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STUDY OF OPERATOR ACTIONS IN THE EVENT OF FAILURE TO CLOSE THE BRU-A IN THE EVENT OF A LARGE NUMBER OF SG TUBE RUPTURES

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In this paper, a large loss of coolant is discussed from primary circuit to the secondary side in the horizontal steam generator (SG) type in the Nuclear power plant (NPP). The study continues the work connected with investigation of NPP behaviour of primary to secondary (PRISE) side large leakage of coolant including the operator actions. The analysis validates the applicability of NPP strategy's "bleed and feed" using the symptom-based emergency operator procedures (EOPs). The study discusses the results of the analysis of operator actions in the event of a large number of guillotine break of SG pipes leading to the opening of the steam dump control valve to the atmosphere (BRU-A). As input data for this analysis, previously published results for such an event were used, in which the guillotine double ended rupture of the 270 steam generator tubes leads to a rapid increase in pressure on the secondary side of the steam generator to the set opening point of the steam release through BRU-A. Assuming a failure of BRU-A to close, all emergency cooling water for flooding the reactor core will be loose in a very short time. The purpose of the presented analysis is to demonstrate the possibility of significantly delaying this loss of emergency water for providing time for the implementation of actions that can ensure the restoration of the plant's operability or to ensure its long-term maintenance in a safe state. The analysis is performed with an integral computer code RELAP5 mod3.3 using the input model for NPP with a VVER 1000 reactor developed at Institute for Nuclear Research and Nuclear Energy (INRNE-BAS). The referent NPP is the Kozloduy NPP equipped with a VVER 1000 / V320 reactor types.

Keywords: PRISE, Loss of Coolant, RELAP5, SB EOPs, Reactor Core Heat up

1. Introduction

Nevertheless, that steam generator tube ruptures are widely investigated (Nematollahi et al., 2008; Groudev et al., 2004), there are some situations connected with this event type that should be analyzed. One of the most dangerous situations is a simultaneous guillotine rupture of certain number of steam generator (SG) tubes, which could cause uncontrolled and irreversible loss of primary circuit coolant due to opening of BRU-A.

A similar situation could be caused by an earthquake. Nevertheless, that possible events are with a very small probability, the consequences could be very severe. The SGs are mounted on special platforms preventing such situation, but due to severity of consequences of such event and due to different magnitudes and specifics of earthquakes these types of events should be analyzed. Such event will cause rapidly decreasing of primary circuit pressure and rapidly increasing of secondary side pressure in damaged SG.
There is a study (Vryashkova et al., 2025) that demonstrates rupturing of a relatively small number of 2.45% of the existing SG tubes in one SG, that causes uncontrolled increasing of secondary side pressure in the damaged SG leading to the opening of BRU-A even when the other existing control valves as steam dump to the condenser valves (BRU-Ks) are also activated. Such an accident will lead to an uncontrolled release of radioactive primary circuit coolant directly to the environment and losing in such way the capability, after some time, to cool down usefully the reactor core.
The presented paper discussed the accident progression of mentioned above event initiated by guillotine double ended breaks of 270 SG tubes in the cold leg of SG#1. In this analysis are included operator actions, which are based on the using of an existing Emergency Operating Procedures (EOPs) (Andreeva et al., 2015; Petkov et al., 1999).
The break flow rate depends on the location of ruptured tubes. In the investigation the presented results should be considered only as a representative since variations of locations of damages tubes and the conditions of operable equipment could result on estimation of exact minimum of SG tube numbers that will cause reaching a set point of opening pressure of regulating valve BRU-A. As the total number of ruptured pipes is very small compare to the total number of 11 000 tubes in one SG, we assume that the important point in the performed analysis is that the rupture of small certain number of SG tubes will cause opening of BRU-A. If BRU-A is opened it should be investigated possible situation when the valve is stuck in open position.

2. Nodalization of the VVER 1000 Model for Computer Code RELAP5

The reference nuclear power plant for this analysis is generic VVER 1000 V320 NPP. This NPP is a typical pressurized water reactor equipped with a four horizontal SGs (Pavlova et al., 2015; Verrier et al., 2021). RELAP5/MOD3.3 computer code has been used to simulate the selected transient for VVER-1000/V320 NPP.
The RELAP5 code is widely validate for VVER 440 and VVER 1000 (Groudev et al., 2006; Pavlova et al., 2006). The RELAP5 is widely used for study of different NPP behaviour (Groudev et al., 2015).
The integral input model with four independent loops has been developed at INRNE-BAS (Groudev et al., 2001; Andreeva et al., 2012) for analyses of different operational occurrences, abnormal events, and design basis scenarios including some beyond design accidents such as station blackout (SBO). In the model are included all control and safety systems (SSs).
On Figure 1 and Figure 2 are presented the nodalization schemes of SG#1 and steam lines for RELAP5 integral VVER1000 model.

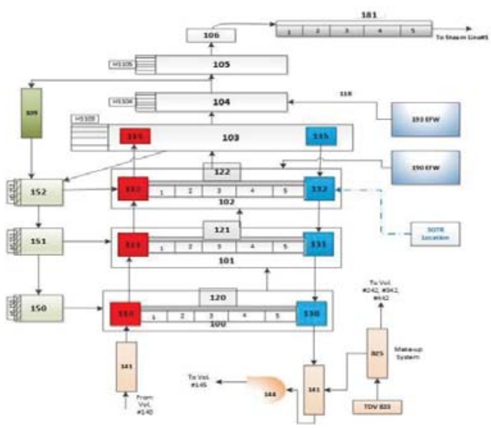

Fig. 1 Steam Generator nodalization

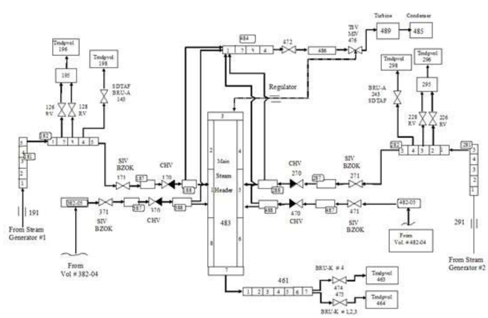

Fig. 2 Steam line nodalization for VVER 1000

3. Event Description and Assumptions

The event primary to secondary leakage is a specific accident which request special attention. If there is one or just few SG tubes breaks and loss of primary circuit coolant to the secondary could be compensated by Make-up/Let down system (TK system) there is a need of operator intervention due to slow progress of the accident, not activation of emergency shutdown of reactor (SCRAM), which led to significant contamination of the secondary side with a radioactive water. In the event of a rupture of 270 or more SG tubes, the operator actions are necessary to prevent rapid loss of all existing emergency water inside of the containment, exposure of the active zone and its further destruction.

If happened guillotine rupture of certain number of SG tubes the behaviour of primary circuit will be similar as a large loss of coolant due to a break in a primary circuit equipment somewhere. The secondary side of damaged SG will have different behaviour, characterized with rapidly increasing of pressure and water level.

Scenario of the of SG guillotine tube ruptures

The main assumptions for the investigated scenario are:

The initial conditions of NPP correspond to the end of life (EOL).

The reactor power is at 100%.

The emergency borated water in this analyses is 3 tanks of 15 m^3 with high concentration of boric acid of 39 g/kg and one command tank of 750 m^3 with 16 g/kg boric acid.

The total volume of emergency borated water is 795 m^3.

The temperature of emergency borated water is 30 $^{\circ}$C.

Additionally, there are 4 hydro accumulators with a total volume of 220 m^3. It is assumed in the analyses that 200 m^3 are injected. Also, in the analyses are injected 795 m^3 from active safety system and 200 m^3 from hydro accumulators (HAs).

The water collected in the sumps and returned to the tank is not used in this analysis. This allows to be compared with a previous analysis (Vryashkova, et al., 2025). In this way the presented analysis is more conservative.

Also, in the analysis it is assumed 5 sec. delay in the delivery of emergency borated water when the SSs start automatically.

The other important assumption is that the operator will realize the nature of the event in approximately 180 sec. when it is assumed that he performs his actions how it is described below. This assumption is based on a discussion with operators of Kozloduy NPP.

To facilitate the total loss of coolant it is assumed availability of all 3 SSs.

The initiating event is postulated as 270 guillotine double-ended tube ruptures in SG#1 on cold header side in the upper part of tube bundles.

Before intervention of operators in the control room the expected list of events is:

1. Reactor SCRAM occurs due to the signal "P_1<150 kgf/cm^2 and N>75%" with 0.4 sec. delay. The control rods fall in 2-4 sec. to the reactor core bottom.
2. The turbine is isolated by close of main isolated steam valve (MISV) in 10 sec after the reactor scram or by reaching 2.65 m water level in the SG#1.
3. All four BRU-Ks opens 1.0 sec. after closing of the MISV and start to maintains the secondary pressure to be the same as the pressure before closure of MISV.
4. Due to lose of primary subcooling of 10°C (Δt_S<10°C), there are activation of the following systems:
   a) The Containment Localizing Valves close.
   b) The Make-up/Let down system stops.
   c) The main coolant pumps (MCPs) stop 15 sec after reaching Δt_S<10°C.
   d) Actuation of automatic step by step load (AASSL) system - all three safety systems with 9 pumps in total start to work in recirculation mode:
      1) all 3 spray pumps start to inject after increasing of containment pressure - P_{cont}>0.2 kgf/cm^2.
      2) all 3 high pressure pumps (HPPs) start to inject after the depressurization of primary circuit to the set point of P_I<110 kgf/cm^2.
      3) all 3 LPPs start to inject after drop to the set point of P_I<26 kgf/cm^2.
5. Due to primary circuit depressurization to the set point of P_I<60 kgf/cm^2 all four HAs start to inject borated water to the reactor vessel. After injection of their entire volume of borated water

they isolated. We assume to be isolated when they reach 5 t in every one.

6. Due to the loss of primary coolant to the secondary side the Pressurizer (PRZ) water level drops and when it reaches level of 4.2 m the Pressurizer Heaters are switched off.

7. The operators isolated all injection of borated water into the primary circuits by active system at 180 sec.

8. Closing the BZOK for isolating by steam the damaged SG #1.

9. Isolating damaged SG #1 by feed water.

10. The operators start to cooldown the reactor core when the all HAs been almost depleted by one of the low-pressure pumps (LPP) at 800 sec. It was used the LPP TQ12. The emergency water was injected in the cold leg.

The operator reduces the flow rate of pump to the level not allowing core exit temperature to be above 450 K. In this way the flow rate used by the operator to cool down reactor core was 15 kg/s initially and was reduced slowly to 6 kg/s in the first 11 000 sec. After that the flow rate is between 5 and 6 kg/s supporting the core exit temperature around 385.0 K.

11. The operator reduce primary circuit pressure by opening two of pressurizer safety valves (PRZ SVs) at 180 sec.

4. Analysis and Discussion of the Results

After the initiating the event the behaviour of primary circuit is similar as large break loss of coolant accident. The primary circuit pressure and PRZ water level reduced rapidly. The investigated transient belongs to the transients with loss-of-flow transients (Grudev et al. 2004). Nevertheless, that the command number of the braked SG tubes are only 2.45% of the total number of tubes in SG#1, the command break flow rate from primary circuit causes rapid increasing of secondary side pressure in the faulted SG and pressure reaches the set point of opening of BRU-A on the steam line of this SG.

The calculated sequence of events for the SG tubes rupture are presented in Table 1.

Table 1. Sequence events for the SG tubes rupture

Events	Time (sec.)
Initiating the Break	0.0
Initiating the signal SCRAM	1.0
Max. break flow rate,1738.0 kg/s	1.1
Opening of BRU-A #1	9.0
Isolating of Turbine due to water level in SG#1 reaches 2.65 m	10.0
Switching off pressurizer heaters due to reducing water level to 4.2 m	10.3
Isolating the TK system.	10.0
Activation of all BRU-Ks	11.0
High pressure pumps start to inject	11.0
Switching off all MCPs	20.0
Activation of HAs#2 and 4 at 60 kgf/cm^2 (lower plenum)	60.0
Activation of HAs#1 and 3 at 60 kgf/cm^2 (upper plenum)	70.0
BRU-K #1, #2, #3 and #4 closed	80.0
All high- and low-pressure pumps are stopped by the operator.	180.0
Closing the BZOK on damaged SG #1	180.0
Isolating the SG feed water to SG#1	180.0
The operator starts to inject emergency water with a LPP	800.0
HAs #2 and 4 are disconnected	830.0
HAs #1 and 3 are disconnected	860.0
Low Pressure Pump TQ12 stops due to depletion of emergency water	132 700.
Core exit temperature reaches 1200 K	1
End of calculation	1

As seen on Figure 3 the primary pressure is reducing very fast to the pressure of secondary side in faulted SG. On Figure 4 is presented main steam header pressure.

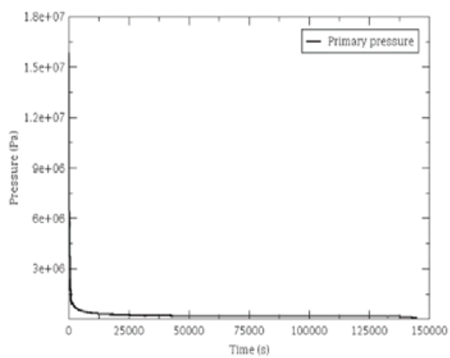

Fig. 3 Primary pressure

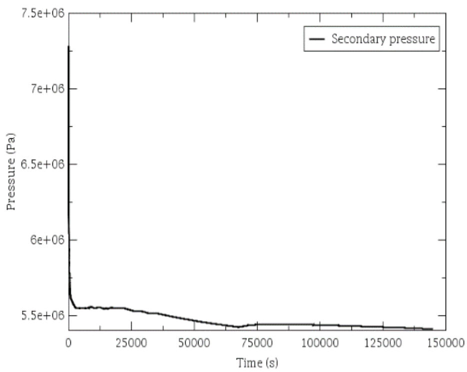

Fig. 4 Main steam header pressure

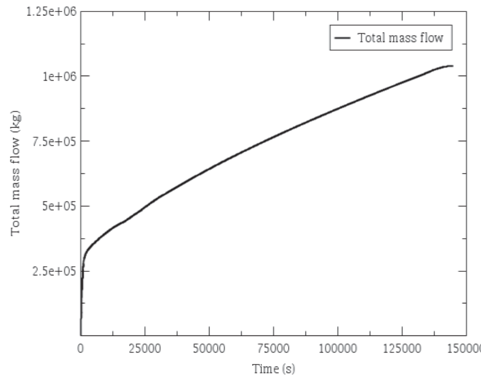

Fig. 7 Integral break flow rate

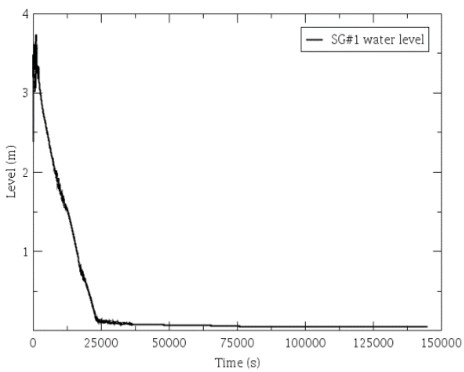

Fig. 5 SG#1 damaged water level

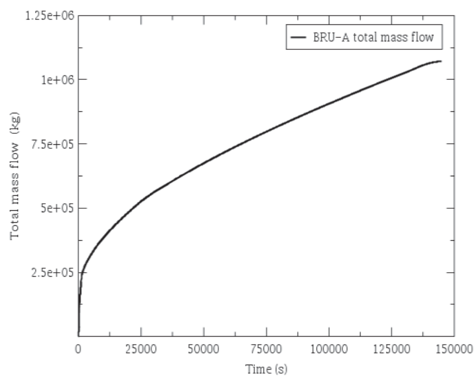

Fig. 8 Total loss of water through BRU-A

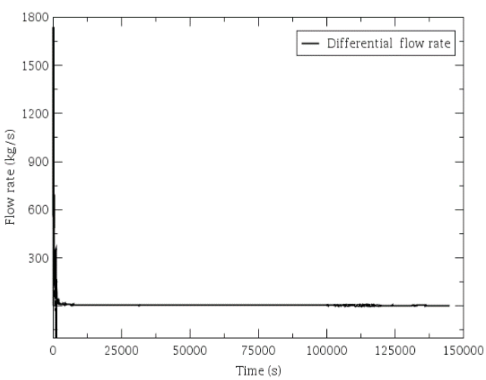

Fig. 6 Differential break flow rate

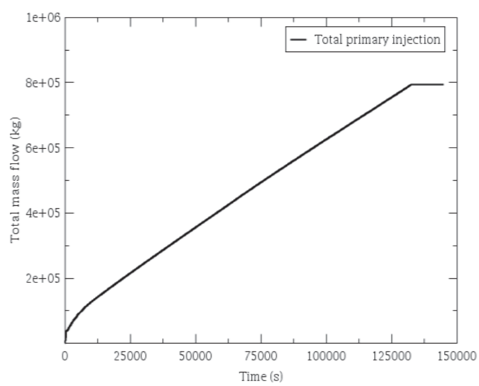

Fig. 9 Total primary injection

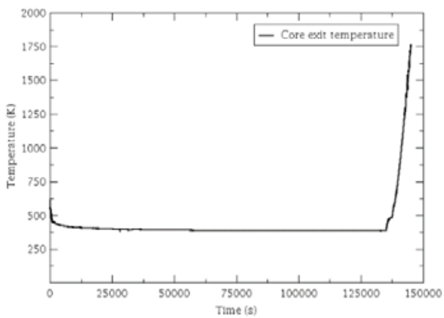

Fig. 10 Core exit temperature

The primary side and the secondary side, practically are directly connected, due to the break of 270 SG tubes.

With opening of two PRZ SVs the primary circuit is connected with the containment. This action allows to preserve water inside of the containment by collecting the water through the sump in the containment bottom to the command borated tank. During the end of transient 143 000 sec. is collected 141 t of water from SVs. The value of this water will depend of how much water will be injected in primary circuit.

Initially the main steam header pressure is stabilized due to work of BRU-K, but after that slowly start to reduce due to cool down the primary circuit by uncontrolled steam dump to the atmosphere through the BRU-A on SG#1.

On Figure 5 is presented water level in SG#1. Initially, the water in SG#1 increases up to 3.8 m due to the break flow rate, but after isolating the SG#1 dry out due to assumption of opening and stuck in open position of BRU-A. In this way the primary circuit is directly connected to the environment.

On Figure 6 is presented the differential break flow rate. After the initiating the event the break flow rate reaches maximum flow rate of around 1738.0 kg/s. After fast depressurization of primary circuit and increasing of the pressure in faulted SG the break flow rate is reduced significantly to the almost 130 kg/s and continue with this flow rate to the end of calculation.

On Figure 7 is presented total loss of water through the BRU-A to the environment. In this analysis the total loss of radioactive water to the environment is around 1 072 t at the end of transient. The release of radioactive water depends strongly from the injected emergency water. The amount is bigger than the injected water as it includes also and water from faulted SG and coolant from primary circuit. In the first few hundred second, while primary pressure and respectively the secondary side pressure in damaged SG and steam line a significantly high the flow rate is increasing faster. At 160 sec. the BRU-A flow rate reaches maximum value of around 650 kg/s, and after that rapidly is reduced to 8 kg/s at 2000 sec. and slowly is reduced to the 4 kg/s to the end of calculation. The reason to be so small is that the flow rate of LPP is close to this value.

On Figure 8 is presented total primary injections only from active safety systems. In this figure is not included borated water that is injected by all 4 HAs. After depletion of emergency borated water in 132 700 sec. there is observe heat up of the reactor core and the core exit temperature reaches 1200 K at 142 300 sec. In the analyses is not used this option just to compare the time for heat up delay in (Vryashkova et al., 2025). Also, there is a possibility of failure of sumps, which is not discussed in this paper.

5. Conclusions

The presented work shows that the operator intervention during PRISE events with an assuming the opening and not closing of BRU-A, is mandatory to preserve the emergency cooling water as longer as possible.

Without operator actions the dryout of the reactor core and heat up to the 1200 K is observed at approximately 12 000 sec. (Vryashkova, et al., 2025), while with an operator intervention the core exit temperature reaches 1200 K at approximately 142 300 sec. The time will increase additionally if will be used water collected by sumps on the bottom of containment. This time is sufficient to eliminate the problem with BRU-A or to deliver borated water from outside of the containment.

The reducing of flow rate of emergency borated water by the operator significantly reduce the velocity of release the radioactive water into the environment. After fixing the problem with a BRU-A the release of radioactive water will be stopped immediately.

With reducing the flow rate of SSs (in this case LPP (TQ12)) is eliminated risk of challenging the critical safety function (CSF) “Integrity” by avoiding rapidly cooldown of reactor pressure vessel.

The water passing through the PRZ SVs is not enough to be organized “bleed and feed” procedure, but it will allow to extend significantly time for safe supporting of reactor core cooldown and further resolving problem with a long term reactor core cooldown.

It was also investigated the place (in hot or in cold legs) of safety system injection. It was concluded that this is not so important.

Based on the analysis, it could be concluded that the operator’s actions significantly delay the heat up of the reactor core, but is not able to resolve the problem with ensuring a long-term cooling of the reactor core. The successful long-term cooldown of reactor core could be guarantee only with providing emergency borated water by external sources. Nevertheless, the operator actions ensure at least one and half day time to performing additional actions, which will resolve this issue.

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