Proceedings of the Korean Nuclear Society Spring Meeting
Cheju, Korea, May 2001

Analysis of MSGTR Events for APR1400 by Means of Best Estimate Thermal-Hydraulic System Code

Ji Hwan Jeong
Cheonan College of Foreign Studies
Anseo-dong, Cheonan, Choongnam, 330-705 Korea

Sang Jae Kim
Korea Advanced Inst. of Science and Technology
Kusong-dong, Yusong-gu, Taejon, 305-701 Korea

Keun Sun Chang
Sunmoon University
Tangjeong-myeon, Asan, Choongnam, 336-708 Korea

Jae Hun Lee
Korea Institute of Nuclear Safety
Kusong-dong, Yusong-gu, Taejon, 305-338 Korea

Abstract

A multiple steam generator tube rupture (MSGTR) event has never occurred in the history of commercial nuclear reactor operation while single steam generator tube rupture (SGTR) event is reported to occur every two years. As there is no history of MSGTR event, the understandings of transients and consequences of this event are not so much. In this study, a postulated MSGTR event in advanced power reactor 1400 (APR1400) is analyzed using thermal-hydraulic system code. The APR1400 is a two-loop, 1000 MWe, PWR supposed to be built in 2009. MARS1.4 is used in this study. The present study aims to understand the effects of rupture location in heat transfer tubes and selection of affected steam generator following a MSGTR event.

The effects of five tube rupture locations are compared with each other. The comparison shows that the response of APR1400 is to allow shortest time for operator action following a tubes rupture in the vicinity of hot-leg side tube sheet and to allow longest time following a tube ruptures at the tube top. The MSSV lift time for rupture at tube-top is evaluated as 24.5% larger than that for rupture at hot-leg side tube sheet. Also, the MSSV lift time for four cases are compared in order to examine how long operator response time is allowed depending on which steam generator is affected. The comparison shows that the cases for both of two steam generators are affected allow longer time for operator action compared with the cases that a single steam generator is affected. Further more, the tube ruptures in the steam generator where a pressurizer is linked leads to the shortest operator response time.

1. Introduction

A steam generator tube rupture (SGTR) in nuclear power plants is of important safety concern as it means a loss of the barrier between primary coolant and secondary coolant. If any tube of a steam generator (S/G) breaks, high-pressure primary coolant could leak to the secondary side so that radioactive inventory may bypass the containment building during the event. Because of this safety concern, SGTR event is classified as a design-bases event (DBE). The steam generator tube ruptures may be divided into two categories: spontaneous and induced. The spontaneous tube rupture occurs due to tube degradation mechanisms such as primary water stress corrosion cracking (PWSCC), outside diameter stress corrosion cracking (ODSCC), intergranular attack (IGA), intergranular stress corrosion cracking (IGSCC), pitting, fretting, wear, thinning, denting, corrosion, erosion, fatigue and cavitation. The induced tube rupture occurs as a consequence of other events. The NRC staff claims that it would seem highly improbable that two random SGTR failures would occur simultaneously but damage or tube failure caused by a foreign object could be a more likely initiator of a multiple steam generator tube rupture [1]. In the SGTR event occurred at Ginna nuclear power plant (NPP) in 1982, the utility examined the steam generator tubes after the event and found that although only one S/G tube had ruptured, more than 20 had been severely damaged. The examiner also found loose parts (baffle plate debris) left in the affected steam generator.

According to risk analyses for typical pressurized water reactors (PWRs), the risk of multiple steam generator tube rupture event is known to be larger than that of single SGTR event even though the probability of SGTR occurrence is larger than that of MSGTR. SECY-93-087 reports that the frequency of a single SGTR is approximately 1.1×10^{-2}/RY and frequency for a MSGTR event is estimated to be 8.4×10^{-4}/RY based on a 50% confidence level. The MSGTR event became a safety issue in the early 90's, even though there is not any report of a MSGTR event, because of two safety concerns. The first one is the containment bypass of radioactive inventory, which is the same as the case of singe SGTR event. The other one is the increase of reactivity of reactor core. The latter concern is raised because a boron-free secondary coolant may flow into the primary loop due to a reverse pressure difference. This concern is specific to passive PWRs such as AP600. However, the former concern applies to evolutionary PWR designs as well as passive PWRs. NRC staff suggested that containment bypass of primary coolant following SGTR should be investigated for the System 80+ design [2]. Following NRC's position, ABB-CE performed analyses of MSGTR event in system 80+ and presented results showing that the realistic response of the system 80+ design is to allow more than four hours for operator action following a single tube rupture and to allow more than 30 minutes following rupture of five tubes before MSSVs would be first lifted [3].

The primary purpose of the present study is to understand transient phenomena and consequences of a MSGTR event that is postulated to occur in APR1400. In addition, the length of time from the initiation of the event until the operator must take action to prevent opening of the main steam safety valves (MSSVs) is evaluated. Basically, the analysis of MSGTR event is very similar with the analysis for SGTR in a standard safety analysis report (SSAR) Chap. 15. The principal difference is in the analysis methods. The MSGTR event is analyzed by means of best-estimate (BE) methods while SGTR analysis in the SSAR Chap. 15 is performed using conservative analysis methods. In practice, arbitrary location of a S/G tube is assumed to be ruptured in the analysis of SGTR event which is a DBE. This assumption is justified by applying a conservative friction loss coefficient at the break. The effects of break location in a tube along axial-direction are examined in this work. The effects of simultaneous tube-failures in both S/Gs are examined as well. As stated above, a criterion is used as the time to open the MSSVs. The analyses in the present work were performed by means of BE methods. The safety and non-safety systems and components are assumed to be in operation in automatic mode and no operator action is assumed during transients in this work.

2. APR1400 modeling for MARS1.4

The MARS (Multi-dimensional Analysis of Reactor Safety) code has been being developed by Korea Atomic Energy Research Institute (KAERI) for a multi-dimensional and multi-purpose realistic thermal-hydraulic system analysis of light water reactor transients[4]. MARS version 1.4 had been developed on April, 1999. The backbones of the MARS 1.4 are RELAP5/MOD3 and COBRA-TF codes which constitute the bases of 1-D and 3-D modules of the MARS code, respectively. New features in RELAP5/MOD3.2.2 β–version have been implemented in MARS1.4. The RELAP5 code is a versatile and robust code based on a one-dimensional two-fluid model for two-phase flows. The COBRA-TF code employs a three-dimensional, two-fluid, three-field model. In order to fully exploit the excellent and well-verified features of each code, the two codes have been consolidated into a single code, MARS, in the form of 1D and 3D modules through the integration of hydrodynamic solution scheme and the unification of various thermal-hydraulic models and I/O features. Then, the code has been fully restructured using the modular data structure and a new dynamic memory allocation scheme of standard FORTRAN 90, which greatly enhances the code readability maintaining the code memory requirements. In addition, the Windows graphics features were implemented for user friendliness. MARS 1.4 now runs on Windows platform and it is used as a multi-dimensional thermal-hydraulic analysis tool for light water reactor transients, experiment facilities and various safety research purposes.

The APR1400 is an evolutionary advanced light water reactor (ALWR), which is a two-loop, 3983 MWt, PWR supposed to be completed in 2010. This NPP was started to develop in 1992 by a Korean next generation reactor (KNGR) project and officially renamed as APR1400 in February 2001. The APR1400 design provides a number of systems for use in mitigation of a tube rupture event. The important APR1400 design features to increase the capability to avoid containment bypass during MSGTR event are [5]:

(1) The steam bypass control system (SBCS) is an automatic system which provides a path to remove steam from the S/Gs. In the event of SGTR, the SBCS will automatically relieve secondary pressure and dump steam to the condenser.

(2) Two N-16 monitors, one per steam generator, to assist in the diagnosis of the event.

(3) The main feedwater control system (FWCS) automatically terminates main feedwater following a reactor trip with reduced primary coolant temperatures.

(4) The APR1400 SDVS discharging to the IRWST is actuated by the operator when MSSVs are challenged.
(5) The APR1400 SBCS directs secondary flow from all bypass valves to the condenser eliminating two paths to the atmosphere.
(6) The IRWST in the APR1400 design is both a large source of safety injection water and a quench tank that confines blowdown fluids within the containment.
(7) The large secondary side volume of APR1400 S/Gs provides extra capacity and therefore extended operator action time before the MSSVs are challenged.
(8) The lowered RCS operating coolant temperature decreases the likelihood of a SGTR event.

In order for analyses of MSGTR event, the APR1400 is nodalized as shown in fig. 1. Nuclear steam supply system (NSSS) and several safety systems are modeled. In general, the secondary system of nuclear power plant (NPP) is not modeled in analyses of loss-of-coolant-accident (LOCA). However, the modeling of them is necessary in an analysis of MSGTR events since they show the response of NPP to MSGTR events. A direct-vessel-injection (DVI) system and safety injection tank (SIT) are modeled as well. If primary system pressure is reduced below 15.24 MPa the pressurizer (PZR) backup heater is actuated with a power of 200kW. If the pressure decreases further and reaches a setpoint value, high-pressure safety injection (HPSI) system delivers emergency core cooling water to the reactor core from safety injection tank (SIT). The SIT is designed to automatically start injection when the PZR pressure becomes lower than 4.346 MPa. It can be said that the primary system modeling is the same as that for LOCA analysis. Among secondary systems, turbine, SBCS, MSSV, and main steam isolation valve (MSIV) are modeled since these affect MSSV lift time during the events.

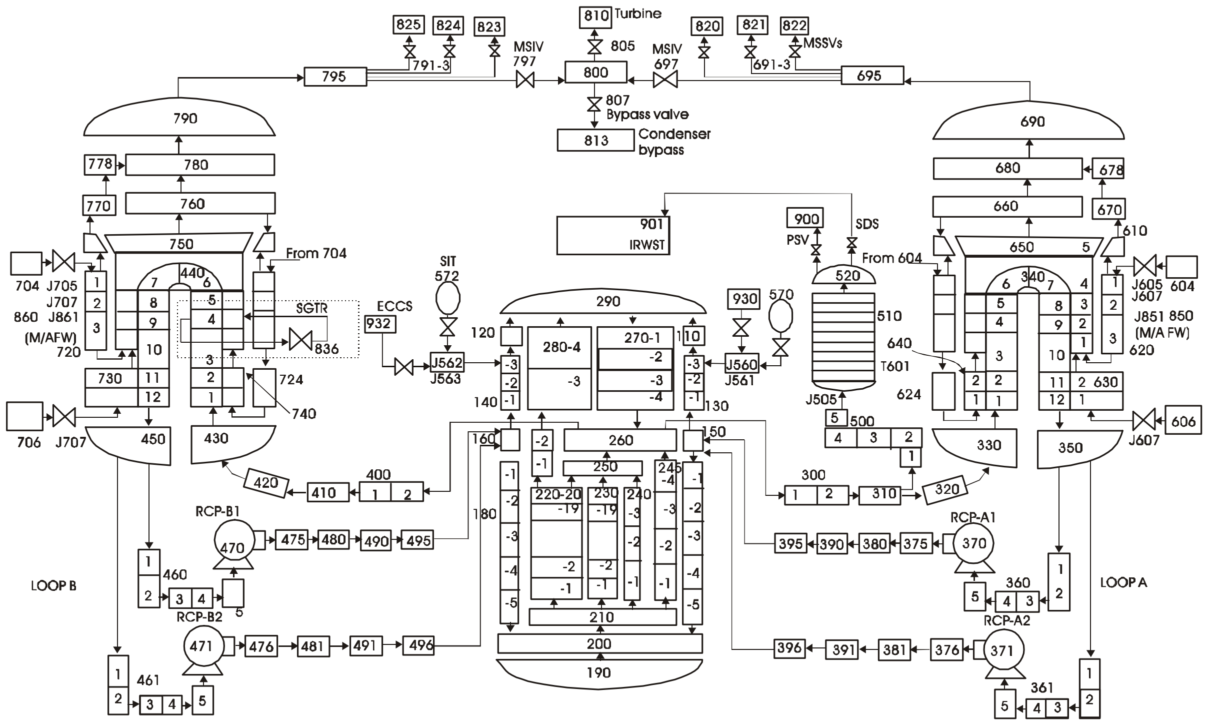

Fig. 1 Schematic diagram of APR1400 nodalization

The APR1400 has two steam generators. The steam generator A represents the one installed in loop A where the pressurizer is connected through a surge line while S/G B represents the one in loop B. Each steam generator has 11,264 tubes whose inner diameter is 0.017094 m. The modeling of the S/G secondary side in general has significant effect in the analysis of SGTR. The S/G secondary side modeling in the present analysis is supposed to cover most of important two-phase flow behaviour. In particular, a recirculation of the S/G secondary side can be treated by node 660 and 610. The area between them is 2.74838 m^2. The forward and reverse loss coefficients from node 660 to 610 are determined as 1.923 and 2.183, respectively. In order for rupture simulation, an imaginary valve (836) is modeled between the tube side and the shell side of a steam generator. The turbine (810) is modeled as a time-dependent volume and connected to a steam header (800) and a turbine stop valve. The turbine stop valve is closed at 5 seconds after reactor trip. The MSIVs have a function of isolation of steam generators from steam header. They are automatically closed at 5 seconds after main steam isolation signal (MSIS) is generated. The MSIS is generated on high level in the affected steam generator whose set point is 95% wide range level. SBCS plays a role of heat sink for the secondary side by bypassing steam until MSIV is closed. The SBCS can bypass up to 55% nominal steam flow in maximum and the system controls the secondary pressure to maintain it at 7.5 Mpa in automatic mode. By the way, the specifications of turbine bypass valves, which are controlled by SBCS, have not determined until this analyses are carried out. By reason of this, the specifications of the valves used in the KSNP (similar to APR1400 but smaller thermal output) are used in the present analyses. Main steam safety valves are installed at each steam generator. These valves protect steam generator from over-pressurization and relieve thermal energy by dumping steam into atmosphere. In general, the MSSVs consists three banks with various lifting set-values. However, all banks of MSSVs are modeled on being lifted at 8.2439 Mpa (1195 psia) in the present analysis. The secondary side feedwater system consists of main feedwater system (MFWS) and auxiliary feedwater system (AFWS). MFWS stops delivery of feedwater on main feed isolation signal (MFIS) in automatic mode. The MFIS is generated at 5 seconds after reactor trip. Each train of AFWS can supply 41.8 kg/s feedwater. The modeling of AFWS is like to be activated at steam generator level of 25% wide range and to be deactivated at 55% wide range.

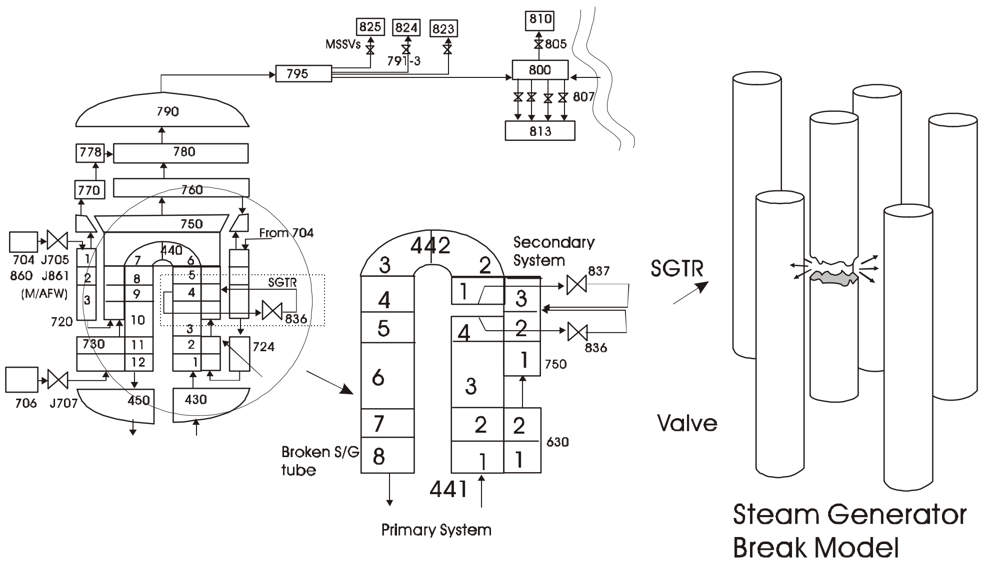

Fig. 2 Tube rupture modeling

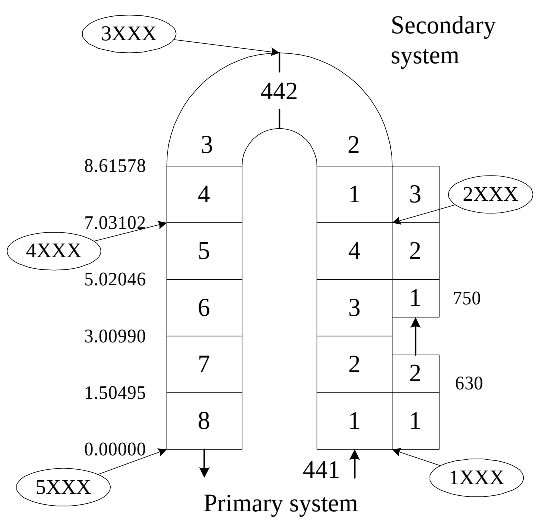

Fig. 3 Tube rupture locations

Figure 2 shows how tube rupture is modeled. A ruptured tube (442) is separately modeled from intact tubes (440). The primary side and the secondary side are modeled as pipe structures and are connected by a heat structure. If a tube is ruptured, primary coolant flows into secondary side. In order to simulate this situation, a valve junction connecting a primary side pipe and a secondary side pipe is introduced. A tube rupture simulation is started by opening the valve junction at a steady state. A multiple rupture is achieved by changing area of valve junction.

In order to evaluate the effect of rupture location, five different tube rupture locations are assumed. The locations are hot-leg side tube sheet, middle of hot-leg side, tube top, middle of cold-leg side, and cold-leg side tube sheet as shown in fig. 3. Figure 3 shows rupture locations and identification for each analysis run. The I.D. of each run consists of 4 digit alphanumeric codes. The first digit represents rupture location. The numbers from 1 to 5 corresponds to hot-leg side tube sheet, middle of hot-leg side, tube top, middle of cold-leg side, and cold-leg side tube sheet, respectively.

3. Procedures and conditions

It is supposed that the APR1400 should have a capability to mitigate a MSGTR event assuming only automatic actuation of components and systems which include both safety grade and non-safety grade equipments. Figure 4 shows an actuation procedure of safety systems, which is set up based on the results of previous single SGTR event analyses.

Normal, full power conditions are assumed in the present analysis except initial reactor power is 102%. The following best-estimate assumptions are made in the present analysis:

(1) Offsite power is available during the transient.
(2) All control systems are available in the automatic mode.
(3) No operator actions are assumed.
(4) Normal plant protection systems (PPS) are assumed to be available and functioning to provide automatic protection during the transient.
(5) Control system actuations during the transient are assumed to be at nominal setpoint values.
(6) The condenser is assumed to have an enough capacity for receiving steam flowing through the turbine bypass valves from the steam generators.

An automatic reactor protection system (RPS) is assumed to be available with relevant reactor trip logics such as VOPT, HPP, LPP, LSGL, HSGL, LSGP, and RCS subcooling trips. Reactor coolant pump (RCP) is automatically shut down on hot-leg saturation signal. After rupture of S/G tube, pressurizer backup heater is actuated due to rapid depressurization of primary side. As the cumulative leakage of primary coolant increases, the RPS trips reactor. Turbine trips right after reactor trip. Systems and components are actuated to regulate secondary side pressure which is increased due to leakage of primary coolant and termination of the main feedwater. Automatic operations of steam dump valves and main steam isolation valves are intended to contribute to robustness of secondary side. If primary coolant leak rate through ruptured tubes exceeds the maximum capacity of steam bypass control system, the secondary side S/G level starts to increase, and finally high-level signal is generated to close MSIV. After the MSIV is closed, the secondary side pressure continues to increase due to both primary coolant leakage and evaporation of the coolant in the shell side. When the secondary pressure exceeds a set value, MSSVs are lifted to relieve the pressure. The calculation of the present study is made beyond this point.

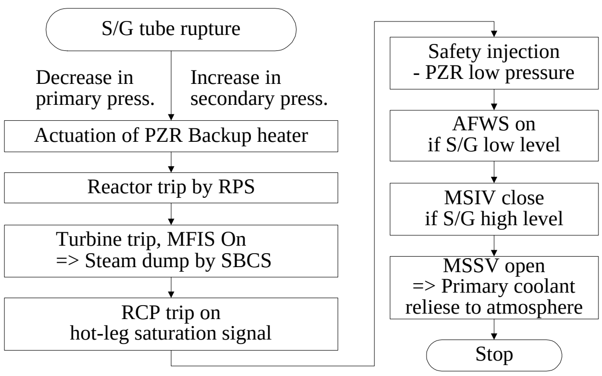

Fig. 4 Procedure of calculations

4. Results and discussions

Transient plots of APR1400 for single tube rupture are shown in figs.5 through 8. Figure 5 shows transients of the primary and the secondary pressures. The pressure transient of steam generator A is quite similar with that of steam generator B. The RCS pressure drops very rapidly following a tube rupture and this leads to a safety injection. Even though pressurizer backup heater is actuated, the primary pressure continues to decrease to reach a safety injection setpoint value. The injection of SI water results in an increase in the RCS pressure as the leak rate of primary coolant is less than safety injection flow rate. The steam generator pressure increases rapidly after a turbine trip to reach and remains at about 7.5Pa, which is the set value for the turbine bypass valve, since it is modulated by turbine bypass system.

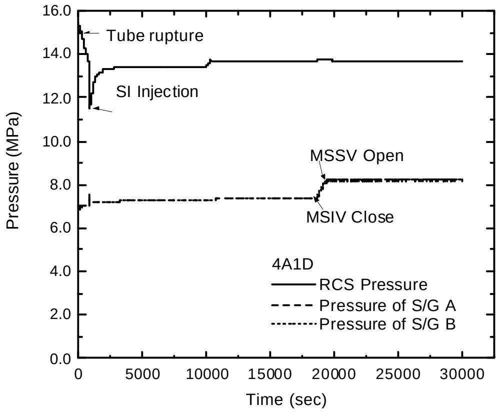

Fig. 5 Pressures vs. time for single tube ruptured

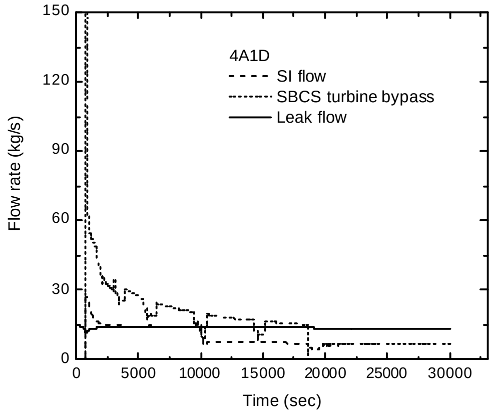

Fig. 6 Flow rates vs. time for single tube ruptured

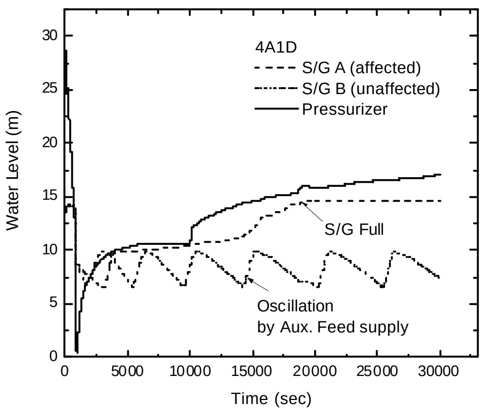

Fig. 7 Levels vs. time for single tube ruptured

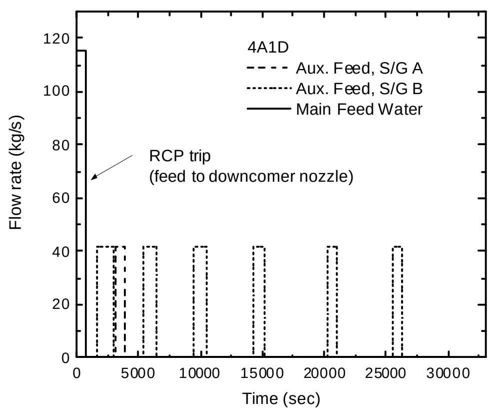

Fig. 8 Feedwater flows vs. time for single tube ruptured

The water level of the affected steam generator decreases rapidly following a reactor trip. It is because the main feedwater supply is terminated after a reactor trip. After this period, steam generator level reach a plateau and later the level starts to increase and MSIV is closed at 18660 seconds. After closure of MSIV, secondary pressure increases rapidly and finally MSSVs are lifted at 19599 seconds. Figure 6 shows flow rates of safety injection, turbine bypass, and leak through the break while fig. 7 shows water levels of affected and intact steam generators and pressurizer. After a period of rapid decline, the pressurizer level recovers due to safety injection. The turbine bypass flow appears to be larger than leak flow rate in its early stages. During this period, steam generator inventory can be under automatic control of turbine bypass system without an actuation of MSIS so that shell side water level of the affected S/G does not increase much. After a trip of main feedwater, unaffected S/G water level oscillates as the auxiliary feedwater is supplied. That is, level of S/G B increases when auxiliary feedwater is supplied and decreases when it is terminated and S/G inventory evaporates. The auxiliary feedwater to S/G A is evaluated to be supplied once after main feedwater trip. Since leak flow continues to exist, auxiliary feedwater system needs not to be operated. As the cold auxiliary feedwater is supplied, however, specific enthalpy of S/G inventory decreases and turbine bypass load also decreases. Figure 6 shows periodic decreases in turbine bypass flow rate looking like sawteeth. These sawteeth coincide with the supply of auxiliary feedwater shown in fig. 8. Since the level of leak flow does not change much while the turbine bypass flow gradually decreases, the leak flow accumulates in the affected S/G shell side. This leads to an increase in S/G level and finally MSIS actuation.

Figures 9 through 12 show the results for five tubes rupture in the APR1400 design. These plots illustrate transients for RCS and S/G pressures, flow rates of leak, safety injection, feedwater and turbine bypass, and levels of PZR and S/Gs. These transients are very similar with those for single tube rupture. The major difference is that the changes of parameter values are more rapid than those for single tube rupture owing to the larger leak flow through the break. The steam generator pressure increases rapidly following the reactor trip and maintains at about turbine bypass opening set value of 7.5 MPa. The steam generator level rapidly decreases after reactor trip since the main feedwater supply is terminated and two-phase mixture level is collapsed due to increase in steam generator pressure. After this rapid decrease in steam generator level, the level of the affected steam generator continues to build up since the turbine bypass flow is much smaller than leak flow through the break. This increase in affected steam generator level leads to a MSIS generation at 2552 seconds. After MSIV is closed, steam generator pressure increases and reach a MSSV lifting set value at 3300 seconds. Since the affected steam generator level continues to increase, there is no chance of auxiliary feedwater system actuation. However, intact steam generator level continues to decrease following the rapid decrease phase since there is no supply of feed water. This level decrease leads to an actuation of auxiliary feedwater system and the intact steam generator level starts to increase.

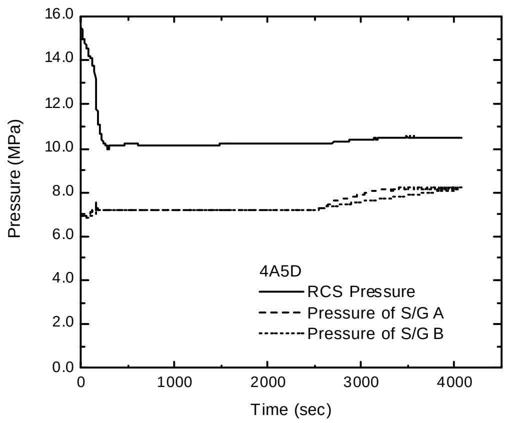

Fig. 9 Pressures vs. time for five tubes ruptured

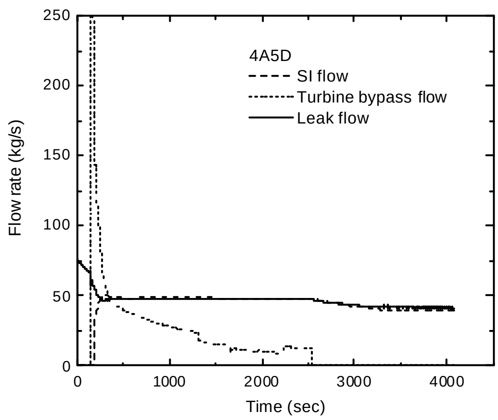

Fig. 10 Flow rates vs. time for five tubes ruptured

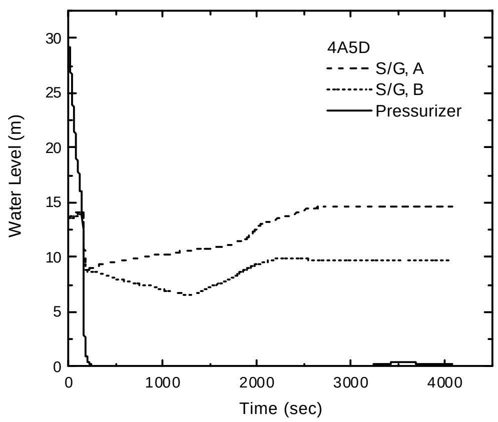

Fig. 11 Levels vs. time for five tubes ruptured

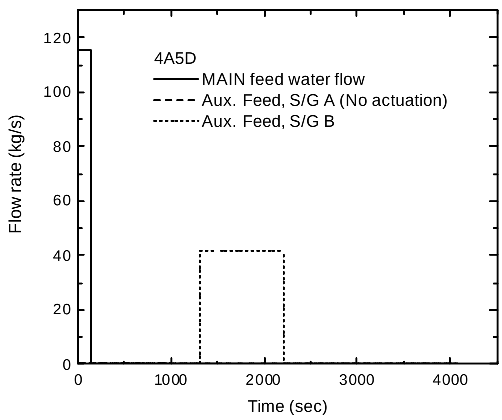

Fig. 12 Feedwater flows vs. time for five tubes ruptured

Figures 13 through 16 show the results for five tubes rupture in each steam generator. They present transients for RCS and S/G pressures, flow rates of leak, safety injection, feedwater and turbine bypass, and levels of PZR and S/Gs. These transients are similar with those for five tubes rupture only in the steam generator A. The major difference is that the parameter values for both steam generators are close owing to the fact that the leak flows in both steam generators are well balanced and this leads to bisymmetric behaviour. Since the water levels of two steam generators continues to increase following a main feedwater trip, neither auxiliary feedwater to S/G A nor B are operated. The total leak flow rate through ruptures is larger than that for five tube ruptures in only steam generator A but each leak flow in steam generator A and B appears to be smaller than that. In consequence, the increases in levels of affected and unaffected steam generator is milder and safety injection flow is larger compared with the case for five tube ruptures only in steam generator A. The levels in both steam generators continue to increase and a MSIS is generated on high-level signal from steam generator B at 3146 seconds. The steam bypass is terminated by MSIV closure and results in increases in pressures of the two steam generators. As this pressure reaches 8.2439 MPa at 4564 seconds, the MSSVs lifts and starts to steam dump into the atmosphere.

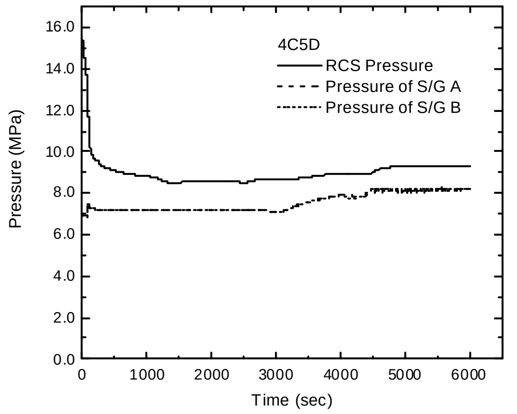

Fig. 13 Pressures vs. time for five tubes ruptured in each S/G A & B

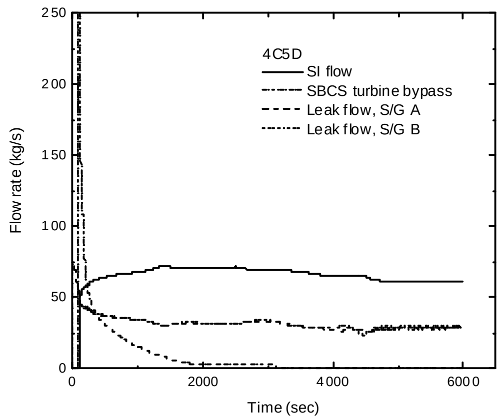

Fig. 14 Flow rates vs. time for five tubes ruptured in each S/G A & B

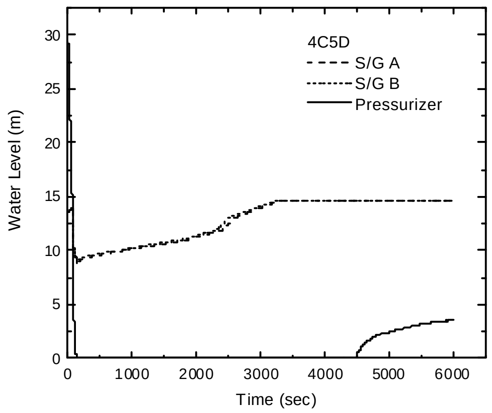

Fig. 15 Levels vs. time for five tubes ruptured in each S/G A & B

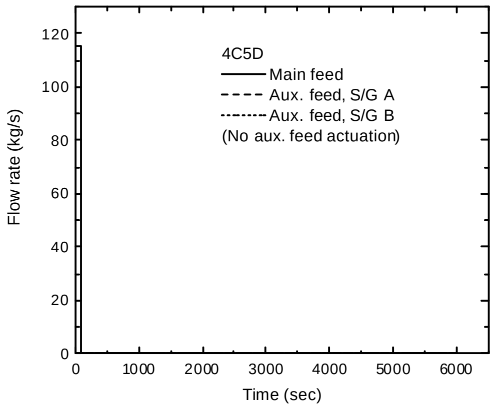

Fig. 16 Feedwater flows vs. time for five tubes ruptured in each S/G A & B

The results of MSGTR event analysis for APR1400 are summarized in table 1. Each event in table 1 is identified by a name consisted of 4 characters except the last case. The first one represents the location of tube rupture as shown in fig. 3. The second one represents affected steam generator. Capital “A” denotes the steam generator A installed in the loop A where the pressurizer is linked through a surge line. Capital “B” denotes the steam generator installed in the loop B. The “C” denotes the case that tube ruptures are occurred in both steam generators. Two cases where both steam generators are affected are examined: 4C5D and 4C23D. The former one denotes the case where 5 tubes are ruptured in each steam generator while the latter one denotes the case where 2 and 3 tubes are ruptured in steam generator A and B, respectively. The third character represents the number of ruptured tubes. And the last one is the same as “D”. Table 1 shows MSIS generation time and MSSV lift time for each event scenario. The MSSV lift time varies in a wide range depending on number of ruptured tubes and rupture location in a tube. It can be seen that the sequence for MSGTR event is the same as the procedure shown in fig. 4.

Table 1 Sequence of events for MSGTR

Run	Leak flow kg/sec	Rx. Trip sec	SI initi. sec	Aux. Feed	sec	MSIS sec	MSSV sec
1A5D	55.6	121	165	B	1525	2066	2765
2A5D	45.9	135	185	B	1321	2570	3429
3A5D	49.9	140	190	B	1308	2640	3442
5A5D	53.7	110	202	B	1578	2194	2889
4A1D	15.2	785	811	1587(B)	3288(A)	18660	19599
4A2D	25.1	360	368	B	1213	7434	8523
4A3D	32.4	238	245	B	1187	4660	5726
4A4D	43.6	182	192	B	1296	3276	4160
4A5D	47.8	141	184	B	1332	2552	3300
4B5D	48.0	146	190	A	1244	2594	3391
4C5D	65.2	70.1	126		-	3146	4564
4C23D	45.9	145	177		-	4454	5704

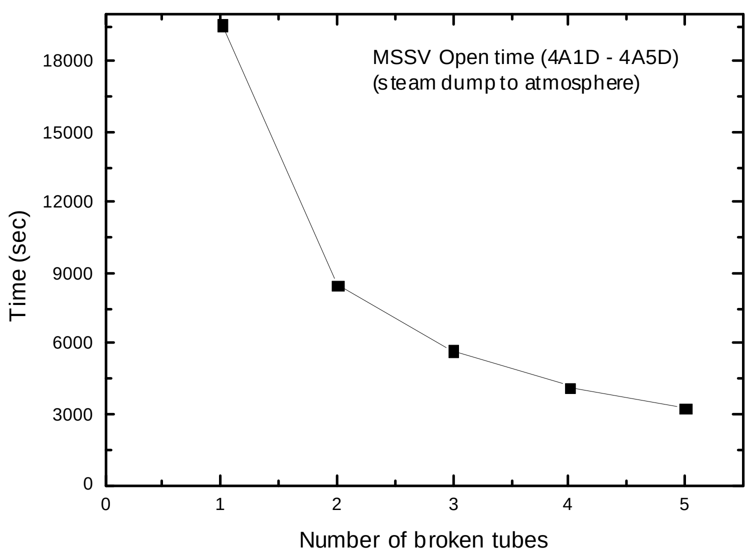

Fig. 17 MSSV lift time vs. no. of tube rupture

The MSSV lift time varies inversely with the number of ruptured tubes as shown in fig.17. The results indicate that the response of the APR1400 is to allow 19600 seconds for MSSV lift following a single tube rupture and to allow 3300 seconds following rupture of five tubes. Figure 17 shows a similar trend with that of KNGR SSAR [5] but values are relatively larger than that. The KNGR SSAR shows that KNGR (APR1400) allows about 1800 seconds for MSSV lift following rupture of five tubes. A fundamental difference between these two calculations is in the leak flow rate through rupture. When five tubes are ruptured, the present analysis of 4A5D expects the leak rate to be 47.8 kg/s while KNGR SSAR expects 78.47 kg/s (173 lbm/s) at 1600 psia. The latter one is about 61% larger than the former one. The differences in the assumptions and modeling methods between the present analysis and KNGR SSAR are thought to cause the discrepancy. The tube modeling of the present analysis is different from KNGR SSAR. A model of single tube with a valve is used in traditional single SGTR analyses. That is, a single tube is modeled such that the tube allows design flow rate and a valve modeled in order for rupture simulation. This modeling method is also used in the KNGR SSAR. However, in the present analysis, a ruptured tube is separately modeled from intact tubes in the present analysis as illustrated in fig. 2. This difference may leads to a discrepancy in upstream flow conditions of broken tubes. As mentioned earlier, specifications of steam bypass valves and MSSVs of the APR1400 in the present input deck may not be correct since they are not fixed and subject to change. In addition, all valves are assumed to open/close instantly with a short delay time. Any stroke time is not considered in the present analysis. Another plausible cause is the selection of discharge coefficient that is applied to the valve junction connecting ruptured tube end to the secondary side. Roth et al. [7] simulated BETHSY test using RELAP5/MOD3 and suggested the discharge coefficient for subcooled water, saturated two-phase flow and superheated steam should be 0.92, 1.25 and 0.97, respectively. Flechter & Schultz [8] also recommended discharge coefficient for RELAP5/MOD3 should be 0.8, 1.2 and 1.0 in the same order. In the first stage of MSGTR analysis in KNGR SSAR, the discharge coefficient (C_D) was adjusted such that a critical flow rate estimated by RELAP5/MOD3 is the same as that by a design code for a single tube rupture. The analysis for MSGTR in KNGR SSAR was carried out with this discharge coefficient fixed [9]. This procedure looks like that the critical flow model of a design code, which is a conservative EM model, was used in developing Appendix 5F of KNGR SSAR. In the present analysis, however, discharge coefficients are set to be 1.0 because there have been no reference experiments that can be compared to multiple steam generator tube ruptures. There may be other various causes but have not examined in this study. In general, the trend of fig. 17 is similar to that of KNGR SSAR and the response of plants to a tube rupture event is reasonable. In this regards, it is believed that the present results of analysis are good enough to be used in sensitivity study while absolute values in terms of time may have errors.

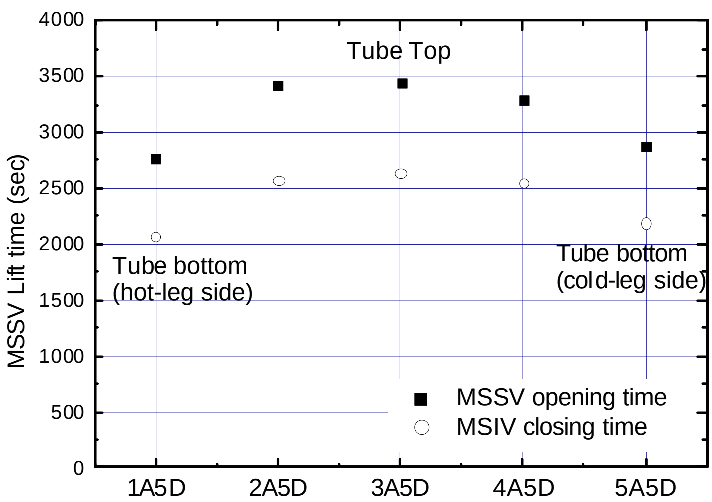

Fig. 18 MSSV lift time vs. rupture location

Figure 18 shows a comparison of MSIV closing time and MSSV opening time among five runs whose rupture location varies as shown in fig. 3. For this comparison, five runs are carried out for five tube ruptures in steam generator “A”. The only difference among them is the rupture location. The MSSV lift time is found to be the shortest when tubes are ruptured in the vicinity of hot-leg side tube sheet while longest when tube top is ruptured. The MSSV lift time for tube-top rupture is 24.5% larger than that for rupture at hot-leg side tube sheet. If tubes are ruptured at tube sheet, primary coolant in hot-leg can flow out to the secondary side with less friction loss since small diameter flow path (tube) is shortened. This effect leads to a larger leak rate, 55.6 kg/s compared with 49.9 kg/s. If leak rate is large, steam generator secondary side level increases faster. The MSIS is generated earlier and finally the MSSVs are lifted earlier. In the meantime, the MSSV lift time for the ruptures at hot-leg side tube sheet appears to be small compared with that at cold-leg side tube sheet. It is because the enthalpy of hot-leg coolant is higher than that of cold-leg coolant. Considering these comparisons, it can be said that tube rupture location considerably affects the sequences and consequences of MSGTR event.

Figure 19 shows how selection of steam generator damaged affects MSSV lift time. Four cases are compared: S/G A with five tube ruptures, S/G B with five tube ruptures, S/G A and B with five tube ruptures each, S/G A and B with two and three tube ruptures, respectively. This plot suggests that multiple steam generator tube rupture only in S/G A give most conservative results in terms of MSSV lift time. The cause of the discrepancy between steam generator A and B is thought to be the difference of connected systems to each loop. A pressurizer is only attached to the loop linking S/G A. As the pressurizer is closer to S/G A than B, the total friction loss from the pressurizer to broken tubes of S/G A is smaller than that of S/G B. Therefore, the coolant coming out of pressurizer flows to broken tubes of S/G A with less flow resistance. This situation leads to a larger leak rate and earlier MSSV lift time even though the difference is small. As can be seen in table 1, leak flow rate for five tube ruptures in steam generator A (4A5D) is 47.8 kg/sec while that for steam generator B (4B5D) is 48.0 kg/sec.

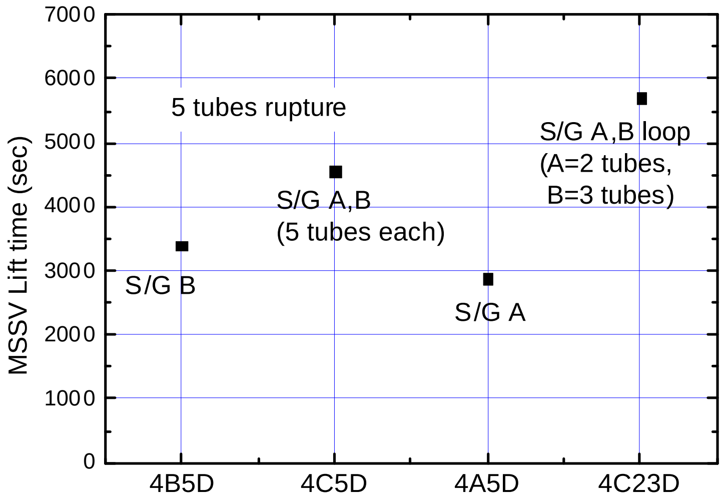

Fig. 19 MSSV lift time vs. affected S/G

The MSSV lift time for the cases that both steam generators are affected (4C5D, 4C23D) are appeared to be larger than that for the single steam generator cases (4A5D, 4B5D). This finding is also valid for the 4C5D case in which five tubes for each steam generator, total 10 tubes are ruptured. That is, if both steam generators are affected, operators are allowed more time to respond even though total number of ruptured tubes is doubled. The cause of this interesting result can be found in a bifurcation of primary leak flow. Leak rate of 47.8 kg/sec is expected when five tubes rupture only in steam generator A (4A5D), while 65.2 kg/sec when five tubes rupture in each steam generator (4C5D). If judged in terms of total leak rate, 4C5D case makes larger leak rate than 4A5D case. However, around a half of the total leak rate of 4C5D is expected in each steam generator since both of two steam generators are affected in the same number of tubes. In the present analysis, even though it is not equivalently split, about a half of it, 33 kg/sec, leaks into the secondary side of each steam generator. This leak rate is smaller than that evaluated in 4A5D case. This situation can be confirmed by comparing fig. 10 and 14. A smaller leak rate makes slower increase in steam generator level and leads to a delay of MSIS generation, and finally results in a delayed MSSV lift time.

5. Concluding remarks

Analysis of postulated multiple steam generator tube rupture events in APR1400 nuclear power plant has been carried out. This event has never occurred in the history of commercial nuclear reactor operation but single steam generator tube failure event is reported to occur every two years. The experience of single SGTR analysis, which is a design basis event, provides bases for transient scenario development. The analysis is performed using a best-estimate system analysis code, MARS1.4.

The results show that MSSV lift time varies in a wide range depending on number of ruptured tubes, rupture location in a tube, and which steam generator is affected. The MSSV lift time varies inversely with the number of ruptured tubes. This trend is similar to that of KNGR SSAR but values of the present calculation are relatively larger than that. A fundamental difference between them is in the leak rate. When five tubes are ruptured, the present analysis expects 61% larger leak rate than that of KNGR SSAR. This discrepancy may be resulted from various causes such as simplified modeling of several systems and components, modeling method of rupture but have not examined in this study. A sensitivity study on this discrepancy would be necessary. A comparison is made for five rupture locations in a tube. The MSSV lift time for tube rupture at the top is evaluated to be 24.5% larger than that for tube rupture in the vicinity of hot-leg side tube sheet. In order to examine how selection of steam generator damaged affects MSSV lift time, four cases are analyzed. The results show that the cases that both of two steam generators are affected allow longer time for operator action compared with the case that a single steam generator is affected.

In this regards, the followings should be considered when a safety analysis concerning a MSGTR event is made:

(1) Largest leak rate through rupture is made when tube rupture is assumed in the vicinity of hot-leg side tube sheet.
(2) If up to five ruptures are assumed, cases that single steam generator is affected gives more conservative results in terms of MSSV lift time compared with the cases that both of two steam generators are affected.
(3) Tube ruptures in a steam generator whose loop has a pressurizer give more conservative results in terms of MSSV lift time than ruptures in the other steam generator.

Acknowledgments

The authors wish to acknowledge the financial support of the Korea Institute of Nuclear Safety made in the program year of 2000.

References

1. SECY-93-087, “ Policy, technical, and licensing issues pertaining to evolutionary and advanced light-water reactor (ALWR) designs,” U.S. Nuclear Regulatory Commission, 1993.
2. D.M., Crutchfield, “CE system 80+ protection against containment bypass during a steam generator tube rupture (SGTR), letter to R.A., Matzie, August 12, 1993.
3. “The system 80+ standard plant design control document, Appendix 5F,” 1997.
4. W.J. Lee et. al, "Improved Feature of MARS 1.4 and Verification", KAERI/TR-1386-99, '99. 8
5. KNGR SSAR, Rev.0.0, Korea Electric Power Corporation, 1999.
6. CE-SSAR, Amendment X, ABB-CE, 1996.
7. Roth, P.A., Choi, C.J., Schultz, R.R., Analysis of two small break loss-of-coolant experiments in the BETHY facility using RELAP5/MOD3, EGG-NE-10353, Idaho National Engineering Lab. 1992.
8. Fletcher, C.D., Schultz, R.R., RELAP5/MOD3 code manual volume V: User’s Guidelines, NUREG/CR-5535, EGG-2596, Idaho National Engineering Lab., 1991.
9. Private communication with a concerning staff at KOPEC, 2000.