*Proceedings of the Korean Nuclear Society Spring Meeting*
*Cheju, Korea, May 1996*

# MIDLOOP Code Analysis of a ROSA-IV/LSTF Experiment for the Loss of Residual Heat Removal System Event During Mid-loop Operation

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## Abstract

The MIDLOOP code has been developed for the evaluation of RCS pressurization transients initiated from a loss-of-Residual Heat Removal System (RHRS) during mid-loop operation after reactor shutdown. It provides a fast running and realistic tool for studying parametrically the response of important plant parameters such as pressure, temperature, and level to various plant combinations of the primary side vent, makeup, and leakage procedures and the steam generator (SG) conditions. The code consists of ten nodes representing the primary and secondary sides of a nuclear power plant and can analyze the effect of air on the primary system pressurization and primary to secondary heat transfer. The analysis results of the MIDLOOP code are in good agreement with the ROSA-IV/LSTF experiment without opening in the RCS.


## 1. Introduction

Following the Diablo Canyon loss-of-RHRS event on April 10, 1987[1], the USNRC published Generic Letter (GL) 88-17[2] where the NRC issued recommendations to implement 'expeditious actions' and 'program enhancements'. In order to adequately address GL 88-17 recommendations, especially in the areas of technical specification improvements, procedures, equipment availability and analyses, some analysis tools for the loss-of-RHRS event during mid-loop operation are needed.

The plant during the mid-loop operation has various plant configurations such as levels of the RCS and SG, time after shutdown (decay heat), vent, makeup, and leakage of the primary side, and the SG conditions. Therefore, many sensitivity studies are required to address the GL 88-17 recommendations. However, use of RELAP5 code[3] which can predict the behavior of noncondensable gas is restricted because of the very long computer CPU time taken to analyze[4]. This is the need of the MIDLOOP code development. The MIDLOOP code has simplified nodes and assumptions to provide fast running than the RELAP5 code and the capability of parametric study with respect to important plant parameters such as pressure, temperature, and water level during the transient.

The objective of the present study is to verify the MIDLOOP code developed as an analysis tool of the loss-of-RHRS event during the mid-loop operation.

## 2. MIDLOOP Code Descriptions

Figure 1 shows the MIDLOOP nodalization for the ROSA-IV/LSTF. Since the air/steam spaces of the RCS for the ROSA-IV/LSTF are blocked by the core support plate, the RCS and SG are represented by six nodes and four nodes, respectively. Each of primary and secondary sides is divided into the hot and cold sides. The constituents of the primary and secondary side regions are water, steam, air, and metal.

Major assumptions used in the MIDLOOP code for simplifying the basic differential governing equations are as follows:

- The cold leg loop seals separate the hot and cold sides of the RCS and the code does not address the effect of the loop seal dynamics. The reactor core is always covered with water or two-phase mixture.
- Once saturation condition is achieved in the liquid portion of the primary or secondary side, steam and water are assumed to be in thermodynamic equilibrium.
- Air is assumed to have the same temperature as the SG hot side for the case with active SG. For the case with no active SG, it is assumed to be identical with initial temperature of region containing water and constant during the transient.
- Heat transfer through the metal wall from the hot side to the cold side is not considered. This assumption results in faster heatup in the hot sides of the RCS and SG.
- For no RCS vent case, the noncondensable gas in the RCS hot side behaves like the piston by the steam generated due to the core boiling and migrates to the active SG U-tubes. The noncondensable gas occupying the RCS cold side and pressurizer remains there during the transient.
- In the noncondensing region of the SG U-tubes, steam and a noncondensable gas are mixed together and have the same temperature as the secondary side of the SG.

The basic differential equations governing mass conservation for each RCS or SG node in Figure 1 were derived by considering the manometric flow between the hot and cold sides of the RCS and SG, steam condensation in the SG U-tubes, steam bubble rise in the hot side of the RCS and SG, makeup flow and leakage flow into the hot leg or cold leg, migration of steam or air through the control rod guide tubes, steam or air vent of the RCS and SG, and auxiliary feed water flow of the SG. The time derivatives of enthalpy for the hot and cold sides of the RCS and SG in Figure 1 were determined by considering the core heat generation and heat transfer through the SG U-tube and metal, heat loss by the vent or leakage, and heat addition by the makeup flow. In order to determine pressures of the hot and cold sides of the RCS and SG, the equations governing volume conservation for hot and cold sides were used.

The ordinary differential equations derived were solved by using a sixth-order

Runge-Kutta method. The global error in numerical computations was $1.0\times10^{-4}$.

Two modes of heat transfer were assumed on the primary side U-tubes: Nusselt falling film condensation when only vapor phase (and possibly noncondensable gas) enters and condensation of a two-phase mixture when the froth level, because of mixture-level swelling, reaches to the SG U-tubes. On the secondary side, heat transfer correlations for natural convection under both nonboiling and boiling conditions were used[5].

The lumped-heat-capacity analysis assuming a uniform temperature distribution throughout the solid body was used in the wall heat model. Temperatures of metal (cladding or base metal) in water regions were assumed to be identical with those of the water. Metals in nodes 5 and 6 of Figure 1 were assumed to have the same temperature as the water in node 2. For the case with an active SG, temperatures of metal in steam or air/steam regions of the RCS and SG hot sides are assumed to be identical with that of the water in SG hot side (region 7 of Figure 1). If there is no active SG, heat transfer to metals in the steam and air/steam regions of the RCS and SG hot sides is neglected for conservatism. Metals came into contact with both water regions of the RCS hot and cold sides such as core shroud were assumed to be equally divided into each region. For conservatism, the metal effect of fuel rods was not considered. The heat transferred to the secondary fluid was assumed to be used for heatup of the SG U-tubes, water and metal of the SG hot side.

The two-phase mixture levels in the hot and cold sides of the RCS and SG are determined from the mass conservation equations. Homogeneous Equilibrium Model (HEM) or orifice model was employed for break flow.

The movement of the noncondensable gas by the steam was modelled by the so-called Piston Model[5,6]. In order to determine the condensing length of the SG U-tube inlet, the Gibbs-Dalton law of partial pressure and an energy balance across the SG U-tubes were employed. Based on the N00 experiment[7], it was assumed that the initial air in the reactor vessel hot side and hot leg and SG inlet plenum of an active SG migrated to the passive region (U-tubes and outlet plenum) of the active SG.

The Wilson bubble rise correlation[8] was employed to calculate the superficial steam velocity after the void fraction was calculated from known pressures and qualities of nodes 1 and 7.

## 3. Results and Discussions

The Large Scale Test Facility (LSTF) of the Rig of Safety Assessment (ROSA)-IV program simulates a Westinghouse-type four-loop 3423 MWt PWR by a volumetrically-scaled (1/48) full-height, two-loop model. In the present study, an experiment without RCS opening (N00) is selected[7]. The facility, schematically shown in Figure 2, includes a reactor vessel, two symmetric primary loops, a pressurizer, and ECCS including a RHR system.

Unlike a KAERI designed reactor vessel, there is no communication between the downcomer and upper plenum/upper head through control rod guide tubes in the LSTF reactor vessel except a very low reactor vessel water level condition.

Algorithms which investigates SG U-tube flooding (if $j_g^* > 0.5$), horizontal stratification in the hot legs (if $V_g > V_{cr}$), flooding at the hot leg bend ($K_{g,HL} > 3.2$), and flooding in the surge line (if $K_{g,SRG} > 3.2$) were used in the code. For simplicity, it was assumed that there is no flow through inactive loops with inactive SGs. Calculations of $j_g^*$, $V_g$, $K_{g,HL}$, and $K_{g,SRG}$ for the ROSA-IV/LSTF experiment were performed by varying the number of active loops ($N_{act}$) or water levels at atmospheric pressure. The results are given in Table 1.

Since all of $j_g^*$ in the Table 1(a) is less than 0.5, flooding in the SG U-tubes for the ROSA-IV/LSTF experiment is unlikely under the mid-loop operation condition. Table 1(b) shows that for $h/D_{HL} = 0.5$ (where h = water level from the hot leg bottom), the loss of horizontal stratification is possible with only one active loop. The results for $h/D_{HL} = 0.75$ indicate that loss of horizontal stratification is possible for all cases considered. From the Table 1(c), flooding at the hot leg bend is unlikely. The fraction of the core generated steam entering into the surge line was varied to determine flooding tendencies. Table 1(d) demonstrates that if all the steam generated in the core enters into the surge line, flooding in the surge line is possible. If 22 percents of the steam produced in the core enters into the surge line, flooding can be occurred.

The detailed initial conditions of the test are summarized in Table 2 and also illustrated in Figure 2. Major event chronology is listed in Table 3.

It takes run time of 30060 seconds to calculate 20000 seconds transient by 100 MHz Pentium™ PC. The run time is thus about 1.5 times as much as the physical time. Figure 3 shows the predicted and measured pressure transients for the N00 experiment. Figure 4 illustrates the predicted and measured temperatures of the RCS and SG. Though the surge line flooding and the heat transfer to the downcomer by the steam condensation on the upper plenum wall were not modeled, the pressure trends predicted are good agreement with those of the experimental data. The prediction of the RCS cold side pressure underestimated the ROSA-IV/LSTF data a little as shown in Figure 3 because heat transfer to the downcomer by the steam condensation on the upper plenum wall was neglected. Also, the prediction of the RCS hot side pressure between the initiation of steam condensation and secondary coolant boiling underestimated the ROSA-IV/LSTF data a little as shown in Figure 3 because the decrease of RCS hot side water due to the insurge flow into the pressurizer was not modeled. The maximum pressurizer water level in the experiment was about 0.35 m[7]. The corresponding water volume in the pressurizer was about 0.0746 $m^3$ which is about 8.2 % of RCS hot side water volume and the corresponding differential pressure between the hot leg and pressurizer was about 3230 Pa at the hot leg pressure of

0.21 MPa.

In the U-tubes of the SG-B with the secondary cooling, steam condensation continued and induced steady steam flow from the upper plenum through the hot leg. No accumulation of condensate by the CCFL was observed in the SG U-tubes as shown in Table 1. Between initiation of steam condensation and the secondary coolant boiling, about 45 % of heat generated in the core was removed by this condensation(7). Figure 5 shows the predicted primary to secondary heat transfer rate.

From calculational results, it was found that air in an inactive SG, a hot leg, a pressurizer, and RCS cold side did not migrate to the passive region of an active SG. Figure 6 represents the effect of the noncondensable gas migrated to the passive region of an active SG on the pressure transient of the RCS. The migration of air in the inactive loop for CASE 1, in the pressurizer and surge line for CASE 2, and in the RCS hot side for CASE 3 to the passive region of an active SG was assumed. The predicted maximum steady state pressure was about 0.73 MPa of CASE 3.

## 4. Conclusion

The MIDLOOP code has been developed for evaluation of RCS pressurization transients initiated from a loss-of-RHRS during the mid-loop operation. The analysis results were in good agreement with the experimental data. Though models such as water hold-up in the pressurizer and SG U-tubes, metal heat transfer in the air/steam regions and between both liquid regions, and loop seal dynamics were not modeled in the code, the code was an effective analysis tool for the loss-of-RHRS events initiated during the mid-loop operation.

## 5. References


1) J.L. Crews et al., NUREG-1269, June 1987.
2) U.S. NRC Generic Letter 88-17, October 17, 1988.
3) C.D. Fletcher and R.R. Schultz, NUREG/CR-5355, EGG-2596, Draft, 1991.
4) K.S. Han and J.H. Song, J. of the Korean Nuclear Society, pp.645-660, October 1995.
5) S.A. Naff et al., NUREG/CR-5855, EGG-2671, April 1992.
6) D.E. Palmrose and R.M. Mandl, NUREG/CP-0119, Vol.3, 1991.
7) H. Nakamura, et al., ASME Winter Meeting, FED-Vol.140, 1992.
8) Joint US/Euratom R&D Program at (11-1)-1186, ACNP-65002, April 15, 1965.

Table 1 Calculations of $j_g^*$, $V_g$, $K_{g,HL}$, and $K_{g,SRG}$ for the ROSA-IV/LSTF N00 Experiment

(a) SG U-Tube Flooding

| $P_l$ (Pa) | 101325 | |
|---|---|---|
| $N_{act}$ | 2 | 1 |
| $j_g^*$ | 0.214 | 0.427 |

(b) Horizontal Stratification in the Hot Legs

| Water Level (m) | 0.25 x $D_{HL}$ | | 0.50 x $D_{HL}$ | | 0.75 x $D_{HL}$ | |
|---|---|---|---|---|---|---|
| $V_{cr}$ (m/s) | 36.53 | | 17.86 | | 6.0 | |
| $N_{act}$ | 2 | 1 | 2 | 1 | 2 | 1 |
| $V_g$ (m/s) | 5.89 | 11.77 | 9.47 | 18.94 | 24.22 | 48.44 |

(c) Flooding at the Hot Leg Bends

| $N_{act}$ | 2 | 1 |
|---|---|---|
| $K_{g,HL}$ | 0.76 | 1.51 |

(d) Surge Line Flooding

| Fraction of steam | 1.0 | 0.22 |
|---|---|---|
| $k_{g,SRG}$ | 14.46 | 3.1812 |

Table 2 Initial Conditions of the ROSA-IV/LSTF N00 Experiment

Primary Loops

| | |
|---|---|
| Pressure | Atmospheric |
| Fluid Temperature(K) | Hot legs : 323<br>Cold legs: 305 |
| Liquid Level | Middle of hot leg |

SG Secondary Side

| | |
|---|---|
| Pressure | Atmospheric |
| Liquid Level(m) | SG-A: Empty<br>SG-B: 10 |

Other Conditions

| | |
|---|---|
| Opening | No opening |
| to simulate | Closed system |
| SG Nozzle Dam | No dam |
| Core Power | 0.6% (430 kW) |
| ECCSs | Manual injection for Tcore > 700K |
| Others | SG-B secondary pressure control |

Table 3 Major Event Chronology for the N00 Experiment and MIDLOOP Analysis

| ROSA-IV/LSTF N00 Results | | MIDLOOP Results |
|---|---|---|
| Time(sec) | Events | Time(sec) |
| 0 | Loss-of-RHRS | 0 |
| 586 | Incipient boiling in the core | - |
| 729 | Saturation of upper plenum liquid | 780 |
| 1586 | Steam entrance into pressurizer | - |
| 2729 | Initiation of steam condensation in SG-B | 1890 |
| 8086 | Steam temperature reached in pressurizer | - |
| 12590 | Initiation of secondary coolant boiling in SG-B | 13080 |
| 16300 | Initiation of SG-B secondary pressure control at about 0.2 MPa | 15340 |
| 20000 | End of experiment | 20000 |

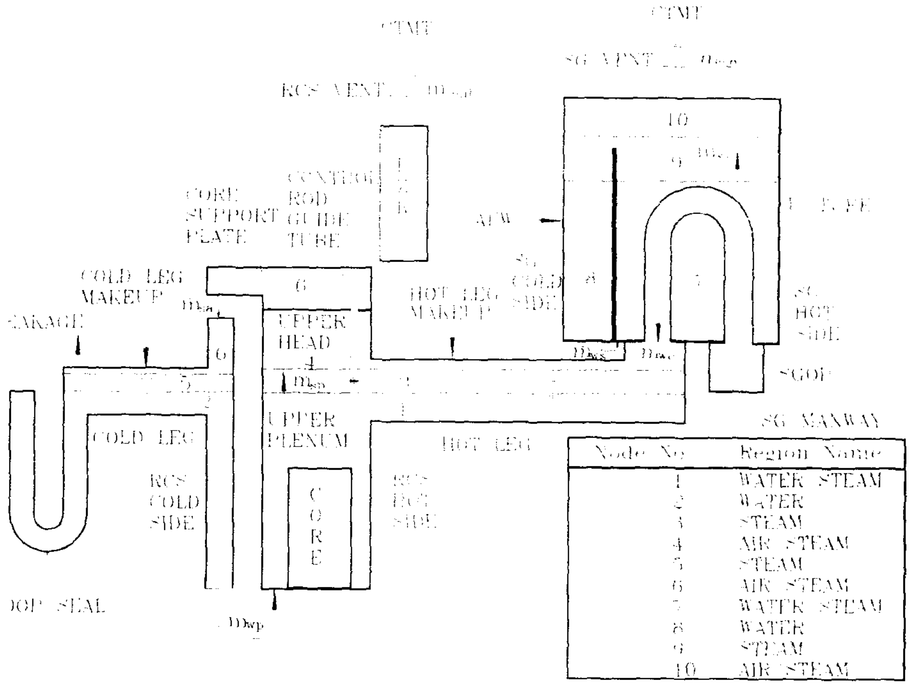


Figure 1 MIDLOOP Nodalization for the ROSA-IV/LSTF

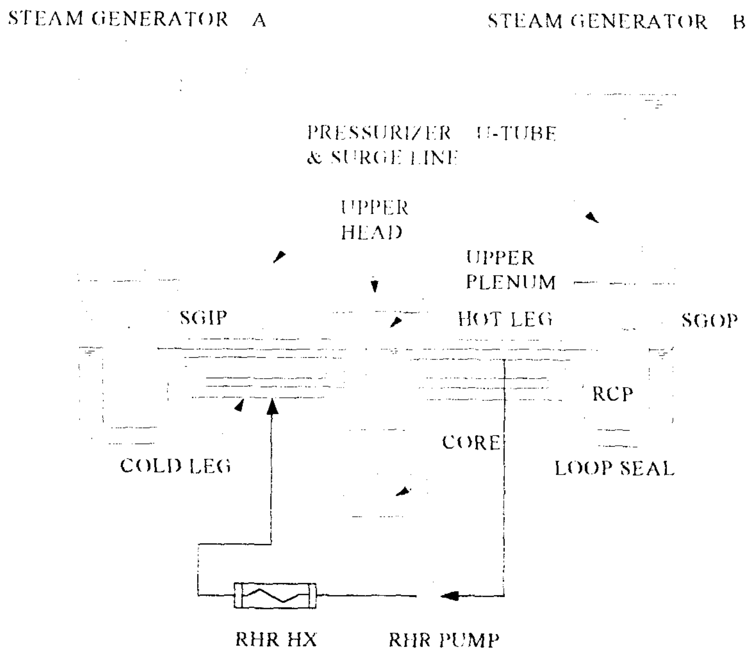


Figure 2 Schematic View of the ROSA-IV/LSTF

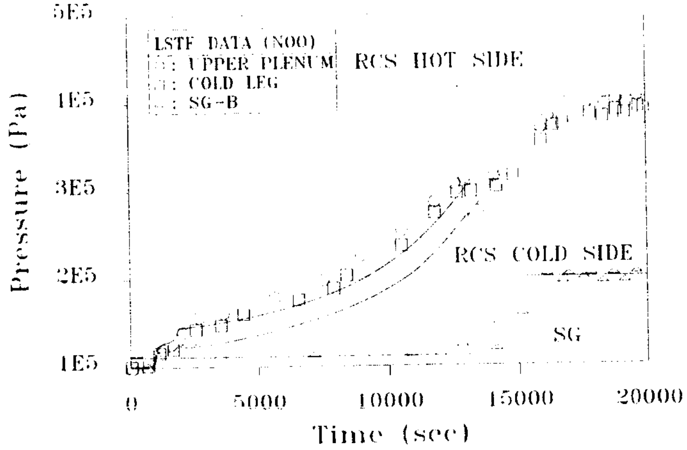


Figure 3 Pressure Transients at the RCS and SG

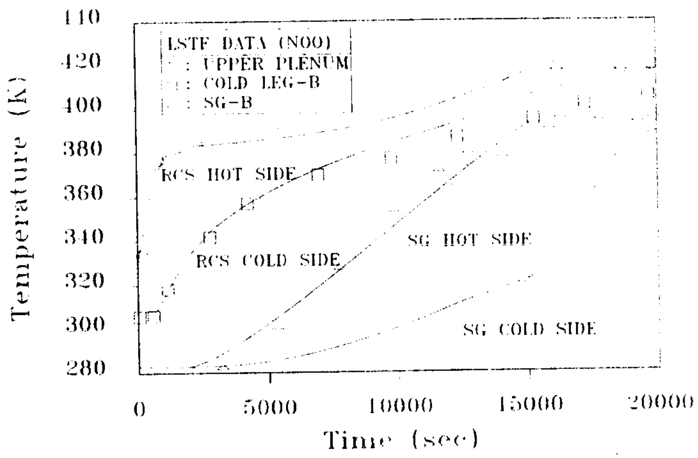


Figure 4 Temperature Transients of the RCS and SG

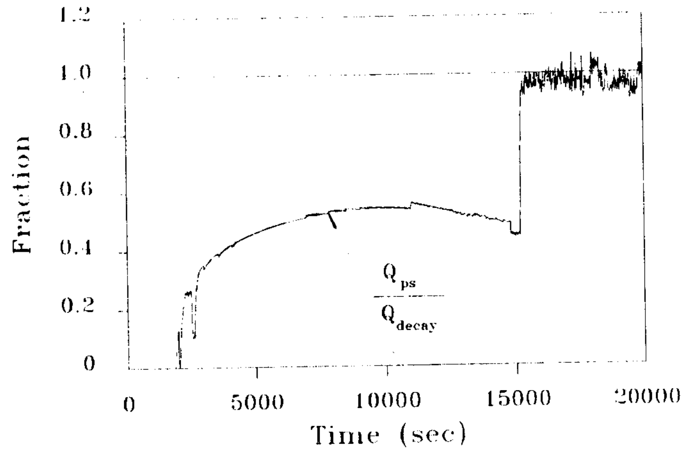


Figure 5 Primary to Secondary Heat Transfer

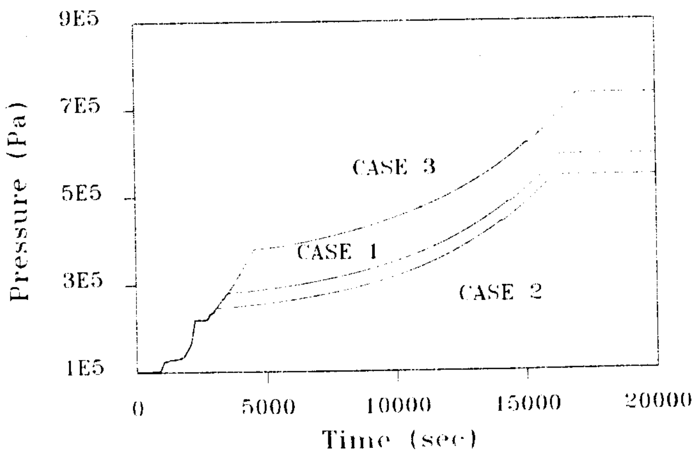


Figure 6 Effect of the Air Migration to the Passive Region of an Active 'G on the Primary Side Pressure