

Thermal-Hydraulic Analysis of Single and Multiple Steam Generator Tube Ruptures in a Typical 3-Loop PWR

Zbigniew Koszela, Łukasz Sokołowski*

Inspecta Nuclear AB, Lindhagensterrassen 1, 112 18 Stockholm, Sweden

Abstract

The response of the full-scale three-loop Pressurized Water Reactor (PWR) RELAP5 computational model on Steam Generator Break Rupture (SGBR) was investigated in this paper. This model was analyzed in terms of its applicability and performance regarding the research task conducted by Warsaw University of Technology and the National Center for Research and Development in Warsaw, Poland. In the paper break sizes corresponding to one, three and six ruptured tubes (which conform to a Loss-of-Coolant event break size area of 0.02%, 0.054 and 0.11%) were studied at three different locations (at the top of the hot-leg side tubesheet, U-bend and at the top of the cold-leg side tubesheet). The reactor at issue was a three-loop PWR of Westinghouse design with thermal output of 2775 MWt.

Keywords: nuclear safety analysis, LOCA, PWR

1. Introduction

A steam generator tube rupture (SGTR) may seriously impact the reliability and safety of a Nuclear Power Plant (NPP). The SGTR event causes a direct flow of primary coolant from the high-pressure reactor coolant system (RCS) to the steam generator (SG) secondary system. This leads to contamination of the secondary system and possible release of radiological products into the environment. The loss of primary coolant can significantly exceed the make-up capacity of the charging pumps.

For these reasons, SGTR is an important safety concern and has been classified as a design-basis event (DBE) for pressurized water reactors (PWRs) [1]. Since the Point Beach incident in 1975, at least 14 incidents of SGTR have been reported worldwide [2]. In most cases the consequences of SGTR incidents are strongly dependent on the number of tubes ruptured. The probability of rupture of several steam generator tubes has been estimated as

extremely low [1], and all reported SGTR accidents involved a single ruptured tube. Reactor shutdown was accomplished without extensive loss of primary coolant or significant release of radiological products [3]. Multiple SGTR may lead to severe accident conditions if the operator's response in respect of mitigating the break flow is inadequate.

SGTRs are usually divided into two categories: spontaneous and induced. Spontaneous tube ruptures are caused by tube degradation mechanisms, including primary water stress corrosion cracking (PWSCC), outer diameter stress corrosion cracking (ODSCC), fretting, pitting, tube sheet and tube support plate denting, wastage, tube fouling and upper tube support plate high-cycle fatigue [1]. Induced SGTRs are a consequence of other events, such as incorrect installation of anti-vibration devices, abnormal secondary water chemistry conditions or loose objects left inside the SGs during earlier maintenance works [1].

Several studies and experiments have been performed to investigate SGTR events in order to verify the features and capacity of the reactor safety systems, as well as to optimize the effectiveness of the emergency operating procedures. Calculations regarding SGTR were conducted by means of a variety of best estimate thermal-hydraulics

*Corresponding author

Email addresses: zbigniew.koszela@inspecta.com
(Zbigniew Koszela), lukasz.sokolowski@inspecta.com (Łukasz Sokołowski*)

system codes as RELAP5, TRACE and MARS1.4 [3–5]. The computational analyses may be divided into two categories: (i) modeling and validation based on experimental data from scale-down facilities of PWRs [6–9] and (ii) computations regarding full-scale reactors [3, 4]. The first category is necessary with regard to testing the accuracy of related thermal-hydraulic and fission product transport models implemented in best estimate thermal-hydraulics system codes [10]. Analyses of full-scale models allow all relevant reactor systems responses to be examined and are thus employed in DBE analyses.

Some of the findings from SGTR computations were similar across most of the studies. The magnitude of the break mass flow rate is dependent on the number of ruptured SG tubes (or area of the break section), assumed rupture location, break modeling approach, primary coolant subcooling and pressure losses along the U-tubes. The primary and secondary sides of SG are modeled as pipe components connected by a heat structure. Usually, all SG tubes are modeled as a common pipe component, or ruptured tubes are modeled separately from the intact tubes [3].

The consequences of SGTR accidents also depend on the efficiency of operator actions. The reactor control and protection systems are designed mainly to preserve core integrity and do not include automatic actions to mitigate radiological release to the environment. The management of SGTRs must, therefore, include operator action aimed at identifying and isolating the ruptured SG, and then terminating the SI (Safety Injection) flow to stop the primary-to-secondary break flow. For thermal-hydraulic safety analyses, however, the initial phases of SGTR events are often simulated without operator action, e.g. [3].

In the present study, the responses of a typical three-loop PWR, to single and multiple SGTR events with a fully available emergency core cooling system (ECCS), are analyzed using the RELAP5Mod3.3 thermal-hydraulic system code. The analyzed event sequences are based on real SGTR incidents evidenced in [1, 2]. The reactor safety and non-safety systems are assumed to work in automatic mode and perform their intended functions. No operator action is assumed during the calculated time periods of the accidents. Effects in terms of the number of broken tubes and different tube rupture locations are investigated. The break sizes correspond to guillotine rupture of one, three and six SG tubes. Three different break positions along the axial direction of the U-tubes were chosen: the top of the hot-leg side tubesheet, the U-bend region and the top of the cold-leg side tubesheet. These

break locations correspond to locations inside recirculating steam generators where tube wall degradations most frequently occur [1]. The results obtained are compared qualitatively with available data and results of other similar calculations published in [3, 4] and [5] among other sources.

2. Computational model

The presented model applies to a typical three-loop PWR of Westinghouse design with a rated thermal power of 2,775 MW_t and a total vessel mass flow rate of 13,800 kg/s. The reactor core is composed of 167 fuel assemblies of the 17×17 lattice design.

The RELAP5 nodalization, shown in Fig. 1, consists of 270 volumes connected with 284 junctions and 353 heat structures. The modeled steam generator is Westinghouse type F design with 5626 stainless steel U-tubes and a total heat transfer area of 4645 m².

The reactor pressure vessel (RPV) model, volumes 100 through 270, consists of the vessel cylindrical shell with hemispherical bottom and upper heads, downcomer, lower plenum, core region and core bypass, upper plenum and guide tubes.

The core region is modeled as a single flow channel, by means of pipe component 170 with 14 axial nodes. The point kinetic model is used. The moderator density and Doppler reactivity coefficients are calculated with the separable model using core averaged, relative axial power distribution for axial weighting.

The plant has three independent primary loops (volumes 300 to 390, 400 to 490, and 500 to 590). Each loop consists of the hot leg (HL), the steam generator (SG) with U-tubes, the reactor coolant pump (RCP), the cross-over leg, and the cold leg (CL). The pressurizer (290) is attached to the hot leg of loop 2 through the surge line (pipe 285).

The SG secondary side (components 600 to 650 for SG1, 700 to 750 for SG2 and 800 to 850 for SG3) is modeled by means of riser, steam separator (“separator” components 610, 710 and 810), steam dryer and steam dome, and downcomer.

The calculation model also contains the main steam lines with the relief valves, steam header, turbine throttle and bypass valves, and secondary water feeding systems including the main and auxiliary feedwater systems.

Three independent safety injection (SI) trains are modeled by time dependent volumes and junctions, and connected to the cold legs of each loop, with one high pres-

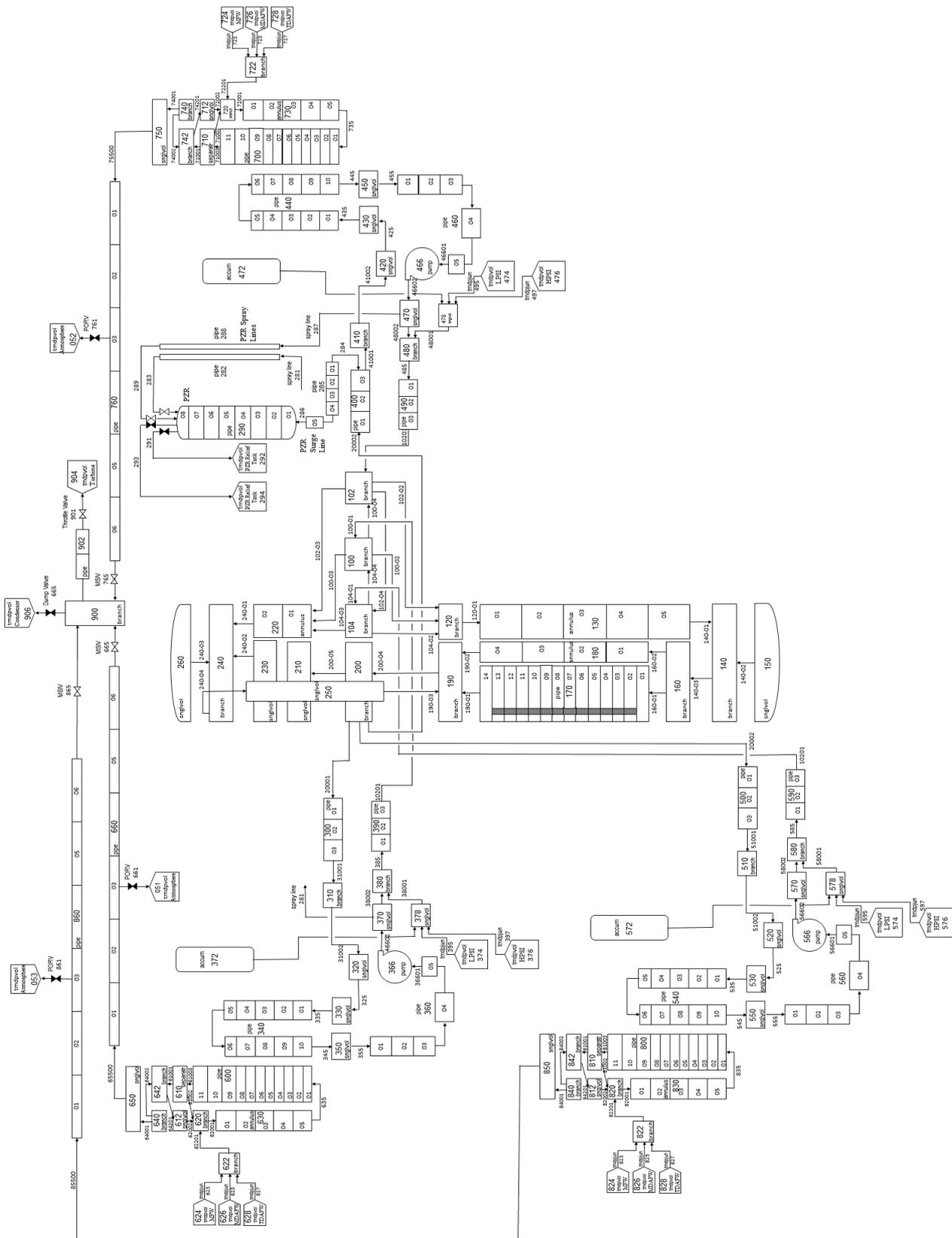


Figure 1: Nodalization scheme of PWR RELAP5 model.

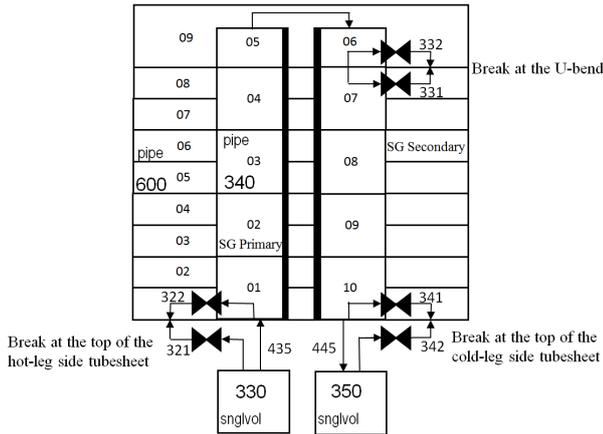


Figure 2: Investigated steam generator break locations.

sure safety injection (HPSI) line and one low pressure safety injection (LPSI) line. The accumulators are modeled by means of the RELAP5 “accum” components.

The control and protection systems are modeled using the RELAP5 specific control variables. Several trips are considered in the model to simulate all the prescribed system responses during the calculated transients.

Investigated break locations are at the top of the hot-leg side tubesheet, U-bend and at the top of the cold-leg side tubesheet of the SG and they are depicted in Fig. 2. All U-tubes are simply modeled as an averaged single flow channel.

A double-ended guillotine break/rupture is modeled by means of two valve junctions connecting a primary side pipe and a secondary side pipe. In order to simulate the tube break, the valve junctions are opened for 0.001 sec. The break flow is calculated using the RELAP5 default critical flow model.

The transient calculations were preceded by the steady-state run. The comparison of the main steady-state features of the plant and calculation model is shown in Tab. 1.

3. Results

3.1. SGTR Transient Analysis

Figs 3 through 5 show the primary and secondary system pressures for the simulated SGTRs at the top of the hot-leg side tubesheet. The tube ruptures cause a decrease in reactor coolant system pressure. For all simulated events, the reactor trip occurs due to the low pressurizer pressure. The reactor trip signal automatically trips the turbine and MFW (Main Feedwater) system. Immediately after that, all RCPs start to coastdown because the PZR water level

Table 1: Initial conditions of the computational model (NSSS = Nuclear Steam Supply System, PZR = Pressurizer, SG = Steam Generator).

Parameter	De-sign	Calcu-lation	Er-ror, %
NSSS Power, MW	2775	2775	0.0
Total vessel mass flow rate, kg/s	13800	13870	0.6
Primary pressure, MPa	15.51	15.51	0.0
PZR water level, %	61	59.9	0.3
Loop mass flow rate, kg/s	4575.7	4626.0	1.1
Core bypass mass flow rate, kg/s	511.9	522.3	2.0
Guide tube mass flow rate, kg/s	226.2	230.6	2.0
Hot leg temperature, °C	323.25	323.25	0.0
Cold leg temperature, °C	288.35	287.75	0.2
Pump speed, rpm	1185	1185	0.0
Feedwater mass flow rate, kg/s	1536	1556	1.3
Steam mass flow rate, kg/s	1536	1556	1.3
Steam line pressure, MPa	6.24	6.28	0.6
SG water level (narrow range), %	50.0	50.2	0.4

drops below 12%. The safety injection (SI) is triggered by the low-low pressurizer pressure signal despite maximum charging flow, and the proportional and back-up heaters being fully on. The pressurizer heaters are turned off when the pressurizer level drops below 17%.

The turbine trip and following feedwater pump trip causes an increase in steam generator pressure and a rapid drop in steam generator water level. The water levels in PZR and steam generators are shown in Figs 6 to 8. The decrease in SG level generates the auxiliary feedwater actuation signal. The turbine driven auxiliary feedwater is supplied immediately. The motor driven auxiliary feedwater is supplied with a small time delay needed for pump loading. After some time, due to the reactor trip and actuation of

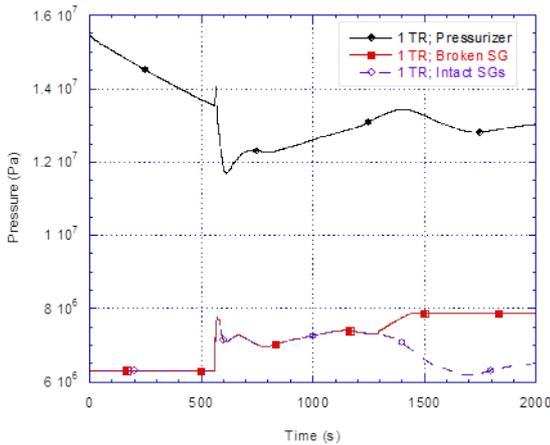


Figure 3: Primary and secondary system pressures. Single U-tube break at the top of the hot-leg side tubesheet.

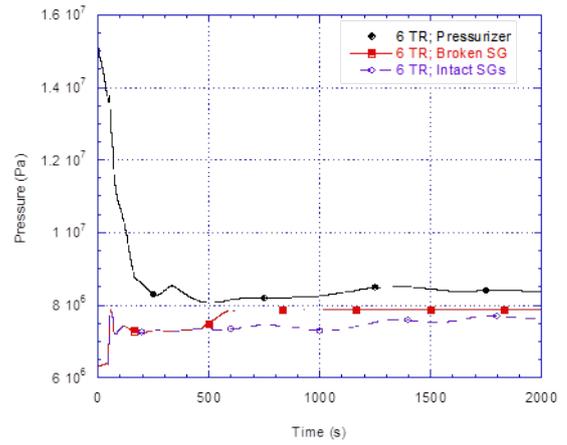


Figure 5: Primary and secondary system pressures. Rupture of six U-tubes at the top of the hot-leg side tubesheet.

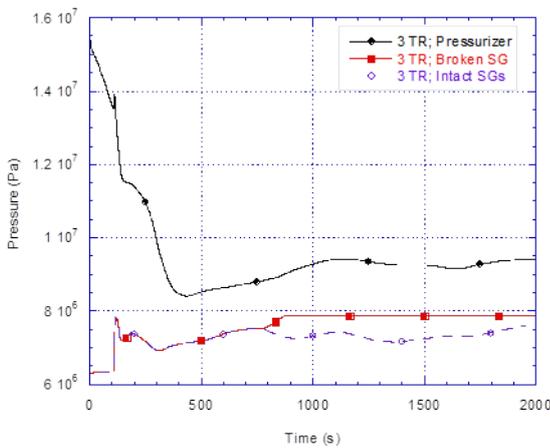


Figure 4: Primary and secondary system pressures. Rupture of three U-tubes at the top of the hot-leg side tubesheet.

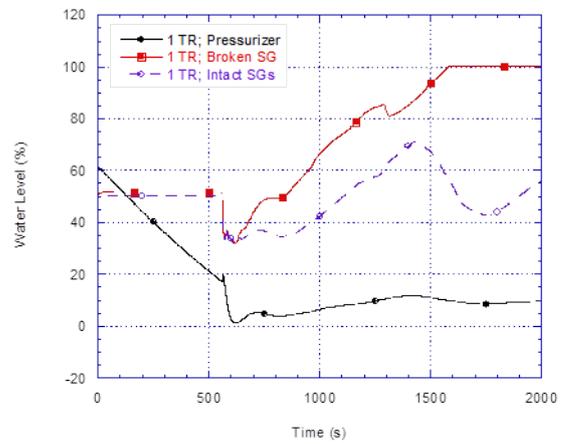


Figure 6: Collapsed water level in pressurizer and steam generators. Single U-tube break at the top of the hot-leg side tubesheet

the SIS (Safety Injection System), steam generation at the steam generator secondary side decreases and the water level starts to rise again.

The auxiliary feedwater is supplied to SGs until the narrow range water level reaches 8%. The main steam isolation valve is automatically closed on a high-level signal in the affected steam generator. After the MSIV (Main Steam Isolation Valve) is closed, the secondary system pressure increases and reaches the MSIV opening set point. Fig. 9 shows the steam flow rate through the PORV (Power-Operated Relief Valve) on the affected SG.

The presented results relate to approximately the first 30 minutes of the transients. After this time operator action is necessary. Operator action consists of the further cooldown of the RCS using the intact SGs and depressurization of the primary system pressure using either pres-

surizer spray or pressurizer PORV to terminate the SI injection and stop the primary-to-secondary break flow. The results shown in Figs 3 through 9 are very similar to results presented in [11] for the simulation of multiple steam generator rupture under similar assumptions, i.e., the offsite power as well as all reactor control and protection systems are fully available during the transient, and there is no operator action. The results are also similar to results of single [12] and multiple [13] SGTR simulations for initial periods of the transients until operator intervention.

3.2. Effects of the Number of Ruptured Tubes (Break Size and Rupture Position).

The higher the number of ruptured U-tubes, the larger the primary-to-secondary break flow. Fig. 10 shows the break flow into the secondary system for one, three, and six

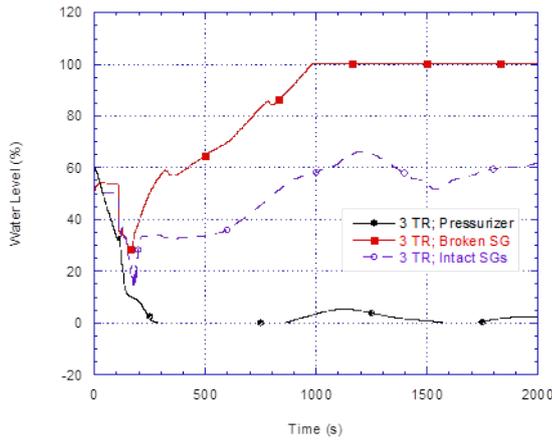


Figure 7: Collapsed water level in pressurizer and steam generators. Rupture of three U-tubes at the top of the hot-leg side tubesheet.

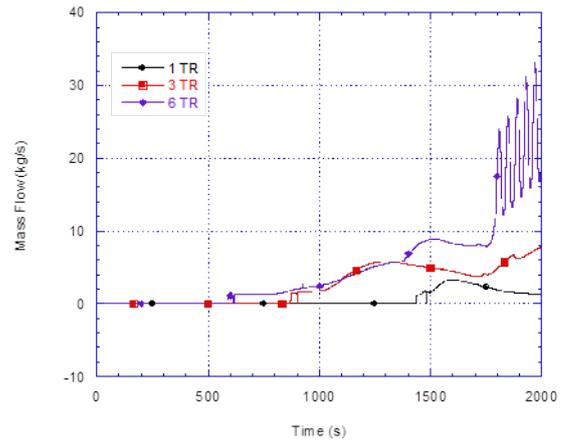


Figure 9: Steam flow rate through PORV on the affected SG.

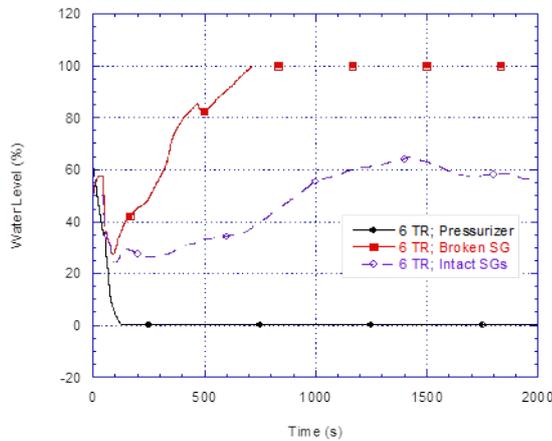


Figure 8: Collapsed water level in pressurizer and steam generators. Rupture of six U-tubes at the top of the hot-leg side tubesheet.

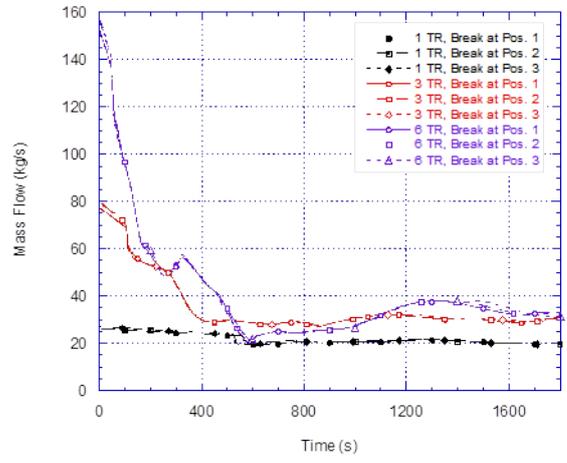


Figure 10: Mass flow break for one, three and six U-tubes at the top of the hot-leg side tubesheet.

ruptured tubes at different break locations. For the tube rupture model used in the calculations the initial pressure difference between the primary and secondary systems is not sufficient to result in critical flow, and the break flow is a pressure difference driven flow during the entire transient. The largest temporary break flow rate as well as the integrated outflow mass occurs for the break at the top of the cold-leg side tubesheet, where the primary coolant density is the greatest.

Fig. 10 shows the break and SI integrated flow masses for the break at U-bend.

The larger break flow results in faster depressurization of the primary system, and in turn, in earlier reactor and turbine trip, and SIS actuation. Figs 12 and 13 show the influence of the number of broken tubes on the SI actua-

tion and MSIV isolation times.

4. Conclusions

The SGTR event is the most common cause of radioactive leakage from the primary to the secondary side and a fairly frequent cause of reactor shutdown.

This paper presents the results of analysis of the single and multiple SGTR events, using RELAP5/Mod3.3. The calculations were performed for a typical 3-loop PWR of Westinghouse design with the classical FSAR assumptions of no operator action for the first 30 minutes of an accident.

The results obtained are very similar to results presented in available literature concerning simulations of single and multiple steam generator ruptures with similar assumptions.

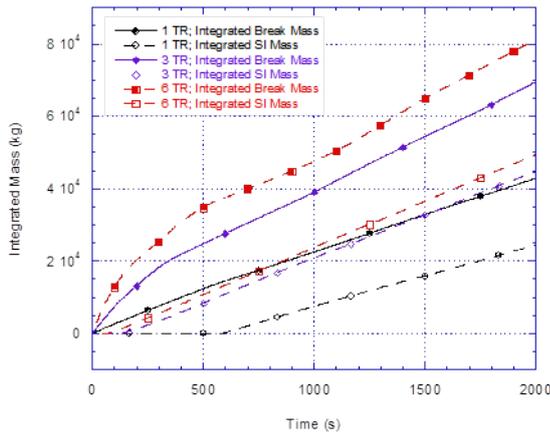


Figure 11: Integrated break and SI flow masses. Rupture of one, three, and six U-tubes at the top of the hot-leg side tubesheet.

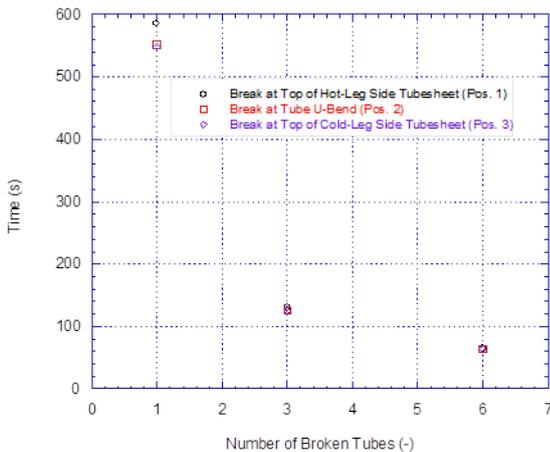


Figure 12: SI actuation time vs. number of ruptured tubes.

Direct rupture modeling is used from the pipe representing all the SG U-tubes to the secondary side with two valve junctions. This is the most conservative approach used for most FSAR calculations, due to the pipe representing the whole package of U-tubes maintaining the same pressure as the rest of the primary system.

The number of ruptured U-tubes has a major impact on the primary-to-secondary break mass flow rate. However, for the applied tube rupture model the influence of the locations of the breaks on the primary-to-secondary break mass flow is insignificant. The largest break flow rate occurred for the break at the top of the cold-leg side tubesheet, where the primary coolant density is the greatest.

Acknowledgments

The research was conducted within the framework of Research Task No. 9 entitled “Development and implemen-

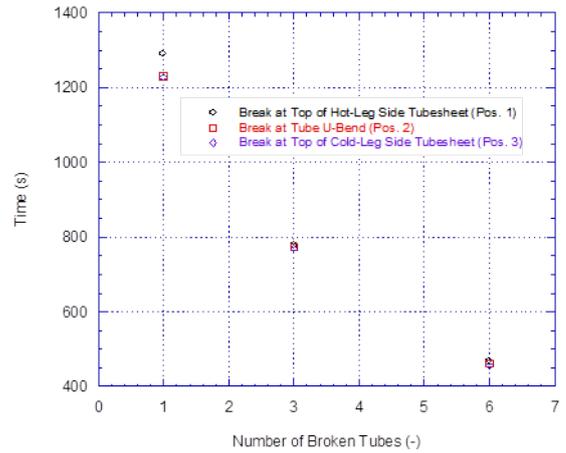


Figure 13: MSIV isolation time vs. number of ruptured tubes.

tation of safety analysis methods in nuclear reactors during disturbances in heat removal and several accident conditions”. The task is part of the project “Technologies for the development of safe nuclear energy” managed by Warsaw University of Technology and the National Center for Research and Development (NCBiR) in Warsaw, Poland.

References

- [1] J. P. Adams, M. B. Sattison, Frequency and consequences associated with a steam generator tube rupture event, *Nuclear Technology* 90 (2) (1990) 168–185.
- [2] P. E. MacDonald, V. Shah, L. Ward, P. Ellison, Steam generator tube failures, Tech. rep., Nuclear Regulatory Commission, Washington, DC (United States). Div. of Safety Programs (1996).
- [3] J. H. Jeong, K. Y. Choi, K. S. Chang, Y. C. Kweon, Effects of tube rupture modeling and parameters on analysis of msgtr event progression in pwr, Tech. rep., American Nuclear Society, 555 North Kensington Avenue, La Grange Park, IL 60526 (United States) (2002).
- [4] W. Van Hove, K. Van Laeken, L. Bartsoen, B. Centner, L. Vanhoenacker, Coupled calculation of the radiological release and the thermal-hydraulic behaviour of a 3-loop pwr after a sgtr by means of the code relap5, *Nuclear engineering and design* 177 (1) (1997) 351–368.
- [5] G. Jimenez, C. Qeral, M. Rebollo-Mena, J. Martínez-Murillo, E. Lopez-Alonso, Analysis of the operator action and the single failure criteria in a sgtr sequence using best estimate assumptions with trace 5.0, *Annals of Nuclear Energy* 58 (2013) 161–177.
- [6] A. Auvinen, J. Jokiniemi, A. Lähde, T. Routamo, P. Lundström, H. Tuomisto, J. Dienstbier, S. Guntay, D. Suckow, A. Dehbi, et al., Steam generator tube rupture (sgtr) scenarios, *Nuclear engineering and design* 235 (2) (2005) 457–472.
- [7] H.-K. Kim, S. H. Kim, Y.-J. Chung, H.-S. Kim, Thermal-hydraulic analysis of smart steam generator tube rupture using tass/smr-s code, *Annals of Nuclear Energy* 55 (2013) 331–340.
- [8] C. Lin, A. Wassel, S. Kalra, A. Singh, The thermal-hydraulics of a simulated pwr facility during steam generator tube rupture transients, *Nuclear engineering and design* 98 (1) (1986) 15–38.

- [9] J. Marn, M. Delić, L. Škerget, Experimental models of medium break loss of coolant accidents with and without steam generator tube rupture, *International journal of pressure vessels and piping* 80 (10) (2003) 737–744.
- [10] G. De Santi, Analysis of steam generator u-tube rupture and intentional depressurization in lobi-mod2 facility, *Nuclear engineering and design* 126 (1) (1991) 113–125.
- [11] J. H. Jeong, Y. C. Kweon, The effect of tube rupture location on the consequences of multiple steam generator tube rupture event, *Annals of Nuclear Energy* 29 (15) (2002) 1809–1826.
- [12] S. Lee, K. Kim, H. Kim, Y. Eun, Analyses of sgtr accident with mihama unit experience, *Nuclear Engineering and Technology* 26 (1) (1994) 41–53.
- [13] D. Mercurio, L. Podofilini, E. Zio, V. Dang, Identification and classification of dynamic event tree scenarios via possibilistic clustering: application to a steam generator tube rupture event, *Accident Analysis & Prevention* 41 (6) (2009) 1180–1191.