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Best-Estimate Analysis of MSGTR Event in APR1400 Aiming to Examine the Effect of Affected Steam Generator Selection

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Abstract

Abundant information about analyses of single steam generator tube rupture (SGTR) events is available because of its importance in terms of safety. However, there are few literatures available on analyses of multiple steam generator tube rupture (MSGTR) events. In addition, knowledge of transients and consequences following a MSGTR event are very limited as there has been no occurrence of MSGTR event in the commercial operation of nuclear reactors.

In this study, a postulated MSGTR event in an APR1400 is analyzed using thermal-hydraulic system code, MARS1.4. The present study aims to examine the effects of affected steam generator selection. The main steam safety valve (MSSV) lift time for four cases are compared in order to examine how long operator response time is allowed depending on which steam generator (S/G) is affected. The comparison shows that the cases where two steam generators are simultaneously affected allow longer time for operator action compared with the cases where a single steam generator is affected. Furthermore, the tube ruptures in the steam generator where a pressurizer is connected leads to the shortest operator response time.

Key Words : steam generator, MSGTR, tube rupture location, MSSV lift time, safety analysis

1. Introduction

Each steam generator (S/G) used in nuclear power plants (NPP) consists of more than 5,000 heat transfer tubes. Even though protective actions such as all volatile treatment (AVT) are applied, S/G tubes are challenged by many degradation mechanisms. 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. 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 [1].

The MSGTR event became a safety issue in the early 90's because of two safety concerns, even though there is no report of a MSGTR event. The first one is the containment bypass of radioactive inventory. The other one is the increase in 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. 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 allows more than four hours for operator action following a single tube rupture and more than 30 minutes following rupture of five tubes before MSSVs would be first lifted [3].

Because of safety concern, a SGTR event is classified as a design-bases event (DBE) and its analysis should be presented in a standard safety analysis report (SSAR). An analysis of MSGTR event is very similar to the analysis of SGTR. The principal difference is in the analysis methods. The SGTR analysis in the SSAR Chap. 15 should be performed using conservative analysis (EM) methods. However, the NRC suggested the MSGTR event which is a beyond design basis events (BDBE) be analyzed by means of best-estimate (BE) methods [1]. BE analysis of an event may produce a realistic figure of transients so that it can be a help to design preventative and mitigative features. However, literature as well as experiences of MSGTR event analysis is rare since there has been no occurrence of MSGTR event in the commercial operation of nuclear reactors and regulatory authorities got concerned with this event recently. In order for building up knowledge of MSGTR event, a best-estimate analysis of this event has been performed. An early stage work aiming to understand the effects of rupture location in a tube along axial-direction has been completed and reported [4]. Considering that APR1400 has two steam generators, it is of concern that which steam generator, when it is affected, gives more severe consequences. This concern is examined in this work. The effect of simultaneous tube-failures in both S/Gs is examined as well. 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. The analyses in the present work were performed using BE thermal-hydraulic system code, MARS 1.4 [5]. 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

Several cases of MSGTR event are to be analyzed in this work using BE thermal-hydraulic system code, MARS 1.4. MARS version 1.4 has been developed on April, 1999 and being tested with benchmark problems such as BETHSY experiments as well as UPTF results in order to show its realistic evaluation capability [6]. Lee et al. made use of the MARS code to analyze a

SGTR event in YGN 1&2 and claimed that it realistically evaluated the sequence of SGTR event [7]. Since the characteristics of SGTR and MSGTR events are quite similar to each other, the MARS code is speculated to be applicable to the analysis of MSGTR events.

The backbones of the MARS 1.4 are RELAP5/MOD3 [8] and COBRA-TF [9] codes which constitute the bases of 1-D and 3-D modules of the MARS code, respectively. New features in RELAP5/MOD3.2.2 have been implemented in MARS1.4. In the present work, 1D module only is used.

The APR1400 is an evolutionary advanced light water reactor (ALWR), which is a two-loop, 3983 MWt, PWR scheduled to be constructed in 2010. The APR1400 design provides a number of systems for use in the mitigation of a tube rupture event. The important APR1400 design features to cope with containment bypass during MSGTR event are steam bypass control system (SBCS), two N-16 monitors, main feedwater control system (FWCS), safety depressurization and ventilation system (SDVS), elimination of paths from the secondary side to the atmosphere, large in-containment refueling water storage tank (IRWST), large S/G secondary side volume and lowered RCS operating coolant temperature. In order for the analyses of MSGTR event, the APR1400 is nodalized as shown in fig. 1. Nuclear steam supply system (NSSS) and several safety systems are modeled. The secondary system modeling is of importance in an analysis of MSGTR events since they affect 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 set value, emergency core cooling water is delivered to the reactor core from safety injection tank (SIT)

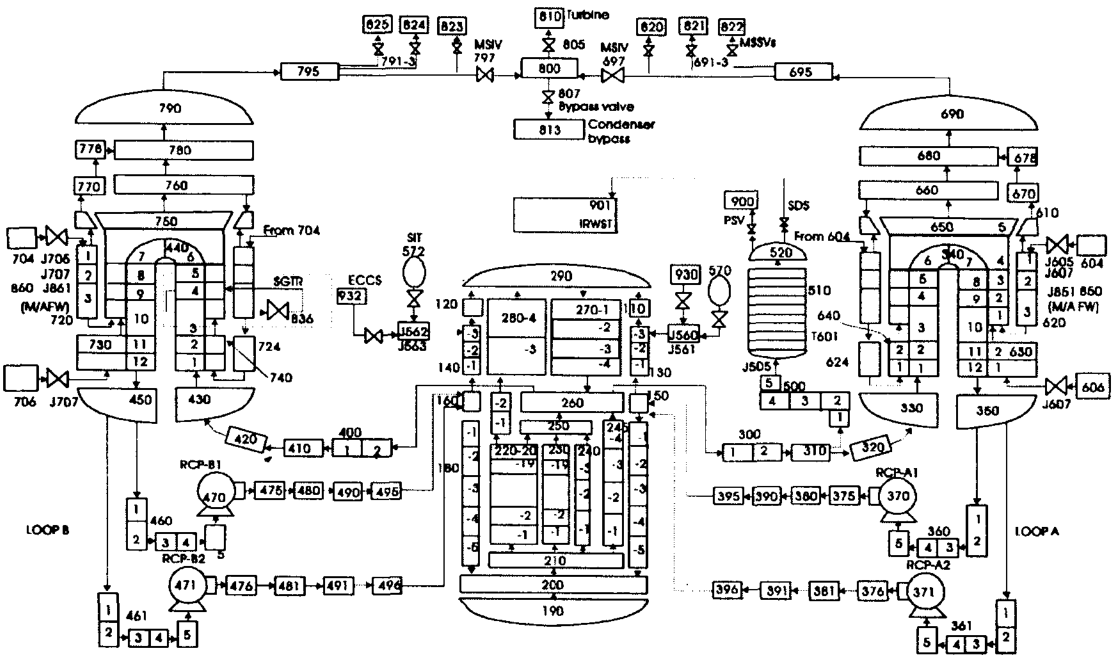

Fig. 1. Schematic Diagram of APR1400 Nodalization

via a DVI nozzle. The SIT is designed to automatically discharge their contents of borated water into the RCS 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.

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 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 introduced 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 to isolate steam generators from steam header. They are automatically closed within 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, were not available when this analysis is 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

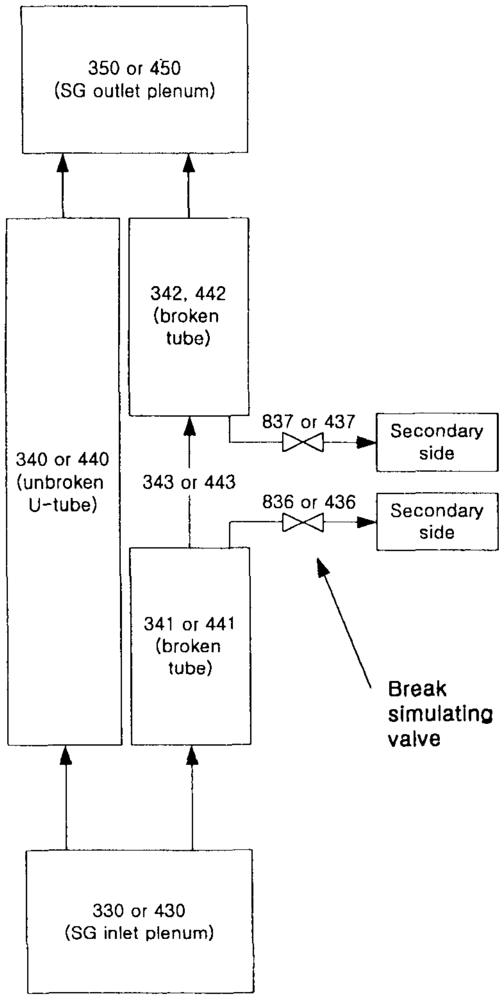

Fig. 2 Tube Rupture Modeling

set-values. However, all banks of MSSVs are modeled 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 AFWS is modeled to be activated at steam generator level below 25% of wide range and deactivated above 55% of wide range.

Figure 2 shows how tube rupture is modeled. A ruptured tube (442) is separately modeled from a bundle of intact tubes (440) in a faulted steam generator. This tube rupture modeling method is different from conventional modeling method used in SGTR analysis in which a single tube is modeled. The primary side and the secondary side are modeled as pipe components 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. Tube rupture simulations are started by opening the valve junction at a steady state. Multiple tube ruptures are achieved by changing flow areas of the valve junction and the broken tube in accordance with the number of ruptured tubes.

3. Procedures and Conditions

It is assumed that a MSGTR event of the APR1400 is mitigated only by the automatic actuation of components and systems which include both safety grade and non-safety grade equipments. Figure 3 shows an actuation procedure of safety systems, which is set up based on the design features of APR1400.

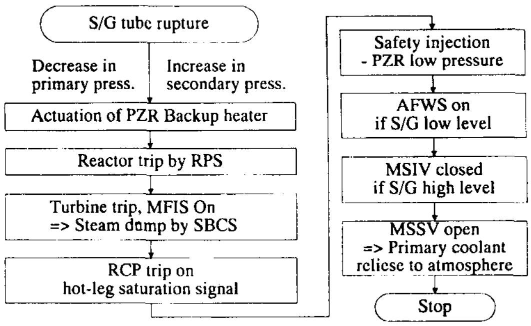

Fig. 3. Procedure of Calculations

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 automatic mode.
(3) No operator actions are assumed.
(4) Plant protection systems (PPS) are actuated by their nominal setpoints and perform their intended functions.
(5) Control system actuations during the transient are actuated by their nominal setpoint values and perform their intended functions.
(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 variable overpower trip (VOPT), high pressure protection (HPP), low pressure protection (LPP), low steam generator level (LSGL), high steam generator level (HSGL), low steam generator pressure (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 the reactor trip. Systems and components to regulate secondary side pressure are actuated by the 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 isolation, the secondary side pressure increase is accelerated 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 right beyond this point.

4. Results and Discussions

A series of MSGTR events has been analyzed. BE analysis results for a single S/G tube rupture and the effect of rupture location in a tube along axial-direction on the consequences of the event are well described in the previous paper [4]. The authors claimed that the MSSV lift time was found to be the shortest when tubes were ruptured in the vicinity of hot-leg side tube sheet while longest when tube top was ruptured. They evaluated that the MSSV lift time for tube-top rupture was 24.5% longer than that for rupture at hot-leg side tube sheet. In the present paper, the effect of affected steam generator selection on the consequences following a MSGTR event is discussed.

The results of MSGTR event analysis for APR1400 are summarized in table 1. This table shows leak flow rate at 1600 psia, reactor trip time, SI initiation time, auxiliary feedwater actuation time, MSIS generation time and MSSV lift time for each run. Each run in table 1 is identified by a name consisted of 4 characters except the last case. The first one represents the tube rupture location along axial-direction. All runs in the present paper assume the same rupture location as the middle of hot-leg side half. The second one represents affected steam generator. Capital "A" denotes the steam generator A installed in the loop A where the pressurizer is

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
4A1D	15.2	785	811	B(1587), A(3288)	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	170.1	126	-	3146	4564
4C23D	45.9	145	177	-	4454	5704

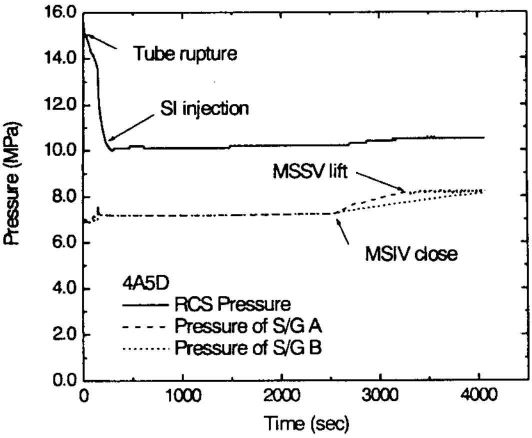

Fig. 4 Pressures vs. Time for Five Tubes Ruptured

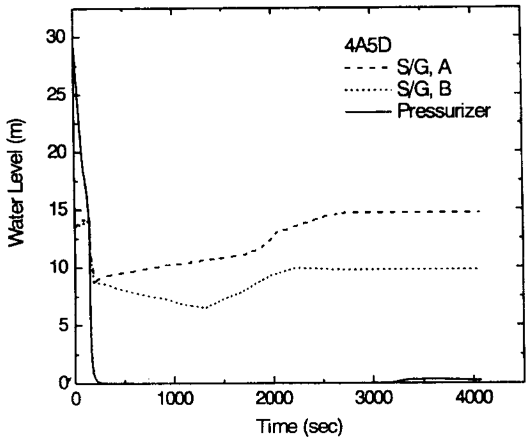

Fig. 6 Levels vs. Time for Five Tubes Ruptured

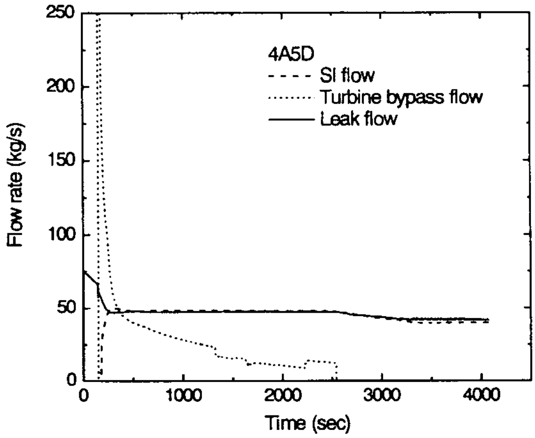

Fig. 5. Flow Rates vs. Time for Five Tubes Ruptured

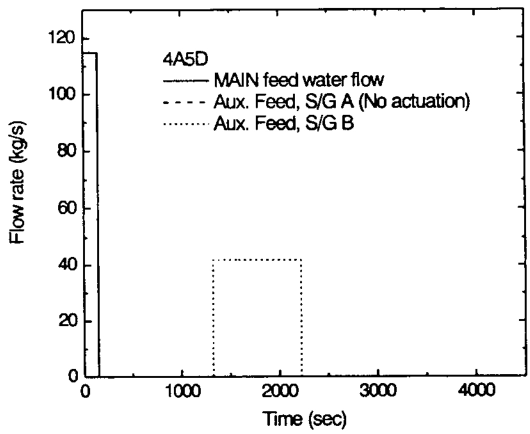

Fig. 7. Feedwater Flows vs. Time for Five Tubes Ruptured

connected. 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 five tubes are ruptured in each steam generator while the latter one denotes the case where two and three 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".

Transient plots of APR1400 for five tubes rupture in S/G A are shown in Fig. 4 through 7. These plots illustrate transients for the RCS and S/G pressures, the flow rates of leak, safety injection, feedwater and turbine bypass, and the levels of PZR and S/Gs. These transients are very similar to those for single tube rupture but the rate of changes of parameter values are more rapid than those for single tube rupture owing to the larger leak flow through the break. 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. Following an initiation of SI water injection, the RCS pressure stops further decrease as the safety injection flow rate is slightly larger than the primary coolant leak rate. The steam generator pressure increases rapidly following the reactor trip and maintains at about turbine bypass opening set value of 7.5 MPa. Both steam generator levels rapidly decrease after reactor trip since the main feedwater supply is terminated and two-phase mixture level is collapsed due to the increase in steam generator pressure. After this rapid decrease in steam generator level, the level of the affected steam generator (S/G A) continues to rise since the break flow into the secondary side is larger than the turbine bypass flow. 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 (S/G B) 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 at 1332 seconds and the intact steam generator level starts to increase. The level of S/G B stops increasing at around 2200 seconds in accordance with a termination of auxiliary feedwater.

Figures 8 through 11 show the results for five tubes rupture in each steam generator. Ten tubes in total are assumed to break in this run. They present transients for the RCS and S/G pressures, the flow rates of leak, safety injection, feedwater and turbine bypass, and the levels of PZR and S/Gs. These transient plots are similar to 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 only in S/G A but each leak flow in steam generator A and B appears to be smaller than that. In consequence, the increases in levels of the two affected steam generators are milder and safety injection flow is larger compared with the case for five tube ruptures only in S/G 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 are lifted and steam starts to be dumped into the atmosphere.

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 the number of ruptured tubes and selection of affected steam generator. It can be seen that the sequence for MSGTR event is the same as the procedure shown in fig. 3. The MSSV lift time varies inversely with the number of ruptured tubes as shown in fig. 12. 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 12 shows a similar trend to that of KNGR SSAR [10] but values are relatively larger than that. The KNGR SSAR

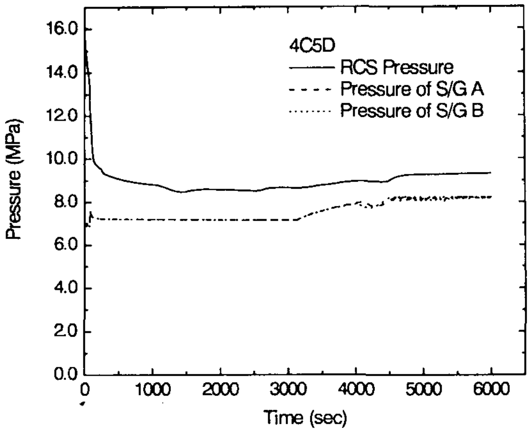

Fig. 8. Pressures vs. Time for Five Tubes Ruptured in Each S/G A & B

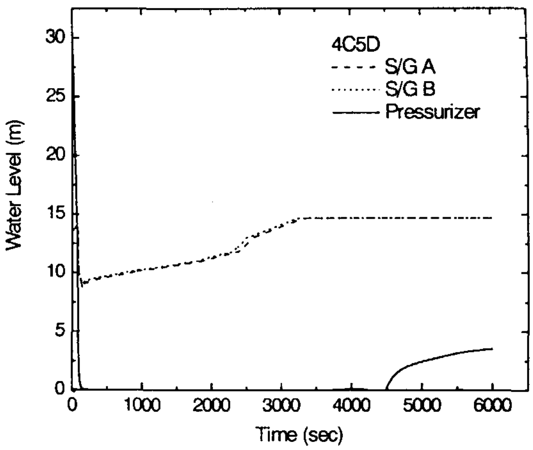

Fig. 10. Levels vs. Time for Five Tubes Ruptured in Each S/G A & B

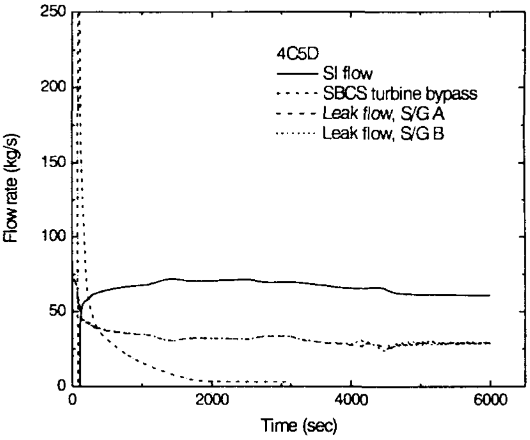

Fig. 9. Flow Rates vs. Time for Five Tubes Ruptured in Each S/G A & B

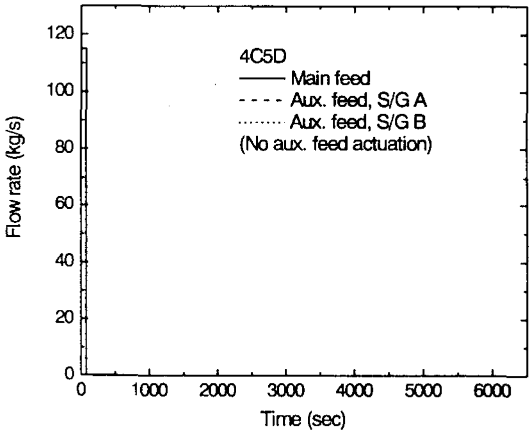

Fig. 11. Feedwater Flows vs. Time for Five Tubes Ruptured in Each S/G A & B

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 predicts the leak rate to be 47.8 kg/s while KNGR SSAR predicts 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 this discrepancy. The tube modeling of the present analysis is different from KNGR SSAR. A model of single tube with a valve is used in conventional single SGTR analyses. That is, a single tube is modeled such that the tube allows design flow rate and a valve is modeled in order for rupture simulation. This modeling method was also used in the KNGR SSAR. However, 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 can be different from those used in the KNGR SSAR since the NPP is still under development. 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. [11] 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. RELAP5 development team [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 (CD) was adjusted such that a critical flow rate estimated by RELAP5/MOD3 is the same as that by Combustion Engineering's design code, CESEC-III [12], for a single tube rupture. The analysis for MSGTR in KNGR SSAR was carried out with this discharge coefficient fixed [13]. This procedure looks like that the critical flow model of the conservative SESEC code was used in developing Appendix 5F of KNGR SSAR. In the present analysis, however, discharge coefficient is 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 been examined in this study. In general, the trend of fig. 12 is similar to that of KNGR SSAR and the

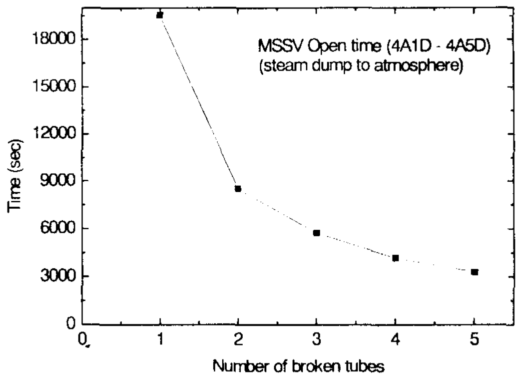

Fig. 12. MSSV Lift Time vs. no. of Tube Rupture

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.

Figure 13 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 gives most conservative results in terms of MSSV lift time. The leak flow rate for five tube ruptures in steam generator A (4A5D) is quite close to that for steam generator B (4B5D). In consequence, the MSSV lift time following a rupture of five tubes in S/G B appears to be very close to that in S/G A.

The MSSV lift times for the cases where 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 ten 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 case is predicted 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. 5 and 9. 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.

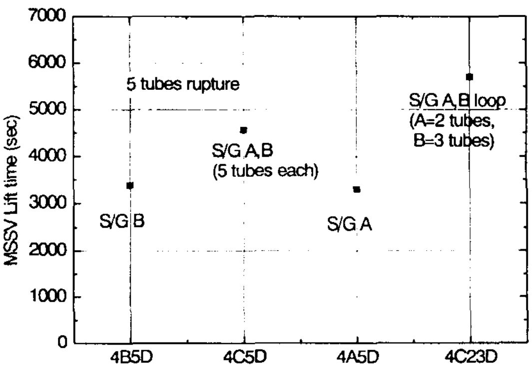

Fig. 13. MSSV Lift Time vs. Affected S/G

5. Concluding Remarks

Analysis of postulated multiple steam generator tube rupture events in the APR1400 nuclear power plant has been carried out. This event has never occurred in the commercial operation of nuclear reactors even though a 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 the number of ruptured tubes 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 longer than that. A fundamental difference between them is in the leak rate. When five tubes are ruptured, the present analysis predicts 61% of leak rate predicted in KNGR SSAR. This discrepancy may be resulted from various causes such as modeling method of rupture and discharge coefficient but have not been fully examined in this study. A sensitivity study on this discrepancy would be necessary. In order to examine how the selection of faulted steam generator affects MSSV lift time, four cases are analyzed. The results show that the cases where two steam generators are simultaneously affected allow longer time for operator action compared with the cases that a single steam generator is affected. It can be said that the cases where single steam generator is affected gives more conservative results in terms of MSSV lift time compared with the cases where two steam generators are simultaneously affected.

Acknowledgments

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