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Analysis of Steam Generator Tube Rupture (SGTR) Accident using RELAP5/MOD 3.3 and TRACE 5.0p5 Codes

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ABSTRACT

NPP Krško (NEK) input deck for best-estimate computer code RELAP5/MOD 3.3 has been developed at Faculty of Electrical Engineering and Computing (FER), Zagreb. Recently, the NPP Krško model for TRACE code based on RELAP5 model has been completed. Currently, for verification purposes for TRACE code, the on-transient qualification is performed by comparing the transient results with RELAP5 code. In this paper the results of Steam Generator Tube Rupture (SGTR) accident for NPP Krško using RELAP5/MOD 3.3 and TRACE 5.0p5 are presented. The SGTR event presents the threat by providing the direct path for primary coolant to the environment via the secondary side relief valves thereby bypassing the containment. Unlike other loss of coolant accidents, an early operator action is necessary to prevent the radiological release to the environment. The primary-to-secondary leakage causes the primary pressure drop which leads to reactor trip either on low pressurizer pressure signal or overtemperature DT (OTDT) signal. Furthermore, the Safety Injection (SI) signal may be generated due to continuous primary pressure decrease. After SI actuation the primary pressure will tend to stabilize at the value where SI flow equals the flow through the ruptured tube. The operator is expected first to determine that the SGTR event has occurred and to isolate the broken SG by closing the main steam isolation valve. The subsequent operator actions consisting of controlled cooldown and depressurization are aimed to stop the primary-to-secondary leakage on one side and on the other side to achieve conditions where Reactor Coolant System (RCS) is cooled via intact SG and the SI is isolated. First, the base analysis assuming the first operator action delay equal to 16 minutes after reactor trip was performed. Thereafter, the analyses were performed to find the delay times that would result in ruptured SG liquid solid conditions and liquid discharge through the ruptured SG relief valve. In general, all main trends were well reproduced by both codes and small differences between them were encountered.

1 INTRODUCTION

The advanced best-estimate computer code TRACE (TRAC/RELAP Advanced Computational Engine), ref. [1], is being developed by US NRC for the analysis of transients and steady-state behaviour in light water reactors. The best-estimate system code RELAP5/MOD 3.3, [2] for modelling of complex thermal-hydraulic systems is being used worldwide for the prediction of nuclear power plant behaviour in the case of transients/accidents. The TRACE model for NPP Krško is being developed on basis of the qualified RELAP5/MOD 3.3 model that is constantly being upgraded along with plant upgrades and modifications, ref. [3] and [4]. Currently, the TRACE model on-transient qualification is performed by comparing the transient results with RELAP5 code. TRACE model has been recently upgraded with detailed model of safety injection system with realistic models of both High Head (HHSI) and Low Head (LHSI) pumps. In this paper the results of analysis of Steam Generator Tube Rupture (SGTR) Accident for NPP Krško using both TRACE 5.0p5 and RELAP5/MOD 3.3 are presented. Previously, the TRACE model for NPP Krško was used to analyze the 3 inch Loss of Coolant (LOCA) analysis and the results were compared with RELAP5/MOD 3.3, ref.[5].

The SGTR event causes the contamination of the secondary side due to leakage of the radioactive coolant from the RCS through the broken SG tube (s). The primary-to-secondary leakage causes the surge from the pressurizer and Reactor Coolant System (RCS) depressurization which leads to an automatic reactor trip (on low pressurizer pressure or OTDT trip) and Safety Injection (SI) actuation. Unlike other loss of coolant accidents, an early operator involvement is necessary to stop the leakage and prevent the radiological release to the environment. After SI actuation the RCS pressure will tend to stabilize at the value where SI flow equals the flow through the ruptured tube. The operator shall determine that the accident has occurred by observing: 1) the difference between steam and feedwater flow (if detected before reactor trip) and 2) the increase of the radiation level in the affected SG. The recovery procedure performed by the operator is primarily aimed to isolate the ruptured SG and to terminate the break flow before water level in the affected SG rises to the main steam pipe and liquid is discharged through the ruptured SG relief valve.

We have analyzed a double-ended break of one U-tube at the tube outlet in the loop with the pressurizer (SG 1). The operator actions begin after automatic reactor trip by isolating the ruptured SG followed by controlled cooldown. The operator will also start the maximum charging flow in order to recover RCS inventory. Further, the operator action includes the cooldown and depressurization in order to terminate the SI and the primary-to-secondary leakage while maintaining the safe plant status, i.e., the adequate RCS subcooling margin, as well as pressurizer and intact SG inventory. Finally, the operator stops the safety injection and establishes normal charging and letdown. A detailed analysis will be presented for the base case analysis with the delay for the first operator action equal to 16 minutes after reactor trip. The additional analyses were performed in order to determine the delay time that would result in ruptured SG liquid solid condition and liquid discharge through the ruptured SG PORV.

2 CALCULATIONAL MODEL FOR NPP KRŠKO

The NPP Krško model for RELAP5/MOD 3.3, ref. [3] and [4] has been developed at FER. The model is being upgraded along with changes accompanying plant modernization modifications, e.g. Steam Generators (SG) replacement and power uprate in 2000, Resistance Temperature Detector Bypass Elimination (RTDBE) in 2013 and Up-Flow Conversion in 2015. RELAP5/MOD 3.3 model for NPP Krško consists of 533 control volumes and 572 junctions. The total number of heat structures is 403 with 2367 mesh points. There are 723 control variables and 197 variable and 221 logical trips to model the control systems as well as protection and Engineered Safety Features (ESF) behaviour, e.g., automatic rod control system, pressurizer pressure and level control system, steam generator level control, steam dump control, safety injection and auxiliary feedwater system. The NPP Krško nodalization for TRACE code has been developed using RELAP5/MOD 3.3 model, see Figure 1 and Figure 2. The model consists of 392 hydraulic components, i.e., 178 PIPEs, 25 BREAK components, 5 TEEs, 10 FILLs, 2 TIME DEPENDENT JUNCTIONs (TDJs), 6 PUMPs and 57 VALVEs, respectively. For the reason of brevity only parts of TRACE nodalization are presented here.

The Reactor Pressure Vessel (RPV) is modeled using pipe components 101 through 175, see Figure 1. Active core is represented with PIPE 111 consisting of 12 fluid cells; the Rod Control Cluster Assembly (RCCA) empty guide tubes inside core are presented with PIPE 113. The region between baffle and barrel is represented with PIPE 115. All the components that are parallel to the active core (113, 115 as well as part of the downcomer that is represented with PIPE 175) consist of 12 fluid cells. The flow paths that bypass the active core include the flow through the baffle-barrel region (PIPE 115), the empty guide tubes inside the core (PIPE 113) as well as upper downcomer (PIPE 165) – upper head (PIPE 151). Hot legs in each loop are modelled with five control volumes (PIPE 201 through 209 for the first loop and PIPE 301 through 309 for the second loop), intermediate legs with five control volumes (PIPE 251 through 259 for the first loop and PIPE 351 through 359 for the second loop) and cold legs with five volumes (PIPE 271 through 279 for the first loop and PIPE 371 through 379 for the second loop, respectively. Reactor coolant pumps (PUMP 265 and 365) are connected with PIPEs 259 and 271 (first loop) and PIPEs 359 and 371 for the second loop, respectively. Steam generator primary side is modelled with PIPEs 215 through 245 (loop 1) and PIPEs 315 through 345 (loop 2).

The break is located at a bottom of one tube in the loop 1 (loop with pressurizer). The sensitivity analyses to identify the location of the break resulting in a maximum break have shown that the outlet-cold side of the tube results in maximum break flow. The broken U-tube (volumes 220, 224, 226, 228, 234, 236, 238 and 242) is modeled separately with realistic tube sectional and heat transfer area.

On the secondary side, see Figure 2, SG 1 downcomer is modelled with PIPEs 411 and 413, heat exchanger section is represented with PIPEs 415 and 417. The region from the top of U-tubes to the bottom of separator is modelled with PIPE 419 entering the separator that is modelled with TEE component 421. The steam outlet and liquid return from the separator are represented with TEE 423 and PIPE 427, and the upper plenum bypass volume is modelled with PIPE 425. The SG 1 steam dome is modelled with PIPE 429 that is connected with the first volume in steam line 1, i.e., PIPE 451. The main steam isolation valves (VALVE 498 in steam line 1 and 598 in steam line 2) connect the PIPE 461 (PIPE 561 in steam line 2) outlet with steam header that is modelled with TEE 601. There are one relief valve (VALVE 482 in steam line 1 and 582 in steam line 2) and five safety valves in each steam line (VALVEs 484 through 494 in steam line 1 and VALVEs 584 through 594 in steam line 2). Currently, the steam dump system is modelled with the TDJ 607 representing the total flow of steam dump valves to condenser (BREAK 608).

Both Emergency Core Cooling System (ECCS) loops have been modelled, see Figure 2. The model consists of 32 PIPEs, 24 VALVEs, 4 PUMPs and 5 BREAK components, respectively. Components 701 (801) to 732 (832) represent the accumulators and high pressure injection system connected to the cold legs (PIPEs 273 and 373). Components 741 through 883 are used to model the low pressure as well as a part of high pressure injection system connected to RPV downcomer (PIPE 171) via Direct Vessel Injection (DVI) lines.

There are 107 heat structures (HTSTR components) in TRACE model and 3 POWER components defining the reactor power (point-kinetics with table lookup of reactivity and moderator as well as Doppler reactivity feedback) as well as pressurizer proportional and backup heaters. NEK model uses cycle 29 Beginning of Life (BOL) point kinetics data. The total number of Control blocks, Signal variables and Trip components is equal to 387, 270 and 74, respectively.

Double ended break is modeled by opening of the two valves (valves 992 and 993) connecting the break ends (volumes 242 and 238, respectively) with the heat exchanger section (volume 415) and closing the valve 994 that connects the ends of tube before break occurrence. The analyses were performed with conservative assumption regarding ruptured SG overfill concern. Thus, the Auxiliary feedwater (AFW) pumps were started immediately after reactor trip and AFW flow in the ruptured SG was terminated first after SG level has exceeded 75%. In the analysis, the loss of offsite power after reactor trip was assumed, i.e., the Reactor Coolant Pump (RCP) trip in both loops as well as the isolation of main feedwater resulted immediately after reactor trip. The steam dump system was assumed unavailable. Thus, the operator used the intact SG relief valve to perform the controlled cooldown and pressurizer PORV valve for subsequent RCS depressurization. The safety injection signal will be actuated on low pressurizer pressure soon after reactor trip. The safety injection flow ensures the RCS inventory following the break, but on the other hand it maintains the pressure difference between primary and secondary side and promotes the increase of the inventory on the secondary side. Thus, the operator actions are aimed to achieve the conditions when SI can be stopped; i.e. the subcooling greater than 19 K and the sufficient RCS inventory indicated by pressurizer level greater than 15% as well as required intact SG inventory assuring heat sink.The operator actions started with the isolation of the broken SG (16 minutes after reactor trip in the base case). Two minutes later the operator started the RCS cooldown by opening the intact SG PORV until the core exit temperature fell below the setpoint. The core exit temperature setpoint was determined using Emergency Operating Procedure (EOP) and it depends on the ruptured SG pressure at time when cooldown started. The operator also established the maximum charging flow (36 m^3/hr) in order to help maintain the RCS inventory. The intact SG narrow range (NR) level is maintained in the range (20, 70%) using auxiliary (AFW) flow. Two minutes after cooldown had been finished the RCS depressurization using pressurizer PORV was performed until either RCS pressure fell below the ruptured SG pressure or pressurizer level exceeded 66%. The operator then stops SI pumps (30 seconds delay) if the conditions for SI termination are fulfilled. Six minutes later, the operator terminates the maximum charging flow and establishes normal charging and letdown flow. After terminating the SI flow and with primary pressure reduced, the primary-to-secondary leakage will stop. The operator is then expected to maintain the stable conditions by controlling the core exit temperature at its setpoint value by ON/OFF opening of the intact SG PORV and to control the intact SG inventory using AFW flow.

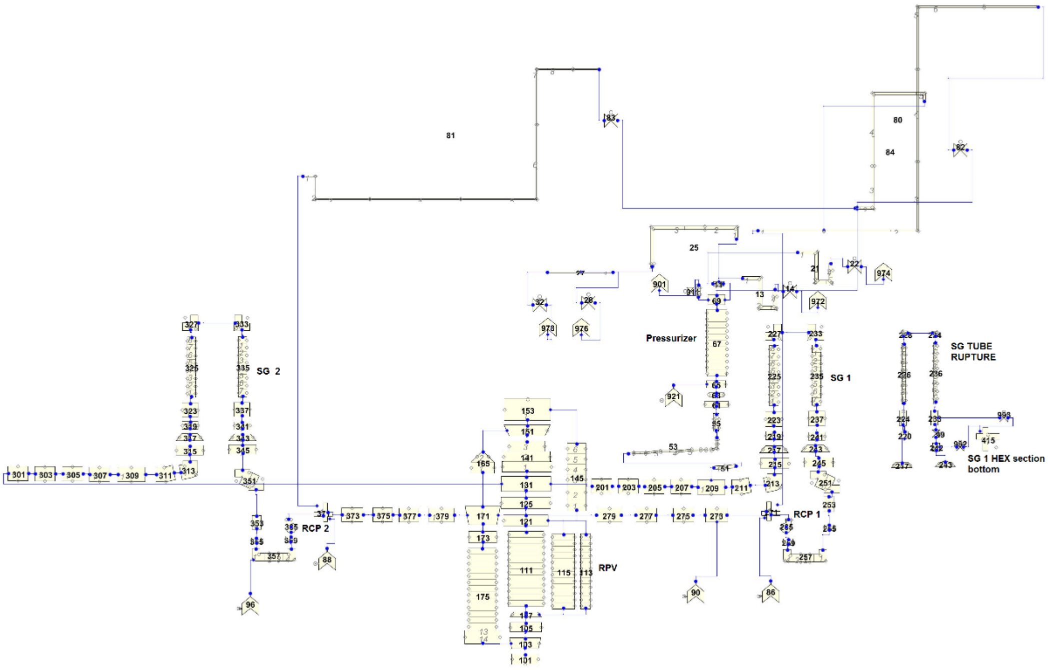

Figure 1: TRACE nodalization for NPP Krško: primary side with broken SG tube

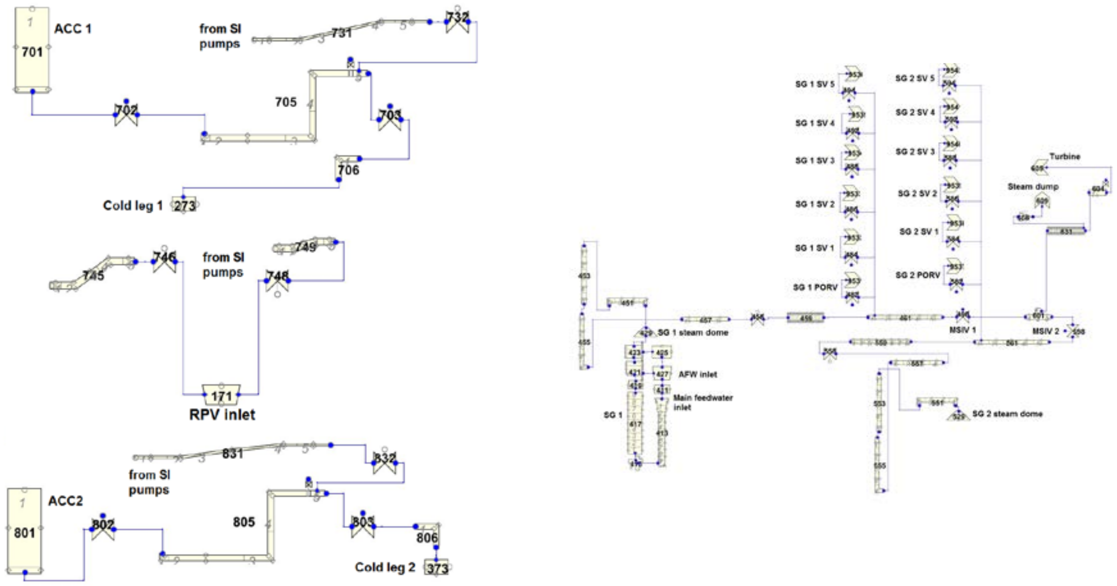

Figure 2: TRACE nodalization for NPP Krško: part of SI system (left) and SG 1 secondary side with main steam system (right)

3 ANALYSIS OF SGTR

Following the transient begin, the primary pressure decreases due to loss of inventory through the break, see Figure 3. The RCS subcooling decreases due to pressure decrease and OTDT reactor trip signal trips the reactor (113.9 s after transient begin in RELAP5 and 112.1 s in TRACE), see Figure 4 and Table 1. Before reactor trip, the reactor power was reduced due to negative moderator reactivity feedback caused by coolant density decrease. In the analysis, the loss of offsite power after reactor trip was assumed, i.e., the RCP trip in both loops as well as the isolation of main feedwater resulted immediately after reactor trip. The CVCS charging and letdown flow were isolated as well. In the analysis it was conservatively assumed that the auxiliary feedwater (AFW) flow (80 m^3/hr for each SG) was initiated immediately after reactor trip and it was isolated in the ruptured SG first after SG level exceeded 75%. For intact SG it was assumed that AFW flow was controlled to maintain SG 2 NR level in the range (20, 70%), see Figure 5. After reactor trip, RCS pressure continued to decrease until safety injection was actuated, see Figure 4. In the base case analysis the operator isolated the ruptured SG by closing the main steam isolation valve (VALVE 498) 16 minutes after reactor trip. Two minutes later (1192 s after transient begin in TRACE and 1195 s in RELAP5) the operator started the cooldown to the core exit temperature (CET) setpoint value that depends on ruptured SG pressure at the start of cooldown (536.8 K for RELAP5 and 539.3 K for TRACE) by fully opening the SG 2 PORV. In the analysis it was assumed that operator maintained the CET at these setpoint values till the end of simulation. The cooldown rate decreased along with the decreasing the SG 2 pressure, see Figure 2. The cooldown lasted 8.7 minutes in RELAP5 and 8.9 minutes in TRACE. The operator also started the maximal charging flow (36 m^3/hr) in order to maintain the RCS inventory due to break. The SG 2 secondary side inventory was significantly reduced during cooldown because the AFW was closed until SG level fell below 20%, see Figure 5. The subsequent ON/OFF behaviour of SG 2 PORV can be observed by temporary SG 2 NR level increase due to rise of water droplets during PORV operation. Two minutes after the cooldown has ended the operator initiated RCS depressurization using pressurizer PORV. The depressurization lasted for 100 seconds in TRACE when the primary pressure fell below the ruptured SG pressure. In RELAP5 the depressurization has ended earlier because the pressurizer NR level exceeded 66%, see

Figure 5. Finally, the SI was terminated with 30 seconds delay after end of depressurization. One can observe the sharp increase of SI flow, see Figure 4, due to RCS depressurization. Six minutes after end of depressurization the maximum charging flow was stopped and the operator established normal charging and letdown. The leakage flow was stopped after SI termination for both RELAP5 and TRACE (1914 s after transient begin in RELAP5 and 1950 s in TRACE). The stable conditions for both codes were attained approximately 2500 seconds after transient begin when the increase of liquid volume in ruptured SG and the discharge through the ruptured SG were terminated, see Figure 6. The maximum liquid volume in ruptured SG (124.4 m^3 in RELAP5 and 122.4 m^3 in TRACE) at the end of simulation (3000 s) were well below total SG volume (152.7 m^3) and the small amount of steam was discharged through the ruptured SG PORV (1941.3 kg in RELAP5 and 1470.4 kg in TRACE), see Table 2 and Figure 6. In general, small differences between RELAP5 and TRACE were encountered and the main trends were well predicted by both codes.

The additional analyses were performed in order to determine the maximum operator delay time for the first operator action that would prevent the ruptured SG overfill and liquid discharge through the ruptured SG PORV (defined by SG PORV flow liquid fraction greater than 10%), see Table 2 and Figure 6. The analyses have shown that for RELAP5 the first operator action should not be initiated later than 33 minutes and for TRACE not later than 43 minutes after reactor trip in order to prevent liquid discharge. The more adverse results regarding filling the ruptured SG and subsequent liquid release in RELAP5 than in TRACE can be attributed to the fact that the break flow as well as SI flow were larger in RELAP5 than in TRACE, see Figure 3 and Figure 4.

Table 1: Time sequence of main events (Base case, delay=16 minutes)

Event	**RELAP5/MOD 3.3**	**TRACE**
Transient begin	0 s	0 s
Reactor trip (OTDT signal), Loss of offsite power	113.9 s	112.1 s
AFW initiation	113.9 s	112.1 s
SI actuation (on low-2 pressurizer pressure)	402.4 s	374.3 s
Ruptured SG AFW isolation	799.4 s	763.0 s
Ruptured SG steamline isolation	1073.9 s (960 s after reactor trip)	1072.1 s (960 s after reactor trip)
Initiation of cooldown	1193.9 s (1080 s after reactor trip)	1192.1 s (1080 s after reactor trip)
Max. charging actuation (36 m^3/hr)	1194.9 s (1081 s after reactor trip)	1193.1 s (1081 s after reactor trip)
Cooldown termination	1718.5 s	1730.4 s
Initiation of depressurization	1838.5 s (2 minutes after end of cooldown)	1850.4 s (2 minutes after end of cooldown)
Depressurization termination	1922 s (PRZ level >66.0%)	1950 s (pressurizer pressure < SG 1 pressure)
Stop SI flow (30 s delay)	1993.8 s	1986 s
Balance charging and letdown flow (stop maximum charging flow)	2282 s (6 minutes after end of depressurization)	2310 s (6 minutes after end of depressurization)
Break flow less than 0	1914 s	1950 s
Integral of SG 1 PORV mass flow (0-3000 s)	1941.3 kg	1470.4 kg
Maximum ruptured SG water volume reached	124.4 m^3	122.4 m^3

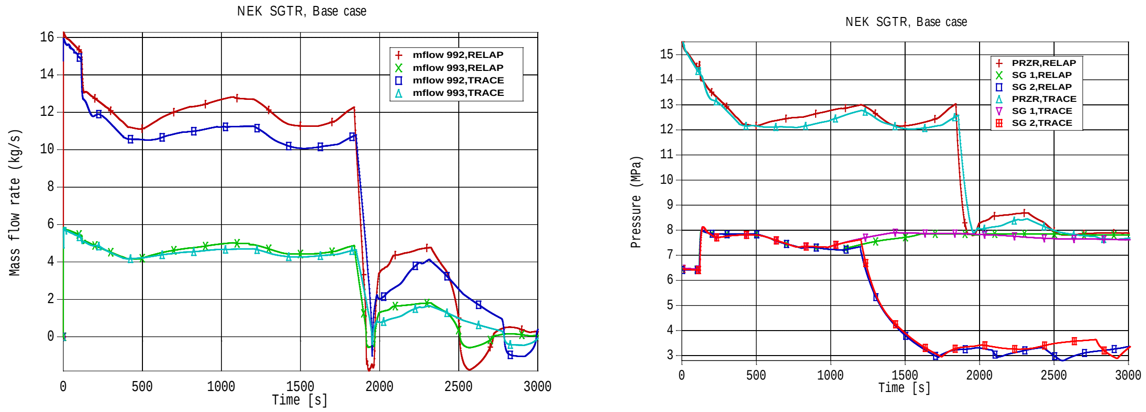

Figure 3: Break flow (left); pressurizer and SG pressure (right)

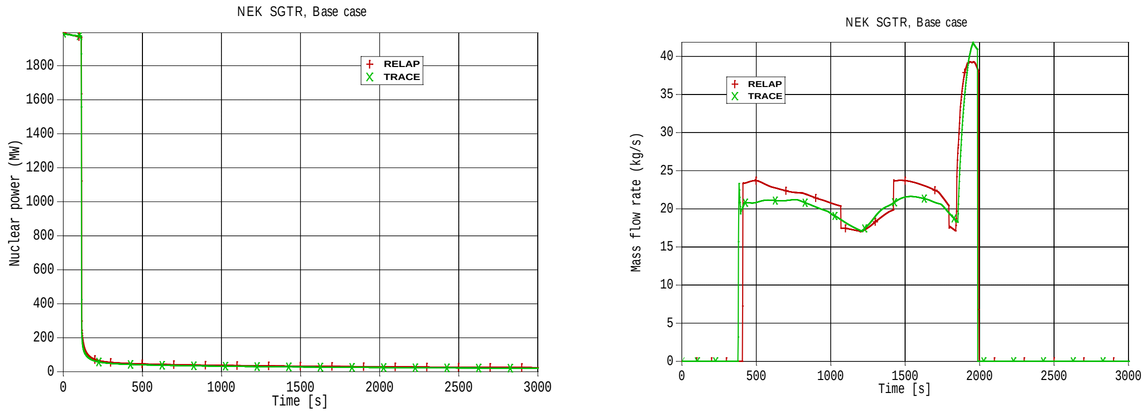

Figure 4: Nuclear power (left); total SI flow (right)

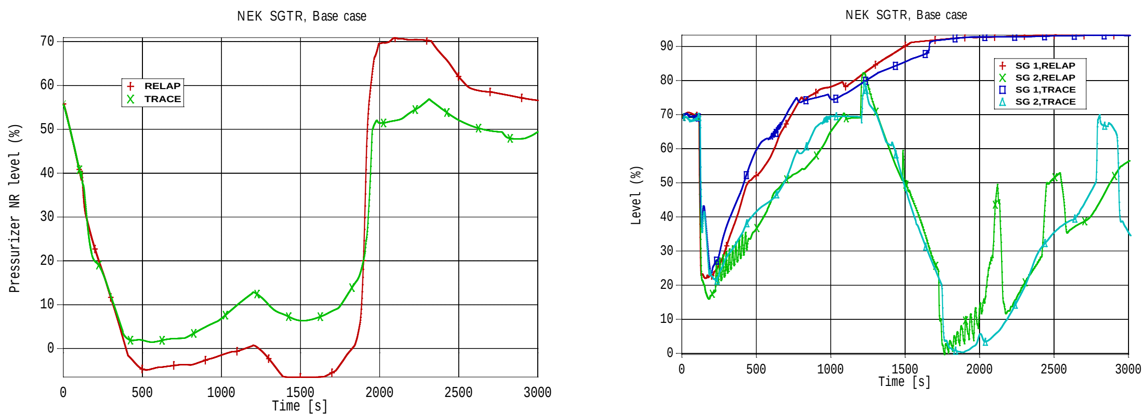

Figure 5: Pressurizer NR level (left); SG NR level (right)

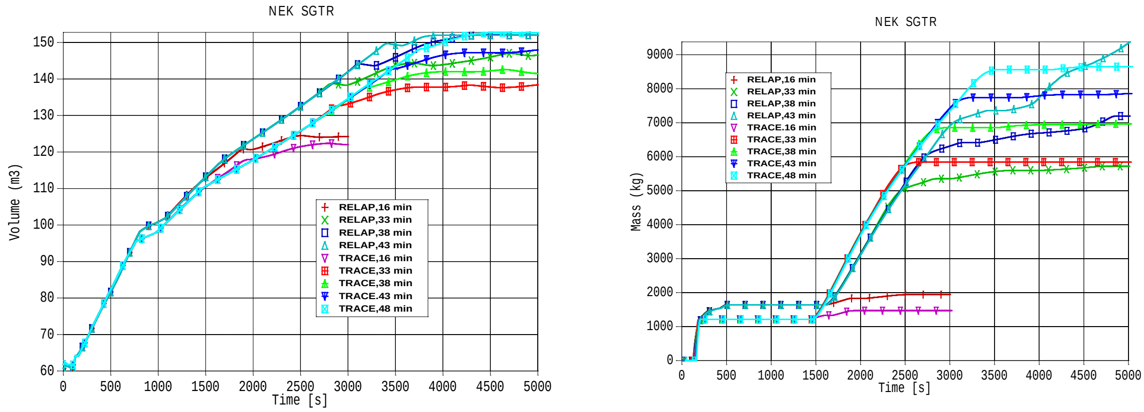

Figure 6: SG 1 liquid volume (left); discharged mass through SG 1 PORV (right)

Table 2: Maximum liquid volume in ruptured SG, maximum SG 1 PORV liquid fraction and discharged mass through SG PORVs

Delay	Max. SG 1 liquid volume	Max. SG 1 PORV liquid fraction (-)	Discharged mass through SG 1/2 PORVs (kg)
16 minutes (0-3000 s)	RELAP: 124.4 m3 TRACE: 122.4 m^3	RELAP: 0.012 TRACE: 0.011	RELAP: 1941.3/36984 TRACE: 1470.4/37890
33 minutes (0-5000 s)	RELAP: 147.1 m^3 TRACE: 138.4 m^3	RELAP: 0.012 TRACE: 0.011	RELAP: 5717.6/46928 TRACE: 5839.9/44305
38 minutes (0-5000 s)	RELAP: 152.0 m^3 TRACE: 142.6 m^3	RELAP: 0.26 TRACE: 0.011	RELAP:7195.7/45706 TRACE: 6966.5/45340
43 minutes (0-5000 s)	RELAP: 152.1 m^3 TRACE: 148.0 m^3	RELAP: 0.36 TRACE: 0.011	RELAP: 9380.0/36993 TRACE: 7853.3/36390
48 minutes (0-5000 s)	RELAP: 152.1 m^3 TRACE: 152.7 m^3	RELAP: 0.84 TRACE: 0.09	RELAP: 13707.8/36472 TRACE: 8685.2/35220

4 CONCLUSION

The Steam Generator Tube Rupture (SGTR) accident was analyzed with operator actions that were adopted from plant procedure for this event. Conservatively, it was assumed that auxiliary feedwater flow started immediately after reactor trip. The operator actions were aimed to isolate the ruptured SG and to stop the primary-to-secondary leakage thus preventing the overfill of damaged SG and the release of radioactivity to environment.

For the base case scenario (delay for the first operator action equal to 16 minutes) a good qualitative agreement of transient results for RELAP5 and TRACE was obtained and the main trends were well reproduced by both codes. The primary-to-secondary leakage was stopped (leakage flow reversed for the first time) around 32 minutes after transient begin for both codes. After terminating SI flow operator continued to maintain the plant at stable conditions by controlling the core exit temperature using intact SG PORV and secondary side heat sink by controlling the AFW flow to intact SG. Forty minutes after transient begin the secondary side liquid volume stabilized and discharge through the ruptured SG has stopped. The maximum liquid volume at the end of simulation (3000 s) were 124.4 m^3 for RELAP5 and 122.4 m^3 for TRACE that is well below the total SG volume (152.7 m^3). The small amount of steam was discharged in the base case (1941.3 kg in RELAP5 and 1470.4 kg in TRACE).

The additional analyses were performed in order to determine the delay for the first operator action resulting in ruptured SG liquid solid condition and liquid discharge through the ruptured SG PORV, i.e., 33, 38, 43 and 48 minutes delay, respectively. The first operator action should start not later than 33 minutes after reactor trip for RELAP5 and not later than 43 minutes for TRACE to prevent the liquid discharge (liquid fraction=10%) through the ruptured SG PORV. The analyses have shown that more adverse results were obtained for RELAP5 than for TRACE, but the differences are small. In general, a somewhat larger break flow in RELAP5 than in TRACE was obtained and it has led to larger ruptured SG inventory in RELAP5 after AFW termination.

The radiological consequences analysis was not performed, but based on limited amount of discharged fluid, they should be small.

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