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Assessment of the Leakages from Primary to Secondary Side for Different Number of Tube Ruptures in the Steam Generator

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Abstract. This study presents an analysis of the results for guillotine double-ended tube ruptures in a steam generator. The reference power plant for this analysis is a Unit 6 at Kozloduy NPP site. It was selected beginning of fuel cycle. RELAP5/MOD3.3 computer code has been used to simulate the transients for VVER-1000/V320 NPP model. The paper discusses comparison of different break sizes of double-ended tube ruptures for Steam Generator in loop#1 with operator actions.

The purpose of this analysis is to consider the behaviour of the nuclear power plant, also the operator actions are considered in several cases to analyse possible errors from the human and technical factor. The main difference of the investigated accident compare to others is that operators have to take actions after analysing symptoms and understanding the event. The other feature of this event is that incorrect actions can lead to bypassing the containment structure and releasing radioactivity into the environment.

The scenario describes the actions of the operator and his reactions, as well as equipment failure. The developed scenario is based on the published investigation on this topic during developing and implementation of Emergency Operating Procedures (EOPs) in KNPP. So, the work is contributing in the reviewing and confirming the correctness and effectiveness of the prescribed actions of the operator, as well as to evaluate the reaction of the Nuclear power plant (NPP) during the progression of the accident.

Keywords: Safety analysis, Steam Generator Tube Rupture, EOPs, Nuclear reactor

1 Introduction

The purpose of the present analyses is to analysed NPP behaviour in case of comparison of different number of SG tubes double-ended ruptures in the Steam Generator (SG) #1 with operator actions and an assessment of primary to secondary side flow rate leakages until creating conditions for further cooldown of the reactor system. Primary to secondary leaks are one of the most serious events threatening in the Defence in depth of NPPs.

The selected initiating event is “Double-ended break of different number of pipelines in SG#1”. It was performed three calculations for SGTR double-ended break of single pipeline in SG#1 (as a case#1), double-ended break of two pipelines in SG#1 (as a case#2) and double-ended break of three pipelines in SG#1 (as a case#3). In this way, a set of calculations are performed for Steam Generator tube ruptures (SGTRs) event with investigation in this way leakages with different break sizes in a Steam Generator #1. The operator actions are included in the performed transient with a RELAP5/mod 3.3 computer code. It also provides an opportunity to assess the progression of a nuclear accident under different conditions. The reference power plant for this analysis is a Unit 6 at Kozloduy NPP site. This research was developed in the frame of the project CAMMIVER [1].

To simulate the SGTR event properly, it is necessary the operator action to be involved. The main difference of this accident compare to others is that the operators have to take actions after analysing symptoms and understanding the event. Such accidents could provide a direct release path for contaminated primary coolant to the environment via the secondary side in case of earlier closing of Fast Acting Isolating Valve (BZOK) and possible opening of Steam Dump to Atmosphere (BRU-As). Accumulation of water in the secondary side can also lead to an overfill condition which can severely aggravate the radiological consequences and increase the likelihood of complicating failures. The scenario describes the actions of the operator and his reactions, also should take into account the operator action time, as well as plant behaviour. The developed scenario is based on the published investigation on this topic during developing and implementation of the EOPs in KNPP as well as resolving some issues connected with a primary to secondary side (PRISE) events for VVER-1000 in Ref: [2, 3]. So, the work is contributing to the reviewing and confirming the correctness and effectiveness of the prescribed actions of the operator, as well as evaluating the reaction of the Nuclear power plant (NPP) during the progression of the accident.

One of the important NPP accidents is a steam generator tube rupture (SGTR) [4, 5] in the field of nuclear safety. The thermo-hydraulics parameters in primary loop under SGTR accident in VVER-1000 nuclear power plant are investigated in [6, 7] using the Relap5 code.

The methodology used in the performed analysis is based on a bounding approach developed by Ron Beelman [8]. It was followed in order to achieve the above-

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mentioned objectives. This methodology was applied successfully during the analytical validation of the symptom based emergency operating procedure (SB EOPs) for VVER reactors at KNPP [9, 10].

In the paper is presented scenario, where are included most of the operator actions and plant respond as: Reactor SCRAM, closing of the turbine stop and regulating valve and work of the Steam Dump to Condenser (BRU-Ks), isolating of the damaged SG by steam and feed water, control of depressurization of reactor coolant system (RCS) by Spray System to Pressurizer. Also, work of Makeup /Letdown system, work of main coolant pumps (MCPs), etc. The analyses are performed until stabilization of the primary and secondary pressure at level for significantly reducing the leakage. The cooldown of RCS further is not considered.

2 Brief Description of the Main Phenomena, Plant Respond and Observed Symptoms in the PRISE Event

- In case of PRISE event, the operator will see decreasing of primary pressure, decreasing of Pressurizer (Prz) water level, increasing of Make-up flow rate for supporting of Prz water level and primary pressure, increasing of water level in the damaged SG, as well as reducing of the feed water flow rate to this SG.
- Due to the leakage from primary to the secondary side it will be a radiation signal in a secondary side due to the N16.
- All these signals will demonstrate that there is a leakage from primary to secondary side. As the break flow rate is small, especially when there is only a single tube rupture, it will take some time for the operator to analyse and understand the situation and to take actions. Compare to many other accidents the PRISE in case of single pipeline break will not cause reactor SCRAM automatically.

3 Short Description of RELAP5 Model of VVER-1000 Reactor Type

RELAP5/MOD3.3 computer code [11] has been used to simulate the transients for VVER-1000/V320 NPP model [12, 13]. The model was developed at Institute for Nuclear Research and Nuclear Energy (INRNE-BAS) for analyses of operational occurrences, abnormal events, and design basis scenarios [14, 15]. The actual four-loop system was modelled by four single loops for primary and secondary sides. The model provides a significant analytical capability for the specialists working in the field of NPP safety and was compared against with the other code [16]. In the RELAP5 model for VVER-1000/V320 NPP are included reactor vessel; core region represented by three channels; pressurizer system including heaters, spray and relief valves; safety system including high-pressure pumps (HPPs), four accumulators and low-pressure injection pumps. In the model, also is presented a make up /drain system including connection (control) with pressurizer.

In the VVER-1000 input model, the primary system has been modelled using four coolant loops each one including one MCP and a horizontal SG. The thermal-hydraulic model configuration provides a detailed representation of the primary, secondary, and safety systems. The reactor vessel model includes a downcomer, lower plenum, and outlet plenum. The pressurizer (PRZ) system includes heaters, spray, and pressurizer relief capability. The safety system representation includes accumulators, high and low pressure injection systems, and the reactor scram system. The model of the make up and blow down systems includes the associated control systems.

Secondary side is developed too and is presented by eight SG safety valves, four BRU-A valves, BRU-K valves, steam pipe lines (including main steam header) and turbine including regulating valve in front of the turbine. The horizontal steam generator (SG) has been modelled. The model of natural circulation in horizontal SG has been presented in this RELAP5 VVER-1000 model. A separator model and the perforated sheet have been modelled in SG model, too. Main cooling pump (MCP) has been developed using homologous curves of real pumps.

RELAP5 heat structures are used to represent vessel structural internals (core barrel, core baffle, lower and upper plates, protective tube block and etc.) and the reactor vessel. Heat transfer from the primary coolant to the water of the secondary side is modelled using heat structure components. For the simulating guillotine double- ended single pipe line break "PRISE" in SG#1 is selected place between the end of pipe and cold collector at the elevation of around 2.0 m. It is shown in Figure 1.

4 The Main Events in the Scenario SGTR

1. Simulation of guillotine double-ended break of different number of pipelines (PRISE) in SG #1 at 0.0 s.
2. Reactor SCRAM by operator action with delay of 600 s (the assumption is made based on a published PRISE events and EOPs analytical validations [17]).
3. Due to the Reactor SCRAM the signal for closing of Turbine Stop Valve (Main Steam Isolating Valve - MSIV) is activated in 10.0 s delay after the Reactor SCRAM. The MISV is closed for 2 s.
4. Isolating of the main feed water (MFW) to the damaged SG#1 at 900 s.
5. Switching off the MCP #1 on damaged loop at 900 s.
6. All BRU-K are opened and start to control secondary pressure in main steam header (MSH) at around 6.0 MPa until 1000 s.
7. In all cases primary pressure will be controlled by spray to Pressurizer (by head created from MCP) and will be supported also by BRU-K at around 6.0 MPa. In case of loosing of a MCP the depressurization could be performed by opening of safety valve (SV).

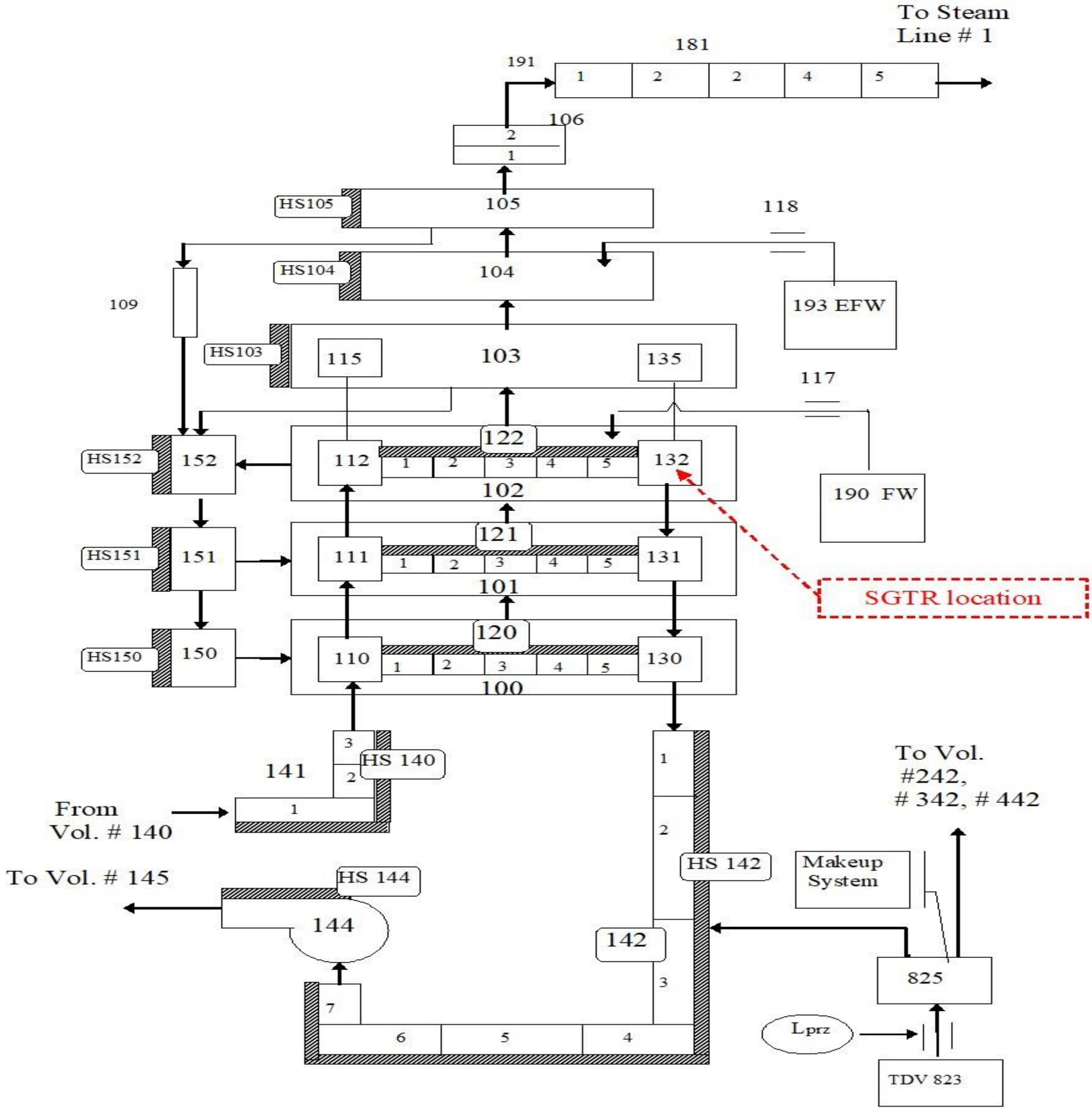

Figure 1. Kozloduy Steam Generator RELAP5 Four Loops Model.

8. Reducing the primary pressure by spray in Presurizer by MCP#1 by opening of spray valve up to 70 kgf/cm^2. After reducing primary pressure operator closes the BZOK. The reason to delay in closing of BZOK (Fast Acting Isolating Valve (FAIV)) is to avoided possible increasing of secondary pressure in damaged SG#1 and opening of BRU-A and this way dumping steam in atmosphere.

9. The operator opens the BRU-Ks to stabilize secondary pressure at 52 kgf/cm^2 and try to keep primary side pressure at around 70 kgf/cm^2 by PRZ spray. The primary side pressure is controlled at that moment by PRZ spray and the work of BRU-Ks support this also, by shrinkage of primary coolant. The purpose of this action is to reduce secondary pressure below primary slightly for avoiding possible reverse flow rate.

10. End of calculation at 5000 s.

5 Initial and Boundary Conditions

- Reactor power: 100%;
- Beginning of cycle;
- Coolant temperature: nominal;
- Primary side pressure: nominal;
- Primary coolant flow: nominal;
- Steam pressure: nominal;

- Feedwater temperature: nominal;
- Pressurizer level: nominal (8.77 m) in the range of control for the nominal power;
- The controllers of the not damaged SGs #2, #3, #4 will continue to support water level to the 2.65 m;
- The SGs FW temperature: In the first 600 s the water temperature is 220°C (nominal). After the reactor SCRAM the FW temperature is changed from 220°C to 164°C due to loss of high temperature heaters (this is the temperature in the deaerator) and after that with reducing of 15°C per hour which is assumed is realistic.

6 Discussion of Results

The paper discuses different calculations for double-ended steam generator tube ruptures with operator actions. The calculations have presented consequences from flowing of the primary coolant to the secondary side until achieving of the plant state safe conditions. The behaviour of the important parameters, describing the accident progression, are shown on Figure 2 through Figure 11 as primary pressure, secondary pressure, primary to secondary side leakage, temperature of the coolant. The calculations were performed up to 5000 s into the transient time.

When the SGTR event occurs, the operator will notice an increase in the make-up flow rate, a decrease in the water level Prz, increasing of water level in damaged SG, as well as reducing of the feed water flow to this SG. Due to all these signals, the operator takes action to shut down the reactor approximately 10 min after the beginning of the transient. Following the reactor trip, initiated by the operator, core power rapidly decreases to decay heat levels. Turbine trip signal appears at 610.0 s due to the reactor SCRAM. Steam flow to the turbine is terminated.

One second after that all four BRU-Ks start to open and to maintain the secondary pressure the same as the pressure before closure of Main Steam Isolating Valves (MSIV). Five minutes after the reactor SCRAM the operator switches off the Main coolant pump #1 and stopped the feed water to the damaged Steam generator. At the 1000 s the operator begins cooling down by BRU-Ks and support 52 kgf/cm^2 in main steam header. The operator close BZOK after reducing the primary pressure at around 70 kgf/cm^2 at 1700 s. During transient the operator should be kept subcooling of 10°C in primary circuit. In this calculation is not reduce primary pressure by further spray after

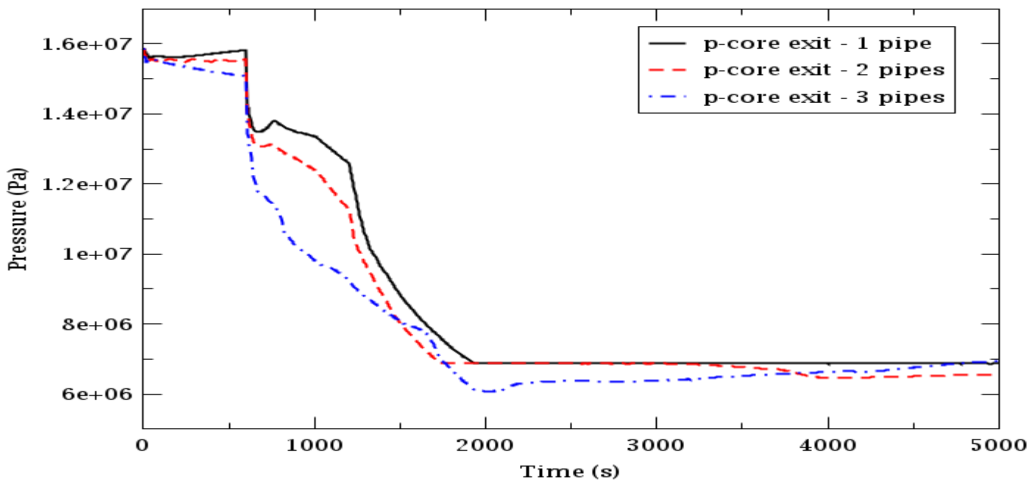

Figure 2. Pressure in primary side.

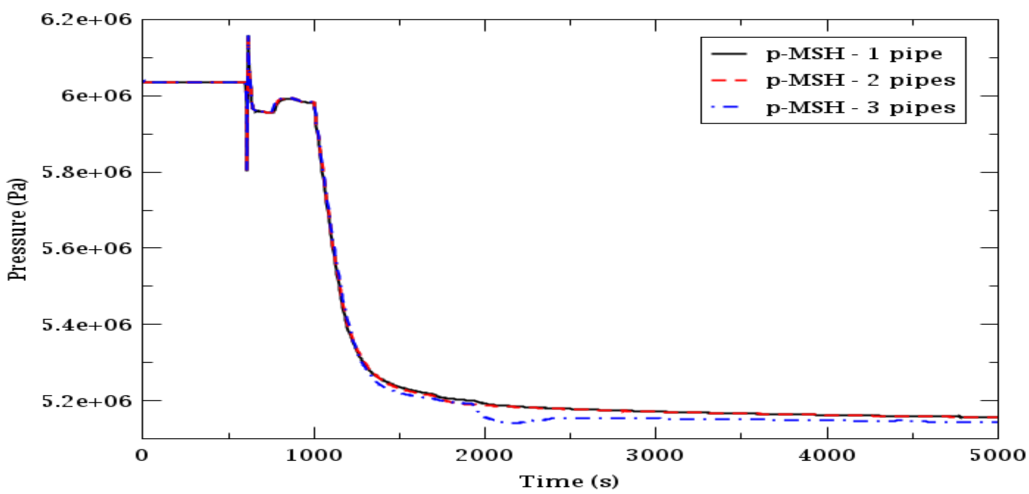

Figure 3. Pressure in secondary side.

Table 1. Chronology of the main events during event SGTR

No	Events	Case#1 time [s]	Case#2 time [s]	Case#3 time [s]
1.	Break	0.0	0.0	0.0
2.	Reactor SCRAM	600	600	600
3.	Turbine is isolated by closing of Main Steam Isolating Valve (MSIV) 10 s after SCRAM	610	610	610
4.	BRU-K #1, #2 #3 and #4 start to open and to support MSH pressure the same as it was 1 s before closing the MSIV	611	611	611
5.	MCP #1 is tripped	900	900	900
6.	The operator stops feed water to the damaged SG #1	900	900	900
7.	The operator starts cooling down by BRU-Ks with 60 deg/h and supporting of 52 kgf/cm^2 in main steam header (MSH)	1000	1000	1000
8.	The operator opens spray line from cold leg #1	1200	1200	1200
9.	The operator closes the BZOK of the damaged SG #1	1700	1700	1700
10.	Stabilization of secondary pressure at 52 kgf/cm^2	2800	2800	2800
11.	End of calculation	5000	5000	5000

reaching 70 kgf/cm^2. In Table 1 is presented chronology of the main events during the accident progression.

Pressure in primary and secondary side are presented in Figures 2 and 3. The results of the investigated events “double-ended steam line break” just before the fast acting isolating valve (BZOK) shows an immediate decrease in pressure in the main steam header and the faulted steam line. Comparison of the primary pressure for three cases, it is shown different behaviour due to different leakage of the break. It is observed stabilization of primary pressure at 70 kgf/cm^2 (6.86 MPa) for case #1, while for other cases the primary pressure is below. The primary side pressure

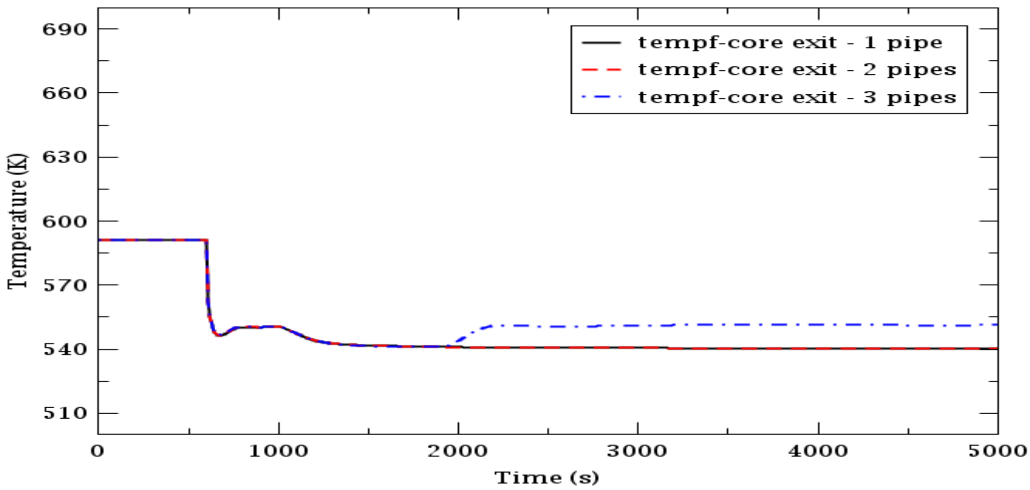

Figure 4. The primary side liquid temperature.

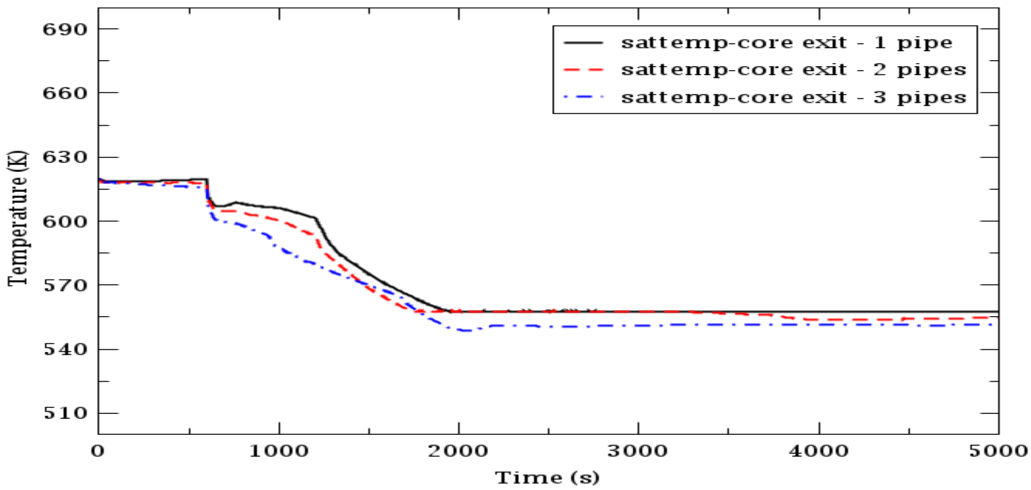

Figure 5. The primary side boiling temperature.

is controlled at that moment by Pressurizer (Prz) spray and the work of BRU-Ks contributes to this, also, by shrinkage of primary coolant. The pressure in the secondary side starts to decrease and following the rapid depressurization of damaged steam generator, there is a significant heat transfer from primary to secondary side, which also causes the primary side average coolant temperature, and pressure to decrease.

Behaviour of primary side liquid temperatures is presented in Figure 4. For depressurization is used spray in pressurizer from cold leg. After establishing more than 10 °C subcooling in hot legs, due to work of BRU-Ks, operator continues to depressurize keeping this margin of subcooling to the end of transient (see Figure 5). All accumulators have been isolated and none of the HPPs are used for cooling down following the procedures.

So, in the hot legs and in the reactor core, there is no void fraction, during the whole transient. After the reactor trip, the primary system transient follows the reactor trip and primary side pressure, coolant temperature and water level in pressurizer are additionally decreased.

The increasing of core exit temperature in case # 3, just before 2000 s, is due to losing of subcooling margin and transition from forced to natural circulation. After that moment all MCPs are tripped in case # 3 and the spray from the cold leg to the Prz is not effective. But, due to the bigger break in the case #3, the primary pressure in case #3 continue not to increase and is around 6.5–6.71 MPa, to the end of calculation. The slight increasing of primary pressure after 2000 s is coming due to transition to natural circulation. The support of primary pressure is also due to work of the BRU-Ks and supporting secondary side pressure around 5.2 MPa.

In Figures 6 to 9 are presented Steam generators water level for all calculations. The water level in damaged SG#1 starts to increase at around 1000 s for all cases, when operator switches on the BRU-Ks and the primary pressure is controlled by the spray from cold leg to pressurizer and use the MCP#1.

SG water level in the damaged SG increases up to 4 m for case#2 and case#3, while for case#1 is up to 3 m and main steam lines start to fill up with mixed primary and secondary coolants. Since the intact and ruptured steam generators are connected via the main steam header, no significant difference in pressures will be evident at this time (at around 1000 s.).

The maximum break flow rate from primary to secondary side is around 14 kg/s at 0 s for case#1, for case#2

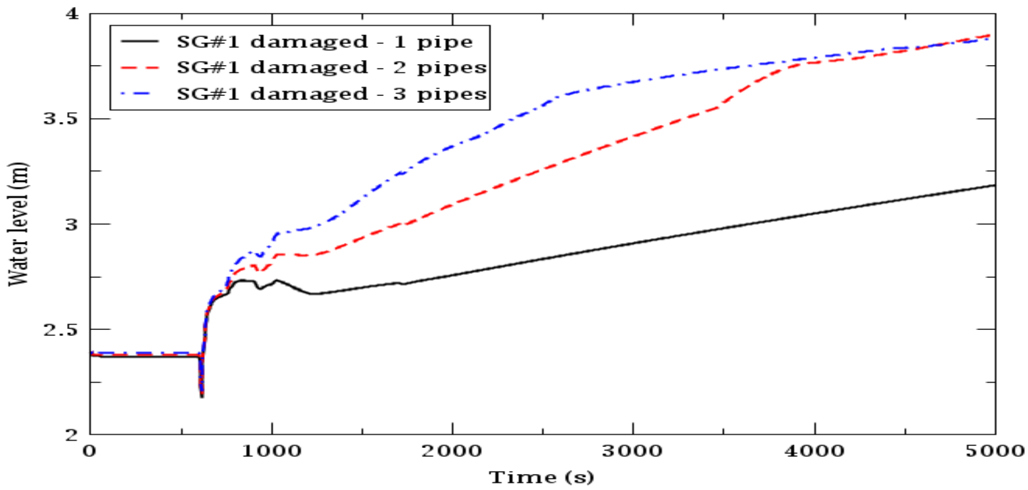

Figure 6. SG#1 water level –– damaged.

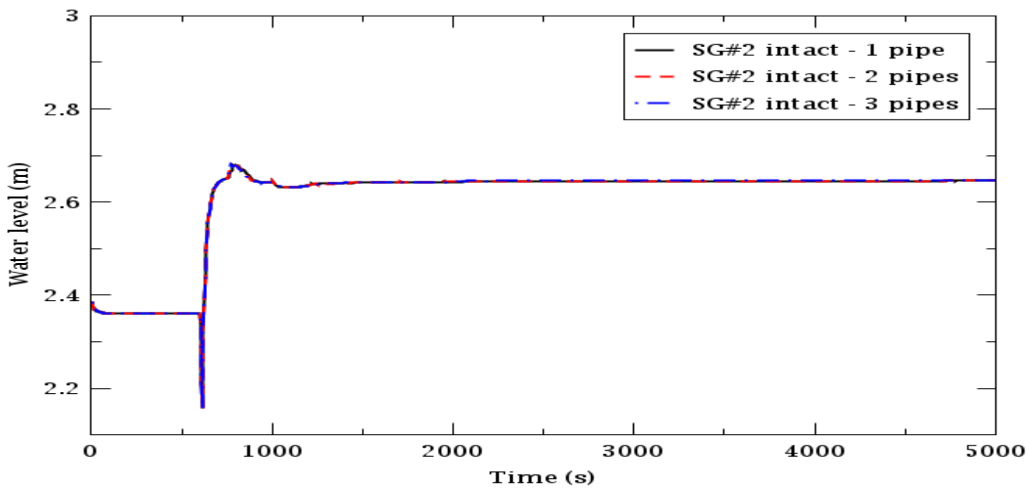

Figure 7. SG#2 water level – intact.

is around 25 kg/s and for case#3 is about 40 kg/s at the moment of the break opening and decreases later on (see Figure 10). It is observed different total mass flow for all cases, because of break sizes of leakage of the coolant. In these calculations, it is observed a decrease of water level in the pressurizer, an increase the coolant of the primary circuit feed water and a decrease the feed water of a damaged steam generator.

In Figure 11 is presented the total break flow rate from primary to secondary side is around 30 ton for the case#1, for case#2 is around 55 tons and almost 60 tons for case#3 at the end of the transient.

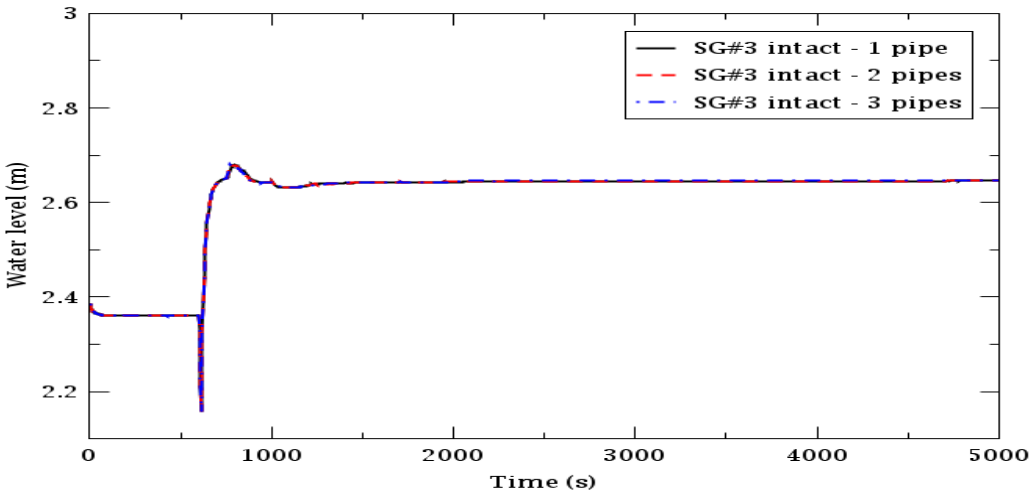

Figure 8. SG#3 water level –– intact.

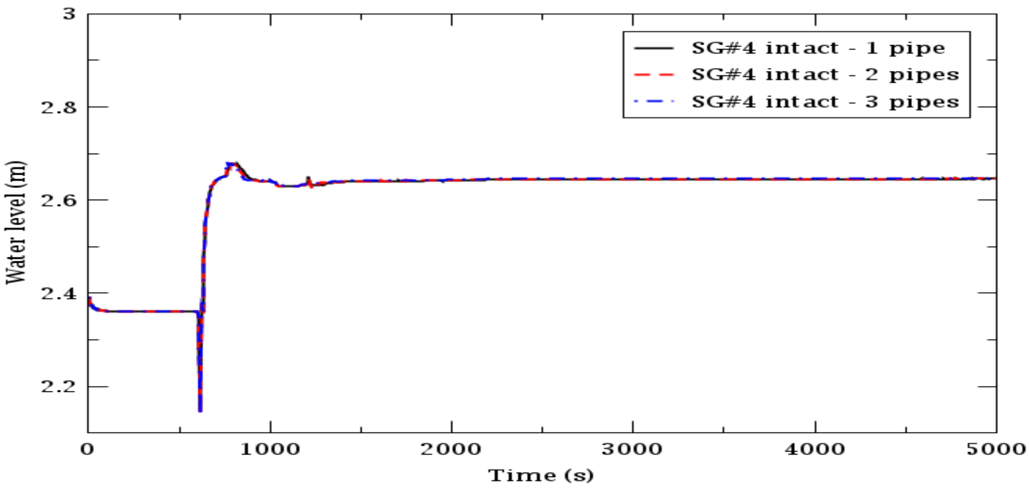

Figure 9. SG#4 water level – intact.

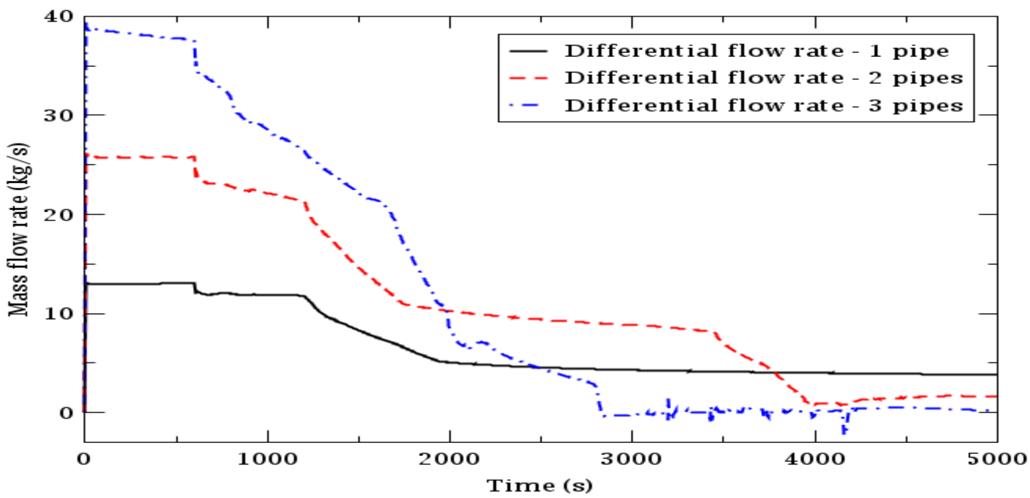

Figure 10. Break flow rates.

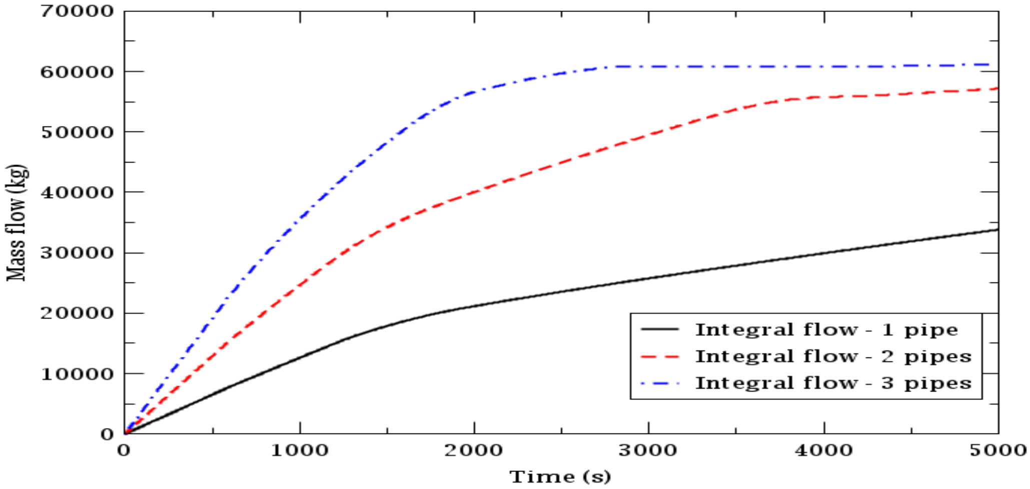

Figure 11. Integral Break flow rates.

7 Summary and Conclusions

In the paper is discussed plant respond on different number SG's tube ruptures, the main conclusions could be summarized as follow:

- In all 3 cases the plant behaviour corresponds to expected behaviour. In all cases is observed primary side depressurization, PRZ water level reducing, increasing of flow rate of Make up system, increasing of water level in damaged SG in any one of investigated cases and etc.
- The observed maxim break flow rate is in case #3 at 0 s is 40 kg/s with breaks of 3 pipes.
- The integral break flow rates are as follow: for 1 tube is 30 ton for 5000 s; for 2 tubes are 55 ton for 5000 s; for 3 tubes are 60 ton for 5000 s.
- Suggested operator actions lead to a successful core cooling and management of this transient if they are performed timely.
- All calculations demonstrate sufficient safety margin in relation to a SGTR event.
- The calculations demonstrate an effective strategy for preventing any secondary radiological release to the environment.
- The pressurizer is cooled down successfully, due to the work of the spray system of PRZ during primary depressurization. In the pressurizer has been injected water from the cold leg #1.

The proposed actions of the nuclear power plant operator lead to a successful management of the steam line break accident. In case of successful actions of the operator in an accident with SGTR in NPP, it did not lead to the core damage. As a whole the presented calculations allow to understood primary to secondary leakages during different number of tube breaks. With increasing the number of breaks it was observed losing of subcooling margin during primary depressurization and due to the transition from forced to natural circulation.

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