ИЗВЪРШВАНЕ НА ПРЕДВАРИТЕЛНИ ПРЕСМЯТАНИЯ С КОМПЮТЪРЕН КОД RELAP5/MOD3.3 НА АВАРИЯ РАЗКЪСВАНЕ НА ГЛАВНА ПАРОВА ЛИНИЯ НА БАЗАТА НА OECD БЕНЧМАРК ЗА ВВЕР1000 НА АЕЦ КОЗЛОДУЙ

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PRELIMINARY RELAP5/MOD3.3 CALCULATION OF MAIN STEAM LINE BREAK ACCIDENT BASED ON OECD BENCHMAK FOR VVER1000 OF KOZLODUY NPP

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Abstract

This paper presents the results from simulation of „Main steam line break“ accident using RELAP5/mod3.3 addressed to VVER-1000 reactor type. The work is based on OECD VVER-1000 MSLB benchmark.
The main objectives of performed work is to provide elements for the safety, validated safety analysis tools, also for improvement of codes and methods for VVER comprehensive safety assessment and to support VVER fuel development and qualification activities.
The investigated scenario is main steam line break in a VVER-1000 between the steam generator (SG) and the steam isolation valve (SIV), outside the containment. This event is characterized by significant space-time effects in the core caused by asymmetric cooling and assumed stuck-out control rods after

scram. One of the major concerns for this case is the possible return to power and criticality after scram, due to overcooling.

1. Introduction

The transient is initiated by a main steam line break in a VVER-1000 between the steam generator (SG) and the steam isolation valve (SIV), outside the containment.

This event is characterised by large asymmetric cooling of the core and large primary coolant flow variations. One of the major concerns for this case is the possible return to power and criticality after reactor scram, due to overcooling. The scenario is based on conservative assumptions that maximise the consequences for a return to criticality. The break is assumed to occur in MSL-4 with ID (inside diameter) 580 mm. Following the break it appear the scram signal based on signal. The MCP of the faulted loop trips to mitigate the overcooling, with a coast down time of 55 s. The vessel mixing pattern makes a transition to reversed flow in one loop and three MCP running normally during the transient.

A mechanical failure of the large feed water-regulating valve in the broken line is assumed. At the time of the steam line rupture the valve starts to open from about 70% to 100% and then remains stuck in the open position. The main feed water flow to the faulted SG terminated by closure of the feed water block valve in 52 s. Steam isolation valve #4 starts to close and the check valve in the broken line closes to isolate the MSH from the break. Turbine stop valves close on protection signal: 10s after scram.

2. Brief description of RELAP5/MOD 3.3 VVER1000 model

The Baseline input deck for VVER-1000/V320 Kozloduy Nuclear Power Plant, Unit 6 was developed by the INRNE-BAS.

The Kozloduy VVER-1000 RELAP5/MOD3.3 [1, 2] model was defined to include all major systems of Kozloduy NPP, Unit 6, namely: reactor core, reactor vessel, Main Coolant Pumps (MCPs), SGs, steam lines and Main Steam Header (MSH), emergency protection system, pressure control system of the primary circuit, makeup system, safety injection system, steam dumping devices (BRU-K, BRU-A, SG and pressurizer safety valves), and main feedwater system.

In the RELAP5 model of VVER-1000, the primary system has been modelled using four coolant loops each one including one MCP and horizontal SG. A hot and average heated flow paths and a core bypass channel represent the reactor core region. The reactor vessel model includes a downcomer, lower plenum, and outlet plenum. The Pressurizer (PRZ) system includes heaters, spray and pressurizer relief valves. The safety system representation includes accumulators, high- and low-pressure injection systems, and reactor scram system. The model of the make-up and blowdown systems includes associated control systems.

RELAP5 heat structure components are used to represent fuel rods, 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 to the secondary side is modelled by heat structure components.

The primary and secondary side modelling are shown schematically on Figures 1 and 2.

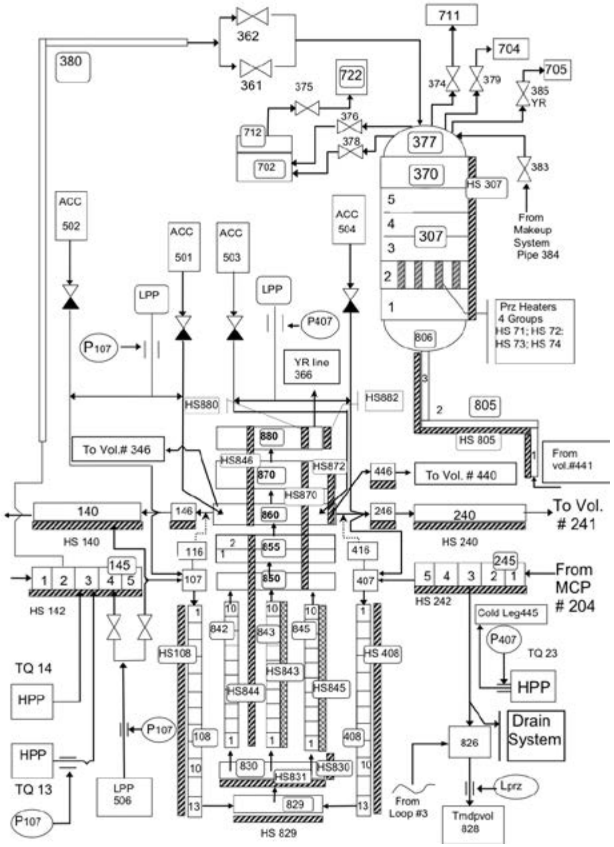

Figure 1 RELAP5/MOD3.3 VVER 1000 Primary side nodalization model

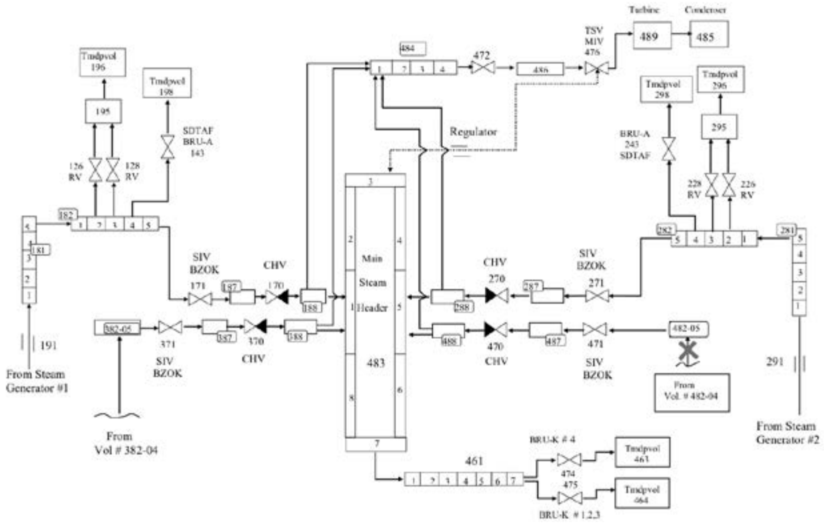

Figure 2 RELAP5/MOD3.3 VVER 1000 Secondary side nodalization model

3. Initial and boundary conditions:

- The reactor is at nominal conditions, and hot full power (HFP)
- 270.4 EFPD (29.90 MWd/kgU average core exposure);
- The used fuel loading is the end of 8-th campaign;
- 0.3g/kg boron acid concentration (53 ppm boron concentration);
- Equilibrium Xe and Sm concentrations.

4. Scenario:

MSLB transient description, summarizing the main assumptions:

1. Size break (ID:580 mm);
2. The activation of reactor SCRAM is based on the following signal: After SG#4 pressure rapidly drops in accordance with signal PSG < 50 kgf/cm^2 and dTs(I-II) > 75 °C.
3. MCP #4 coast down in 55 sec;
4. Make up and Let down systems are not activate;
5. BRU-K is activated for secondary side pressure regulation;

Feed water valve fails and remains open (additional FW into SG).

5. Results and discussions:

The calculated results are presented on Figures 3 to 10.

After the main steam line break is observed in loop #4, the initial steam flow increase rapidly, which lead to the overcooling of the primary system and accordingly, the temperature in unaffected loops and pressure decrease. The SCRAM signal is activated at 6 sec, due to signal P in SG < 50 kgf/cm^2 and dTs(I-II) > 75 °C. The MCP #4 coast down for app. 55 sec. Feed water valve fails and remains open (additional FW into SG).

After the reactor trip, the reactivity increases (see Figure 7) mainly due to the overcooling of primary coolant, from the other side the primary coolant density, also increase (see Figure 9). Furthermore, the fuel temperature reactivity coefficient has impact on reactivity behaviour. After steam line isolation the pressure in the non-affected SG rises and this SG does not represent the heat sink, but it heats-up the primary system, what is beneficial to the accident. After reaching the set points for secondary side regulation activation, the BRU-K open at 6.65 MPa and reduce the pressure, and close when the secondary pressure is app. 4.47 MPa (the close signal in not real set point, but it is accepted in OECD benchmark for BRU-SN). After BRU-K close the pressure increase slowly to 5.5 MPa until the end of transient (see Figure 10).

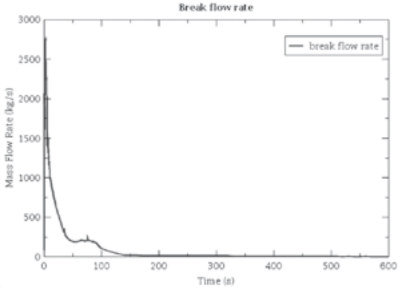

Figure 3 Total break flow rate

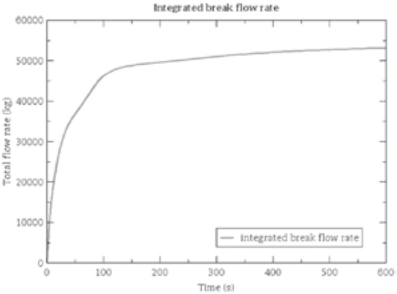

Figure 4 Integrated break flow rate

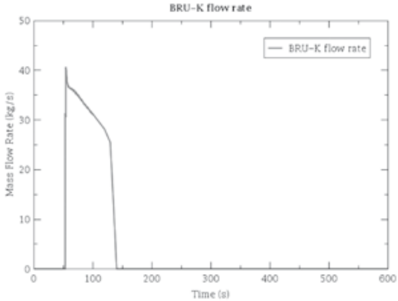

Figure 5 BRU-K flow rate

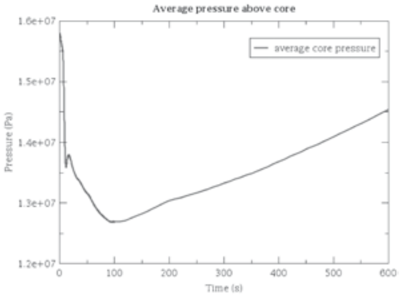

Figure 6 Average pressure above the core

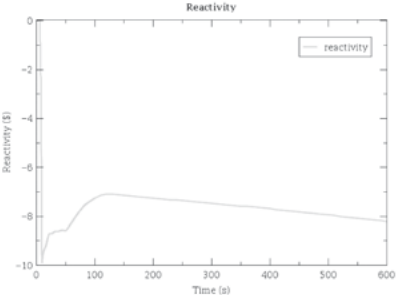

Figure 7 Reactivity

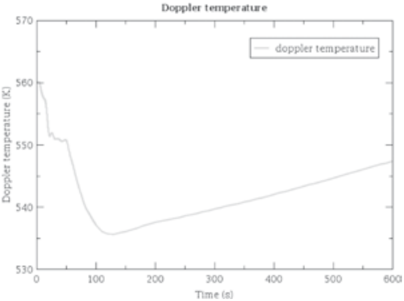

Figure 8 Doppler coolant temperature

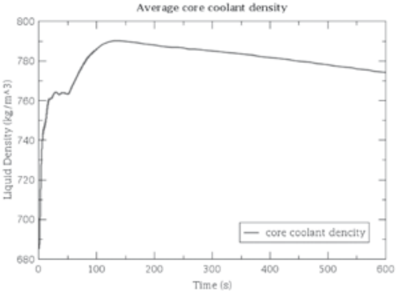

Figure 9 Average coolant density

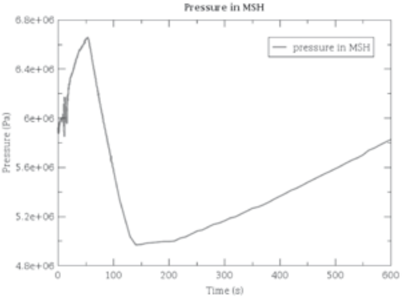

Figure 10 MSH pressure

After the water inventory from the faulted SG is lost, the temperature in the primary system start to increase slowly, which lead to reactivity decrease.

The decay heat is again transferred to the non-affected steam generator and this effect is superior in the cases with RCPs available what resulted in the slower reactivity decrease.

6. Conclusions

In this paper is discussed the results of thermal-hydraulic calculation of “Main Steam Line Break” analysis at full power reactor for VVER-1000/V320 units at KNPP. The performed analysis is based on a previously OECD “MSLB” Benchmark. Based on, it has been simulating MSLB accident using RELAP5/ MOD3.3.

The presented paper demonstrates the stability of the reactor system during the accident progression of a size break steam line break. The results demonstrate that the system is in the acceptable margin of cooldown rate.

7. References

[1] V. T. Berta, et. al., “RELAP5/MOD3.3 Code Manual”, NUREG/CR-5535/ Rev P5-Vol II App A (2016);

[2] International Atomic Energy Agency, Guidelines For Accident Analysis For VVER Nuclear Power Plants, VVER-SD-094, Vienna (1995);

[3] VVER-1000 Coolant Transient Benchmark [PHASE 2 (V1000CT-2) Vol. III: MSLB Problem – Final Specifications, NEA/NSC (2006);

[4] M. Pavlova, M. Andreeva, P. Groudev, Steam Line Break investigation at full power reactor forVVER-1000/V320, Nuclear Engineering and Design 285 (2015);

[5] Petya Ivanova Vryashkova, Pavlin Petkov Groudev, INVESTIGATION OF A STEAM LINE BREAK ACCIDENT IN RESPECT OF THE CRITICAL SAFETY FUNCTION “SUBCRITICALITY”, BAS papers, Comptes rendus de l’Acad_emie bulgare des Sciences, Tome 73, No 10, 2020.

8. ACKNOWLEDGEMENTS

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