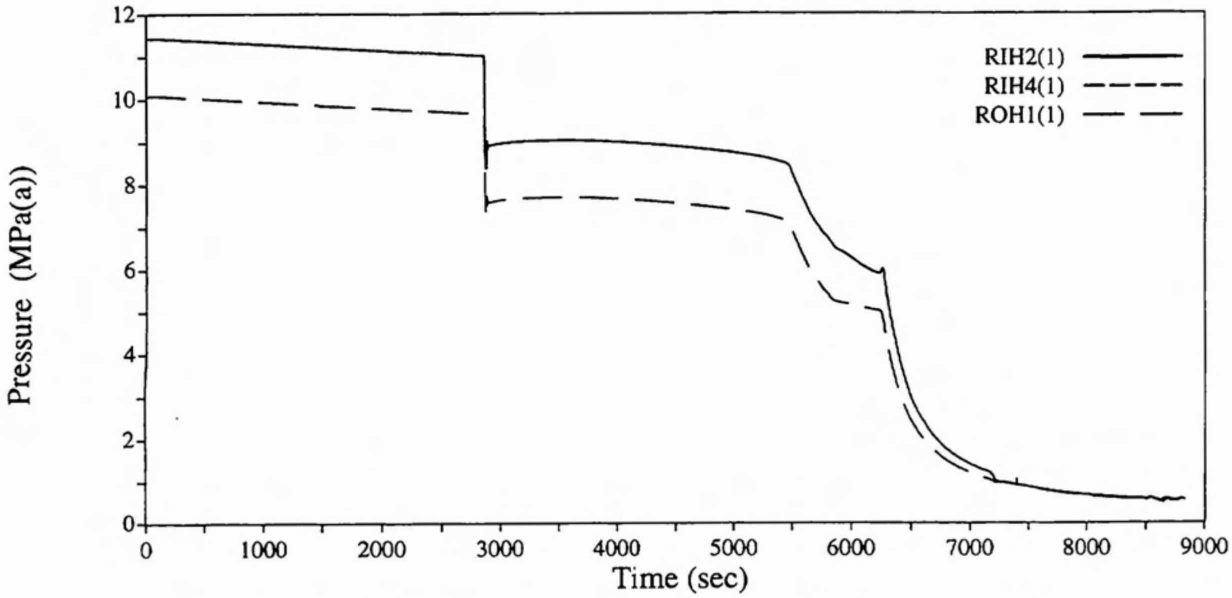


Figure 17 Reactor Header Pressures For SGTR With LOECC and No Operator Intervention

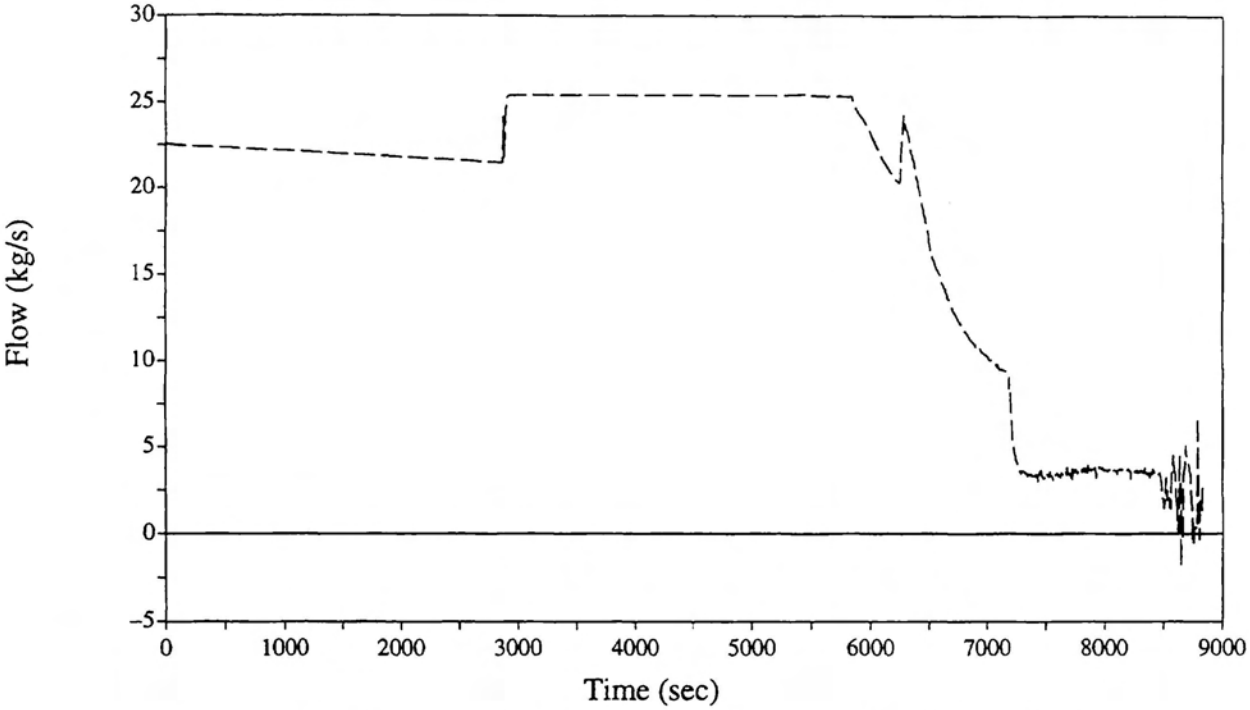


Figure 18 Channel Flow of the Single Channel Model – Pass 4 Header Conditions For SGTR With LOECC and No Operator Intervention

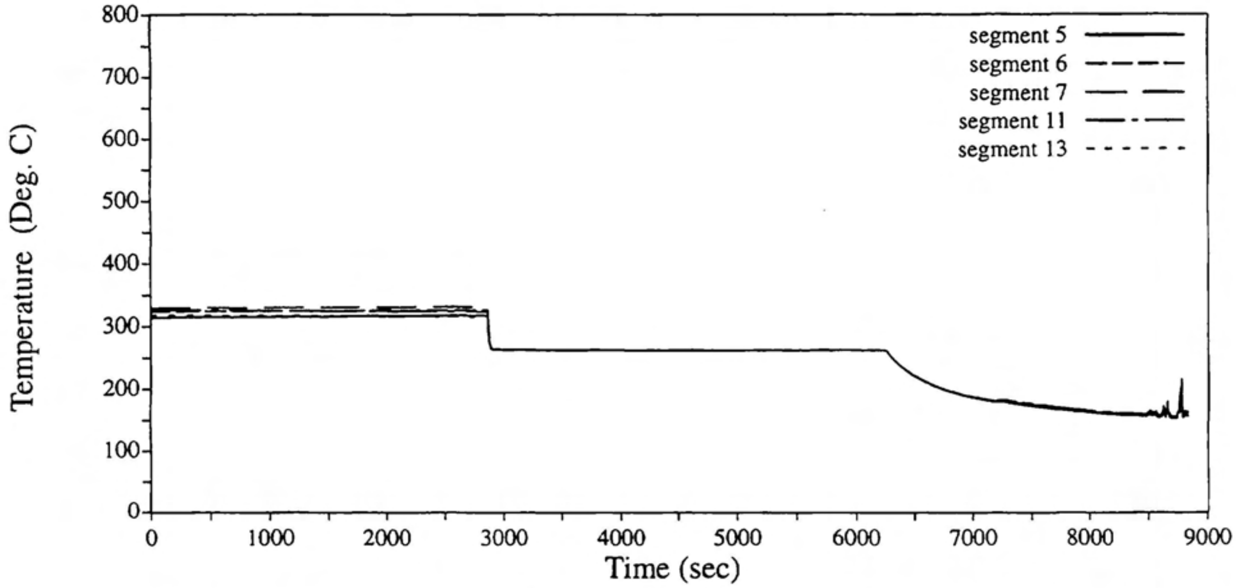


Figure 19 Fuel Sheath Temperatures of the Single Channel Model – Pass 4 Header Conditions For SGTR With LOECC and No Operator Intervention