Proceedings of the Korean Nuclear Society Spring Meeting

Kori, Korea, May 2000

Development of a System Code MEDUSA and Its Application to the Feedwater Line Break

Sang Yong Lee, Cheol Shin Lee, and Chul Jin Choi

Korea Power Engineering Company, Inc.

Safety Analysis Department, NSSS Engineering and Development

150 Duck jin-dong, Yusong-gu, Taejon, 305-353, Korea

Abstract

A computer program, MEDUSA, has been being developed as a thermal-hydraulic analysis tool for the nuclear power plant system. It is based on the COBRA-TF computer program that deals mainly with the two-phase core thermal-hydraulics of the vertical channels. The noteworthy improvement is to introduce the Main Steam Safety Valves and the Pressurized Safety Valves for the Non-LOCA type accident analysis in addition to the features described in the reference [2].

The system analysis capability has been being evaluated by applying the code to the various analyses of the system transients. As a continuing effort from the LOCA analysis [2], Feedwater Line Break (FLB) accident for the typical pressurized water reactor is selected for this study. The analysis results are compared with those of the design code CESEC-III for several important system parameters. This comparison shows that there is no problem to apply the new code to the FLB type accident analysis.

1. Introduction

The computer program, COBRA-TF[1], is developed mainly for the application to the analysis of the core two-phase thermal-hydraulics. On the other hand, COBRA/TRAC[1], is constructed to utilize the above capability in the system analysis by combining COBRA-TF with system code TRAC-PD2[3]. It is also worthy of mentioning that a well-known system code, RELAP5[4], is also combined with COBRA-TF[5].

Although the combinations of codes are theoretically possible and justifiable, there still remain a lot of practical issues such as, the burden for users, the inconsistency of numerics, and the maintenance problem. With these potential issues, and also with the fact that there is no practical problem to extend COBRA-TF itself to system code, MEDUSA development project has been set forth. Since the detailed descriptions of the improvements are already presented [2], only the description of the one-channel section is to be repeated in this paper.

The system analysis capability of MEDUSA has been being evaluated by applying the code to the various accidents. The brief results of the large break LOCA analysis has been presented in the reference [2]. In this paper, the analysis result of FLB for the typical pressurized water reactor will be described.

In Section 2, the descriptions for the improvements will be presented and follows a presentation of the results of the plant application. Then the conclusions will be drawn.

2. Description of the Improvements

2.1 Introduction of the One-Channel Section

The necessity of the horizontal channel comes from the fact that the present reactor systems have the configurations that connect vessels with pipes such as cold and hot legs of Pressurized Water Reactors (PWR). Therefore, having the capability to model the horizontal one-dimensional piping is essential for the system modeling. With the original COBRA-TF, the modeling of the horizontal one-dimensional channel might be done by laying out a series of one node vertical channel and connecting them by gaps. However, this approach has a great deal of problems such as bulky input preparations, clumsy output editions and lack of flexibility to model the pipe bends.

These problems can be avoided by introducing the one-channel section that connects the vertical sections. One channel section which consists of only one channel becomes very flexible to model one-dimensional pipes. The gravitational term, $g_j \Delta x_j$ in the momentum equation of the one channel section is modified to $h_j g_j \Delta x_j$, where h_j is the gravitational sense, g is the gravitational constant and Δx_j is the node length. The gravitational sense is to be specified by input. The horizontal nodes are given zero while the vertical nodes can be given either 1.0 or –1.0 depending on the upward or downward nodes respectively. This change is implemented in the section data input specifications.

More important feature of the one-channel section is that it provides the connection mechanism between the vertical channel and the horizontal channel. Although the original COBRA-TF can be used to model only one connected vertical sections, the implementation of the one-channel section with the connection mechanism makes it possible for users to model any number of connected vertical sections that are connected by pipes. Figure-1 shows the case in which two vertical sections are connected by a one channel section

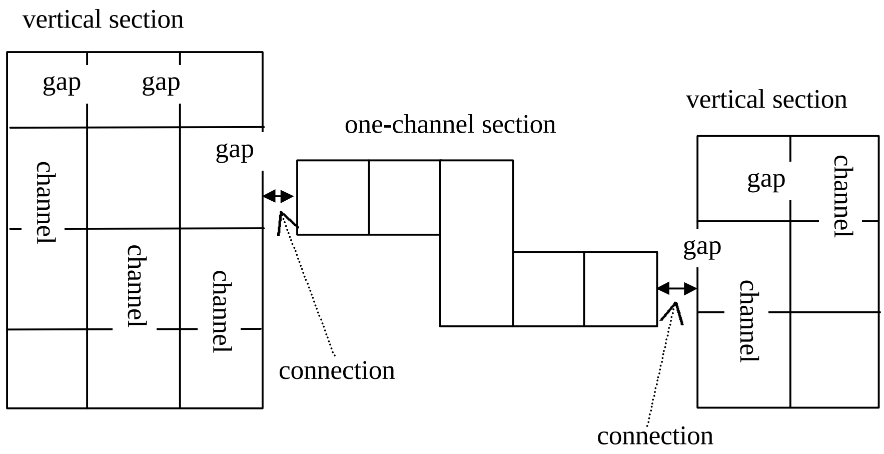

Figure-1. One Channel Section

Devising the method to combine the axial cell side face of the vertical channel and the axial cell top face of the one-channel section can do the implementation of the connection mechanism. When the axial cell top face of the one-channel section is connected to the vertical cell side face, the vertical cell side face is to be declared as a connection gap at the input preparation stage. The modified code, named MEDUSA, identifies the connection gap and the axial cell face of the one-channel section during the solution procedure.

There are two types of momentum equations solved in COBRA-TF, one is the axial momentum equation for the axial cells, and the other is the transverse momentum equation for the transverse cells. Since both of the transverse and axial momentum cells are involved at the connection point, one of them should be chosen to avoid any conflicts. The gap momentum equation solving procedure, which comes first in the original COBRA-TF solution scheme, is chosen to be active while the vertical momentum equation solving procedure is bypassed for the connection point. The subsequent solution procedure of MEDUSA is the same as that of the original COBRA-TF. The explicit flow rates and the pressure derivative of the flow rates are obtained for the individual momentum cells including the connections. The pressure matrix for the system is, then, constructed and solved to get the pressure increments for the individual nodes. The final flow rates for the individual connections are updated with the final pressures of the connected cells. Then, the dependent variables are updated for the individual cells.

2.2 Pressurizer and Main Steam Safety Valves

One of the major functions to mitigate the abrupt pressure excursion during the feedwater line break accident is the steam release through the Pressurizer Safety Valves (PSVs). In addition, the Main Steam Safety Valves (MSSVs) of the intact steam generator removes the core power after Main Steam Isolation Valves (MSIVs) are closed by the low steam generator pressure signal. General operation characteristics of the spring-loaded safety valve can be summarized as follows; when the system pressure reaches the valve opening setpoint, a certain fraction of total valve area is opened. If the system pressure increases further and reaches the setpoint, which is called the accumulation setpoint, then the valve is fully opened. The valve closure characteristic differs from the opening characteristic described above. If the system pressure decreases due to the opening of the safety valve, the valve does not close at the opening setpoint, but closes at the setpoint somewhat less than the opening setpoint, which is called blowdown setpoint. These operation characteristics for the spring-loaded safety valve have been encoded in the MEDUSA as subroutines. These subroutines calculate the valve area and use it as the boundary condition of the channel. Critical flow model already implemented for LOCA simulation was used to calculate the mass flow rate through safety valves.

3. Applications to KNGR

3.1 Input Preparation for the System Transient Analysis

Korean Next Generation Reactor (KNGR) is a logical up-rate design of the well-proven Korean Standard Nuclear Power Plant (KSNP). General arrangements for the main heat transport system configurations are the same for both plants. A reactor vessel, two hot legs, four cold legs with four pumps, two steam generators and a pressurizer are the main components.

The system nodalization is shown in Figure-2. The reactor vessel consists of 5 sections. One channel is assigned to the downcomer. The core is divided by two channels, one for normal channels (ch3), and the other for hot channels (ch4). All hot and cold legs are modeled as horizontal channels that connect the vessel with steam generators. One hot leg (HL-A) is connected to a pressurizer through a horizontal channel. The accumulator is connected to the upper downcomer (ch12) through a horizontal connection. The HPSI is also connected to the upper downcomer. The Reactor Coolant Pumps (RCPs) are located at the channel 71 node 4, at the channel 72 node 4, at the channel 73 node 4 and at the channel 74 node 4. Each steam generator has four sections, the first section for inlet/outlet plenum, the second for u-tubes, riser and downcomer, the third section for separator and the last one for steam dome. A constant pressure boundary condition is set at the exit of steam dome while the constant flow boundary condition is set at the entrance of the riser channels (ch37 and ch57). A valve is installed at the exit, which is closed as breaks open

One of the three fuel rods is located at channel 3 to simulate the average fuel. The other two fuels are located in channel 4 that simulate arbitrarily chosen 9 hot channels. One of them has the power of 115% of the average power. The other that represents the hot pin has the peaking factor of 1.55. Since MEDUSA has no reactor kinetics solver, power table from the RELAP5 run[6] is used. Thirty-four heat slabs are used to model the heat structures such as piping and vessel walls. Four of them are u-tubes of steam generators.

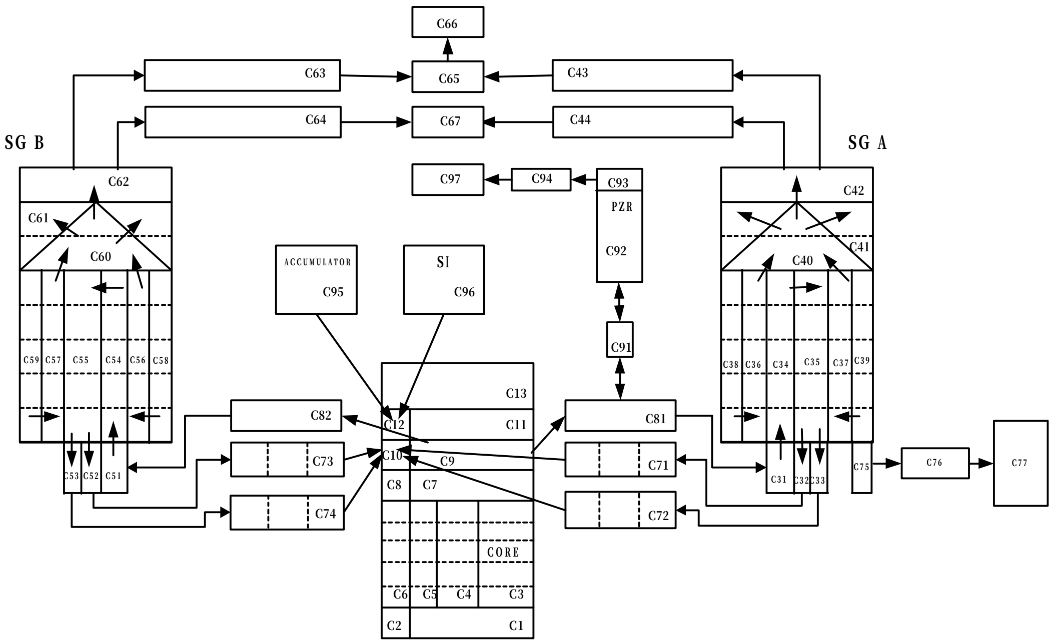

Figure-2. System Nodalization

3.2 Input Preparation for the Feedwater Line Break

FLB event, which results in the dramatic decrease in the heat removal by the secondary system, was selected to evaluate the applicability of the MEDUSA computer program in simulating the thermal hydraulic responses to the Non-LOCA events. Primary and secondary system safety valve model has been newly implemented to simulate the mitigation of pressure excursion in primary and secondary systems as described in Section 2. In addition, the main steam system and the feedwater system have been nodalized to incorporate the feedwater line break and the main steam isolation actuation.

Figure-2 shows the system nodalization for FLB revised from that of LOCA presented reference [2]. Channels for the main steam system and safety valves have been added to the nodalization for LOCA. Channels for modeling the guillotine break in cold leg have been removed. Feedwater line break has been modeled by addition Channels 75 through 77 to steam generator A. Critical flow is calculated at the momentum cell in Channel 76. Channels 43, 63, 65, and 66 have been added to model the main steam piping, the main steam header and the turbine. Valve models are implemented to the momentum cells of Channel 43 and 63 as boundary conditions. Main Steam Safety Valves (MSSVs) are modeled by addition of Channels 44, 64, and 67 and Pressurizer Safety Valve (PSV) by Channels 93, 94, and 97. Critical flow model is adopted at the momentum cells in Channels 44, 64, and 94. In addition, Channel 92 representing pressurizer has been divided into 9 cells (scalar cells) to refine the complicated two-phase phenomenon in this specific region.

3.3 Results and Discussions

The results of MEDUSA have been compared with those of the CESEC-III computer program [7], a licensing program for Non-LOCA thermal hydraulic simulation of CE-type plants. The break size of 0.6 ft^2 is selected. Initial conditions for major parameters provided in Table-1. They are tried to be made similar as close as possible for both programs. Among various probable reactor trip functions expected to occur during FLB, only high pressurizer pressure trip function was credited.

Table 1. Comparison of Initial Conditions

Parameter	CESEC-III	MEDUSA
Primary System Power, MWt	4,000	4,000
Core Inlet Temperature, ^{O}F	554	555.7
Core Flow Rate, lbm/sec	44,960	41,642
Pressurizer Pressure, psia	2,251	2,255.6
Pressurizer Liquid Volume, ft^3	900	900
Steam Generator Pressure, psia	1,003.1	1,000.3
Break Area, ft^2	0.6	0.6

After null transients of 300 seconds for steady state, feedwater line break is initiated by opening the valve for the break. The variation of Reactor Coolant System (RCS) pressure is shown in Figure-3. High pressurizer pressure trip setpoint of 2460 psia is reached at 25.4 seconds and 53 seconds for CESEC-III and MEDUSA, respectively. The results show that the pressure increase rate of CESEC-III is faster at the early stage of the transient even though the peak pressures are almost same in value. The pressure excursion characteristic is mainly dependent on the primary-to-secondary heat transfer.

Figure-4 provides the variations of the heat transfer rates of both intact and affected steam generators. The affected steam generator heat transfer rate for CESEC-III decreases rapidly at about 23 seconds while the results of COBRA-TF shows much smoother reduction. This is considered to be the major cause of the different RCS pressure responses of both computer programs. In CESEC-III, it is assumed that saturated liquid is discharged until no liquid remains in the affected steam generator. After that, saturated vapor is assumed to be discharged. This artificial phase discontinuity in discharged fluid can be seen in the Figure-5, which provides the variations of the mass flow rate through the break. In reality, the discharged secondary coolant would be in two-phase condition and the mixture enthalpy would be higher than that of pure saturated liquid. The anticipated two-phase condition of the discharged fluid can be simulated realistically in MEDUSA. This assumption in CESEC-III results in conservatively faster blowdown of the affected steam generator and conservatively less heat removal by the secondary system in the viewpoint of RCS pressurization. In addition, heat transfer area was conservatively assumed to decrease abruptly when the SG inventory is almost empty in CESEC-III. The difference in the break flow models adopted in both programs has an influence on the heat transfer behavior of the steam generator.

Steam generator pressure variations are shown in Figure-6. Different responses in RCS pressure affect the behavior of steam generator pressures. The timing of Main Steam Isolation Signal (MSIS) generation proceeds that of the reactor trip for MEDUSA. However, those two events are in opposite order for CESEC-III. The increase rate of the intact steam generator pressure for MEDUSA is much faster because the core power remains at full power level after MSIS.

As a summary, the overall thermal-hydraulic responses to the FLB transient have been evaluated to be justifiable and explainable even though some difference in timing of major sequence of events. The difference stems largely from the conservative assumptions adopted in licensing computer program. The differences in the various correlations including the heat transfer models and the critical flow models also contribute to the different results.

4. Conclusions

The system analysis capability of MEDUSA has been being evaluated by applying the code to the various accidents. The simulation of a Non-LOCA transient, FLB, shows that MEDUSA can be utilized for Non-LOCA system analysis. However, some more Non-LOCA transients such as Steam Line Break and Steam Generator Tube Rupture are required to be evaluated before any final conclusion can be made.

In parallel, some urgent improvements have to be made to use this code as a Non-LOCA analysis tool. The incorporation of the models, such as, the critical heat flux correlations, the reactor kinetics solver and the generalized control logic, are the examples.

References

1. M. J. Thurgood, et. al., “COBRA/TRAC - A Thermal-Hydraulics Code for Transient Analysis of Nuclear Reactor Vessels and Primary Coolant Systems”, NUREG/CR-3046, 1982.

2. S. Y. Lee, et. al., “Development of a System Code by Extending the Capability of COBRA-TF”, KNS Autumn Meeting, Seoul, Korea, 1999.

3. D. R. Liles et. al., “TRAC-PD2, An advanced Best-Estimate Computer Program for Pressurized Water Reactor Loss-of-Coolant Accident Analysis”, NUREG/CR-2054, 1981.

4. K. E. Carlson, et. al., “RELAP5/MOD3 Code Manual”, NUREG/CR-5536, 1990.

5. S. Y. Lee, et. al., “COBRA/RELAP5, A Merged Version of COBRA-TF and RELAP5”, Nucl. Technol., Vol.99, pp177, 1992.

6. H. R. Choi, et. al., “Large Break LOCA Analysis for KNGR Using R5V322beta”, The 12th CAMP Working Group Meeting, 1999.

7. CESEC-III; Digital Simulation of a Combustion Engineering Nuclear Steam Supply System, CENPD-107, 1974.

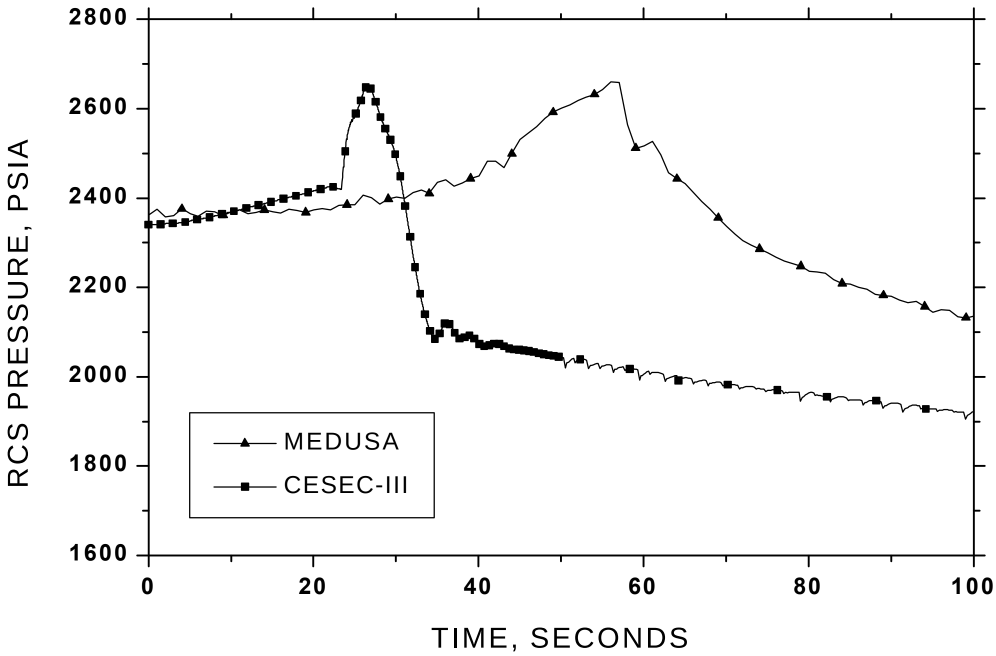

Figure-3. Reactor Coolant System Pressure Variations

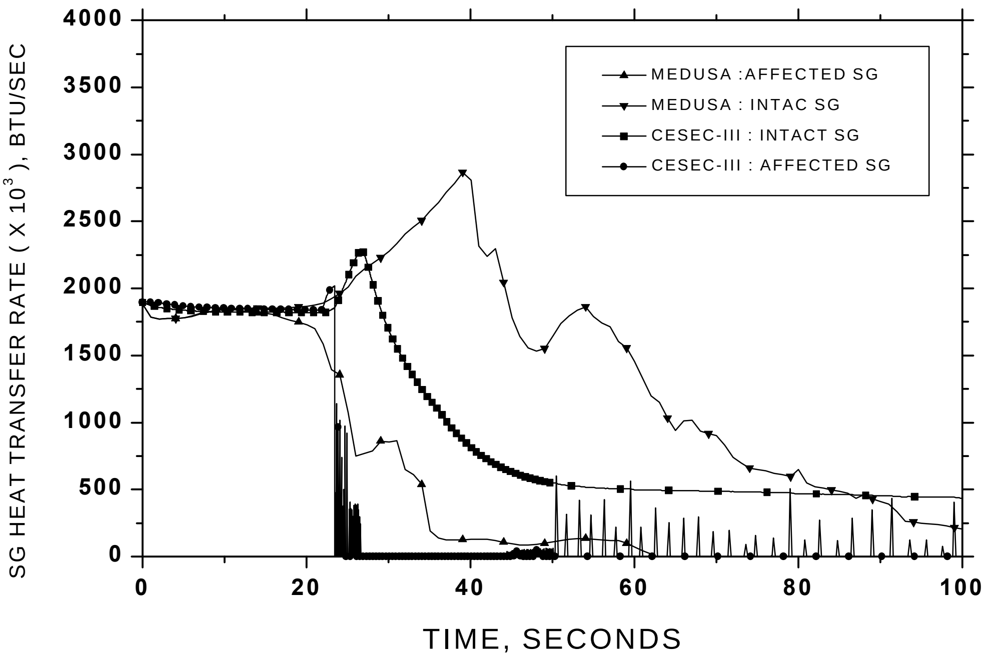

Figure-4. Primary to Secondary Heat Transfer Rate Variations

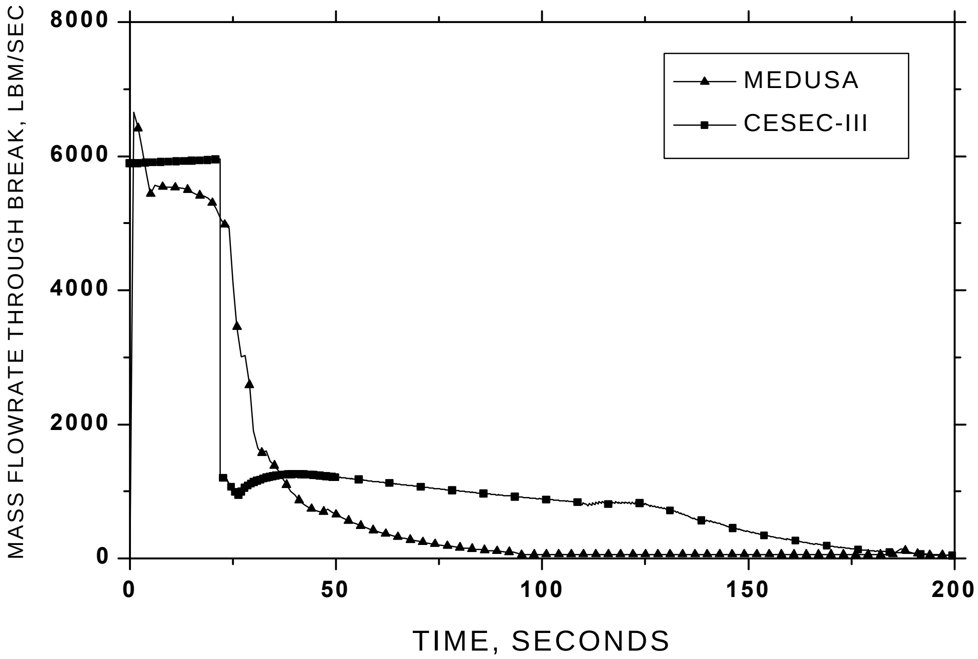

Figure-5. Mass Flow Rate through Break Variations

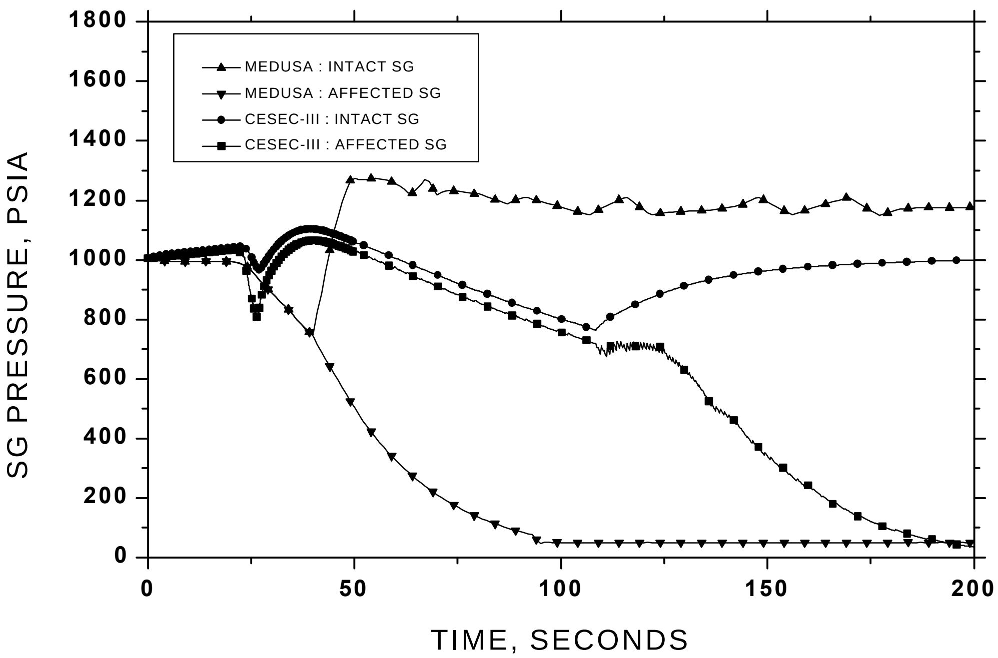

Figure-6. Steam Generator Pressure Variations