Determination of the LB LOCA Ranges in the Containment for the Purposes of PSA Level 1 for KNPP

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Abstract. In this paper a short description for specific application of the RELAP5/MOD3.3 Patch 5 for Kozloduy NPP (KNPP) safety analyses is presented. More specifically, results from determination of the LB LOCA ranges in the containment are discussed. The analyses are part of the supporting activities related to the modelling of the accident sequences and determination of the success criteria for the considered groups of IE for the purposes of the update of PSA level 1 for KNPP units 5 and 6.

Keywords: KNPP, PSA, RELAP, LOCA, CD, PCT, containment, range, margin.

Nomenclature

CD: Core Damage

ECCS: Emergency Core Cooling System

FSIV: Fast Acting Steam Isolation Valve

HA: Hydroaccumulator

HPIP: High Pressure Injection Pump

LOCA: Loss of Coolant Accident

LPIP: Low Pressure Injection Pump

MCP: Main Coolant Pump

MSLB: Main Steam Line Break

NPP: Nuclear Power Plant

PCT: Peak Cladding Temperature

PRZ SRV: Pressuriser Safety Relief Valve

PSA: Probabilistic Safety Assessment

SCRAM: Safety Control Rod Axe Man (reactor shut-down)

SDA: Steam Dump to Atmosphere

SDC: Steam Dump to Condenser

SG: Steam Generator

1 Introduction

The NPP Kozloduy (in Bulgaria) is a 4-Loop reactor installation based on the Russian design VVER-1000 (V320). The commercial operation of the plant started in 1991 (for unit 6). The NPP is currently active with introducing innovations, backfit measures and the continuous demonstration of the safety standards in the light of a long-term operation.

The conduction of PSA level 1 is an important part of the state-of-the-art safety status of each contemporary NPP and is performed in periodic manner. The quality of the probabilistic assessments is highly dependent on the available deterministic analyses which are performed for various projects which are aimed at assessment of the plant safety, considering the up to date condition of the plant. On the other hand, during the development of the probabilistic models, various questions arise which require conduction of additional, specific thermo-hydraulic analyses. These analyses give the necessary clarifications related to specified accident sequences and the phenomena which occur during a certain accident progression. The determination of the LOCA ranges in the containment boundary is one of the fundamental tasks in the frame of PSA level 1 development. These ranges envelop: very small break (VSB), small break (SB), medium break (MB), and large break (LB) LOCA. In the current paper the results for the LB LOCA range which is subdivided into three sub-ranges are briefly presented.

For the analyses performed a RELAP5/MOD3.3 Patch 5 version with the following executable file (for Windows operating system) is used: **relap5-m33p5(km)-win32-ifc-opt-b2-snap.exe**.

For the calculations, a contemporary HP Z8 G4 Workstation with two CPUs is utilized (Intel (R) Xeon (R) Gold 6254 CPUs).

2 Short Description of the Plant Model

The plant model configuration includes the following main parts:

- Reactor vessel and internals.
- Reactor point kinetics.
- Four primary circulation loops with the respective MCPs and SGs primary side.
- Pressurizer and surge line.
- Primary chemical and boron control system.
- Emergency core cooling systems – active and passive part.

- Primary overpressure protection system.
- Emergency gas removal system.
- SG secondary side – SG vessels, steam tubes, and collectors.
- Four main steam lines and the main steam header.
- Main and auxiliary feedwater system.
- Emergency feedwater system.

The main nodalization schemes of the primary and secondary sides are shown in Figure 1 – Figure 4.

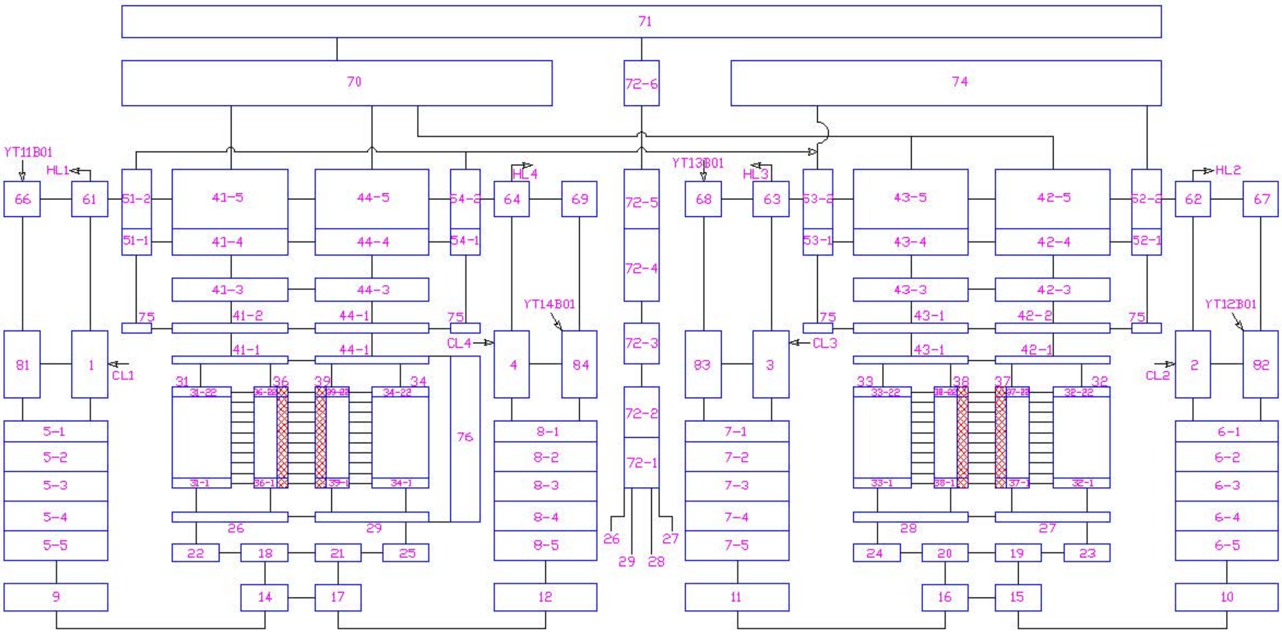


Figure 1. Nodalization scheme of the reactor pressure vessel.

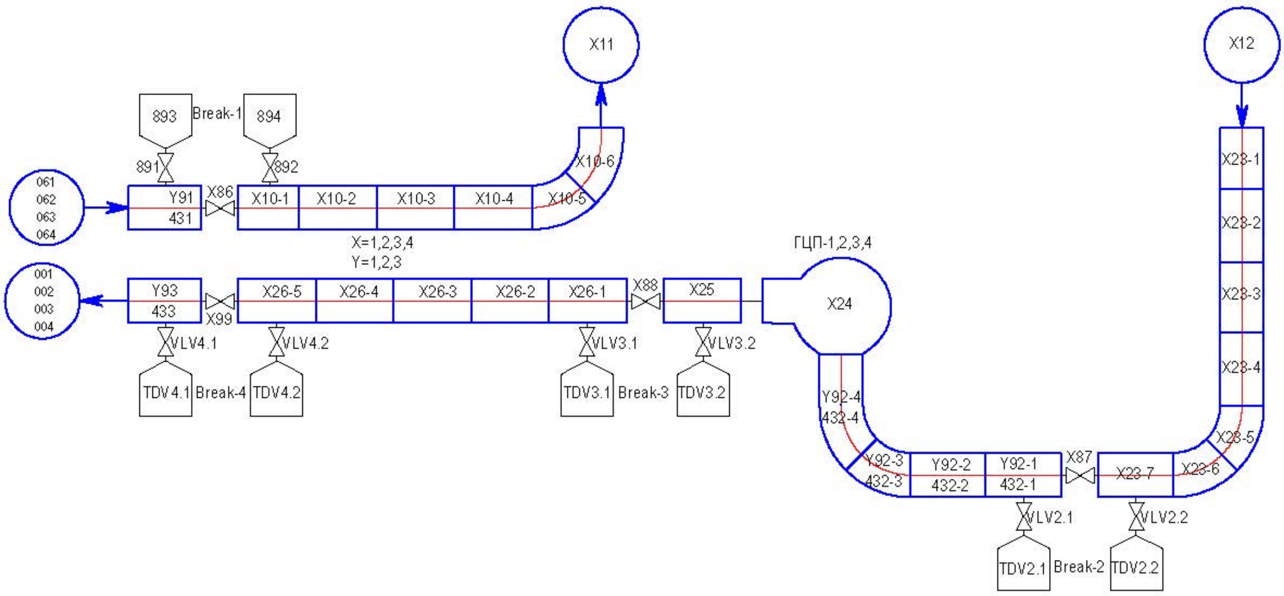


Figure 2. Nodalization scheme of the main circulation pipelines.

3 Verification and Validation of the Plant Model

The verification of the plant model is accomplished in the frame of various projects for KNPP. This verification is internal and is implemented by experts who developed the respective part of the model and by other experts who reviewed the model but did not played active role in the development process. This model has been used for miscellaneous safety analyses for more than 13 years. It has been updated and improved regularly during this period of time.

The validation of the plant model encompasses benchmark with independent plant model for RELAP5 (developed by different organization), comparison against operation tests and transients

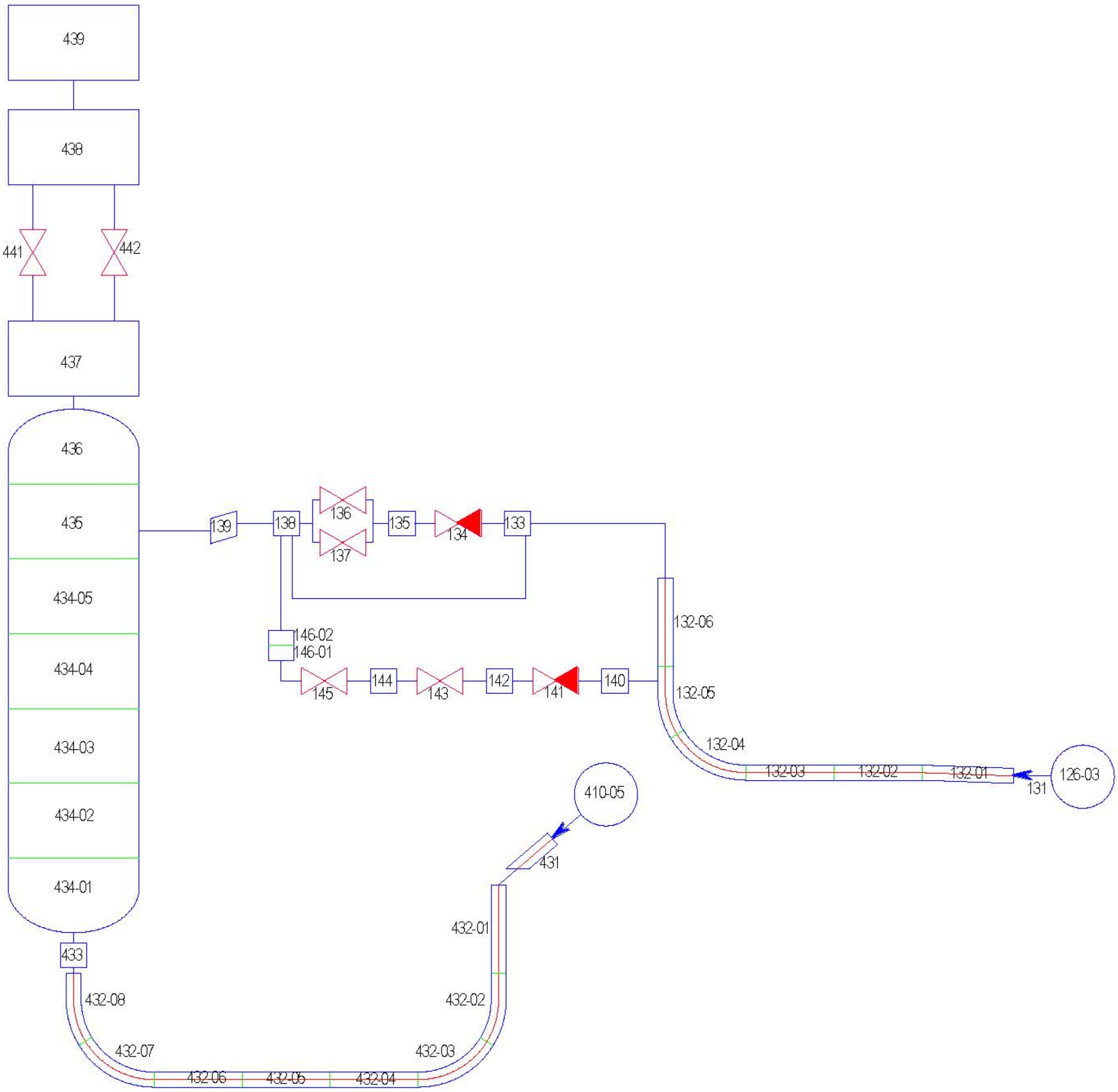


Figure 3. Nodalization scheme of the Pressurizer system.

from the real plant, comparison against experimental facilities for VVER and testing operation of the respective plant system against its dedicated function in the plant (to see if the model correctly represents this function).

The benchmark includes the following accident sequences:

- Inadvertent opening of a single PRZ SRV and its stuck in open position.
- Inadvertent closure of 1 out of 4 FSIV on the main seam line 4.
- Various independent analyses (with the plant model) against original Russian projects for KNPP.

The comparison against real plant operational test and transients includes:

- 1 out of 4 MCP trip. The comparison incorporates short-term (the first 30 min of the transient) and long-term period (the period after undertaking measures for the plant cooldown) of the transient.
- Turbine trip due to actuation of differential protections of the plant electrical generator.
- Test for establishing a stable natural circulation (this test is accomplished at the very initial start-up process of the KNPP unit 6).

The comparison against PSB-VVER experimental facility includes: Total Black Out accident. The aim is to represent the tendency of the parameters change over the transient (qualitative assessment).

All of the plant systems models are tested in real projects conditions and their retesting is not necessary for the current update of the plant model. Nevertheless, the logic for reactor power decrease which lead to partial load configuration of the plant is retested due to the changes which are implemented in the model (new setpoint are established in the real plant together

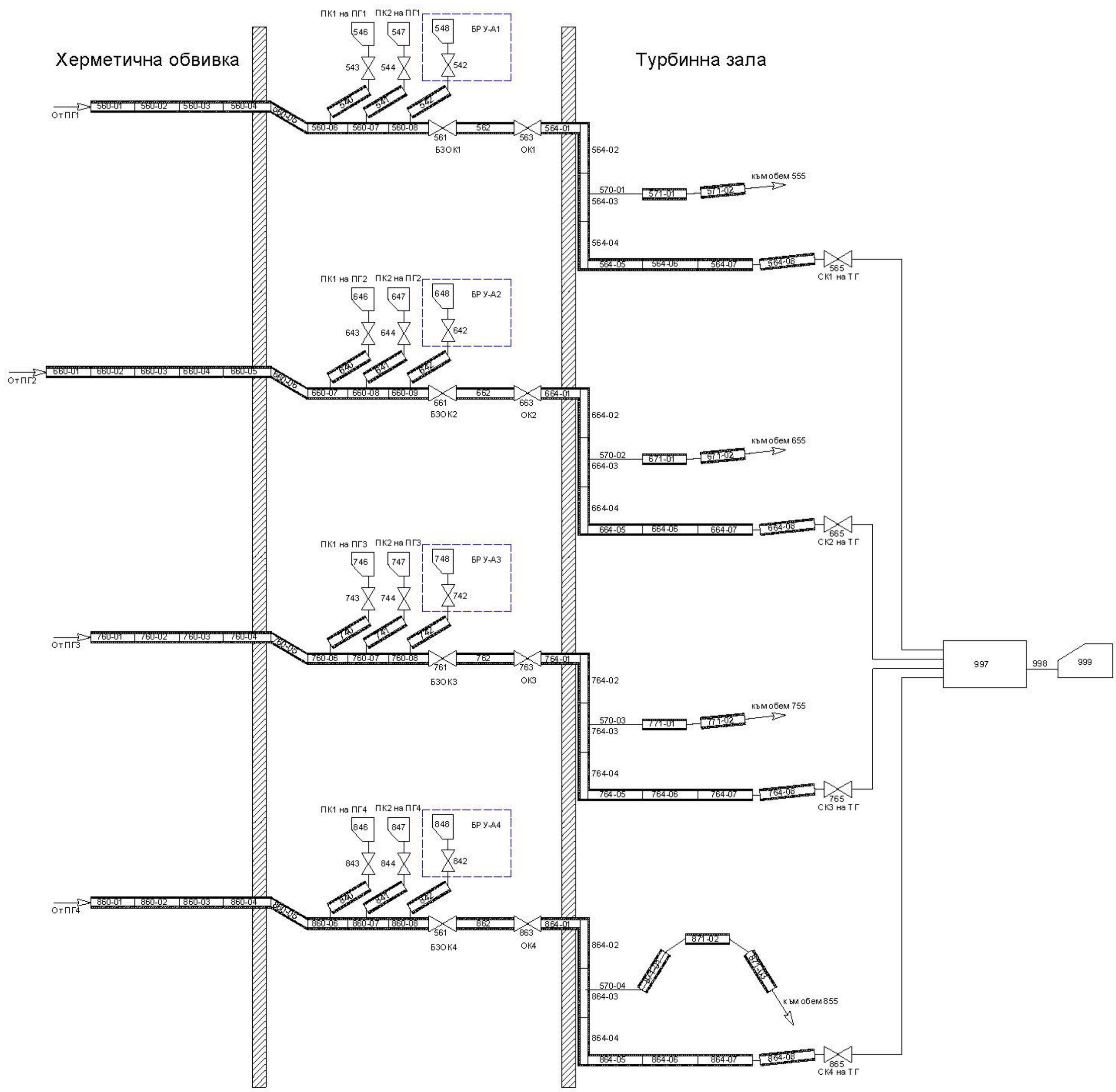


Figure 4. Nodalization scheme of the main steam lines.

with changed characteristics of the power regulation groups for the current fuel cycle). This includes:

- Trip of 1 out of 4 MCP – leads to reactor power decrease to 60% of the nominal power.
- Trip of 2 out of four MCP - leads to reactor power decrease to 41.6% of the nominal power.
- Turbine trip or 1 out of 2 main feedwater pumps trip – leads to reactor power decrease to 40% of the nominal power.

4 Definition of the LB LOCA Ranges

In the framework of the analyses for determining the range of LB LOCA from the primary circuit, the following three groups are considered:

- Group 1: LB LOCA for which the pressure in the primary circuit is reduced to values for stable operation of the LPIP, but at least 2 out of 4 hydroaccumulators (HA) are required to avoid damage to the core (core damage (CD)).
- Group 2: LB LOCA for which the pressure in the primary circuit decreases to values for stable operation of LPIP, but the PCT remains below 760°C without injection from any HA.
- Group 3: LB LOCA for which the pressure in the primary circuit decreases to values for stable operation of LPIP, but the PCT rises above 760°C. The possibility of core geometry deterioration without injection from HA increases significantly, i.e. the injection from 2 out of 4 HA is mandatory.

Definition: A LB LOCA is a leak for which the pressure in the primary circuit decreases very quickly and an effective supply of boron solution from the LPIS is achieved automatically, practically after a few minutes from the beginning of the accident process. The residual energy in the core is much less than the en-

Table 1. Scenarios considered for the respective groups

No	Scenario	Coals of the calculation
	Considered leaks for Group 1	
1.	**L1.G1-1-1**: LOCA with DN 70 mm close to the reactor vessel for the cold leg 1.	The conducted analyses are necessary for determination of the lower margin of LB LOCA (the lower margin for Group 1). The goal is to be determined whether in case of HPIPs failure and availability of 2 out of 4 HA, the pressure in the primary side will decrease to a value which allows stable operation of 1 out of 3 LPIPs.
2.	**L1.G1-1-2**: LOCA with DN 80 mm close to the reactor vessel for the cold leg 1.	
3.	**L1.G1-1-3**: LOCA with DN 110 mm close to the reactor vessel for the cold leg 1.	The conducted analyses are necessary for determination of the LB LOCA range in the containment boundary for which the injection from 2 out of 4 HA is required for avoiding CD, together with availability of 1 out of 3 LPIPs. Through these analyses the upper margin of the LB LOCA from Group 1 is determined and therefore the lower margin of the LOCAs from Group 2 is also specified.
4.	**L1.G1-1-4**: LOCA with DN 115 mm close to the reactor vessel for the cold leg 1.	
	Considered leaks for Group 2	
1.	**L1.G2-1-1**: LOCA with DN 370 mm close to the reactor vessel for the cold leg 1.	The conducted analyses are necessary for determination of the upper margin of the LB LOCA range in the containment boundary for which the injection from HA is not required, i.e. the maximum PCT remains below 800°C. The availability of 1 out of 3 LPIPs is considered.
2.	**L1.G2-1-1**: LOCA with DN 380 mm close to the reactor vessel for the cold leg 1.	
	Considered leaks for Group 3	
1.	**L1.G3-1-1**: LOCA with DN 2×850 mm close to the reactor vessel for the cold leg 1.	The conducted analyses are necessary for assessment of the maximum PCT when HA are not available and availability of 1 out of 3 LPIPs is considered.
2.	**L1.G3-1-2**: LOCA with DN 2×850 mm close to the reactor vessel for the cold leg 1. It is assumed that the coolant temperature in the containment sump is 90°C (the ECCS heat exchangers are not available).	
3.	**L1.G3-1-3**: LOCA with DN 2×850 mm close to the reactor vessel for the cold leg 1. It is considered a case with blockage of fuel rods bundle which consists of 37 fuel pins with 40% (see [1,2]).	
4.	**L1.G3-1-4**: LOCA with DN 2×850 mm close to the reactor vessel for the cold leg 1. It is considered a case with blockage of fuel rods bundle which consists of 37 fuel pins with 40 % (see [1,2]). This bundle is isolated from the remaining part of the core (cross flow connections are not considered).	
5.	**L1.G3-1-5**: LOCA with DN 2×850 mm close to the reactor vessel for the cold leg 1 and availability of 2 out of 4 HA.	Sensitivity analysis for Scenario **L1.G3-1-1** with availability of 2 out of 4 HA.
6.	**L1.G3-1-6**: LOCA with DN 2×850 mm close to the reactor vessel for the cold leg 1 and availability of 4 out of 4 HA.	Sensitivity analysis for Scenario **L1.G3-1-1** with availability of 4 out of 4 HA.

ergy dissipated by the leakage and the operation of the secondary circuit systems is not important for the cooling of the core (the cooling via the secondary side is not credited). Description of the considered accident scenarios is presented in Table 1.

5 Initial and Boundary Conditions

The initial conditions for the analyses performed are presented in Table 2. The boundary conditions for the LOCA analyses are as follows:

- Break is considered close to the RPV vessel for the cold legs for circulation loop 1 (sensitivity analyses show that the results for different circulation loops are similar).
- It is assumed ATWS.
- Primary chemical and boron control system is not available.
- All HPIP are failed.
- 1 out of 3 LPIP is operable.
- 2 out of 4 or 4 out of 4 hydroaccumulators are available.
- Main, auxiliary, and emergency feedwater systems are not available.
- SDC and SDA are not available.
- Operator actions for the accident management are not credited.

Table 2. Initial conditions

Parameter	Dimen-sion	Refer-ence	In the plant model
Reactor thermal power	MW	3120	3120
Pressure at the reactor outlet	MPa	15.69	15.69
Reactor inlet temperature	°C	290	289
Reactor outlet temperature	°C	321	320
Total mass flow through the core	m^3/h	87300	87303
Pressurizer level	m	8.77	8.77
Pressure in SG	MPa	6.27	6.16
SG level	m	2.40	2.40

6 Acceptance Criteria

The applicable acceptance criteria are as follows:

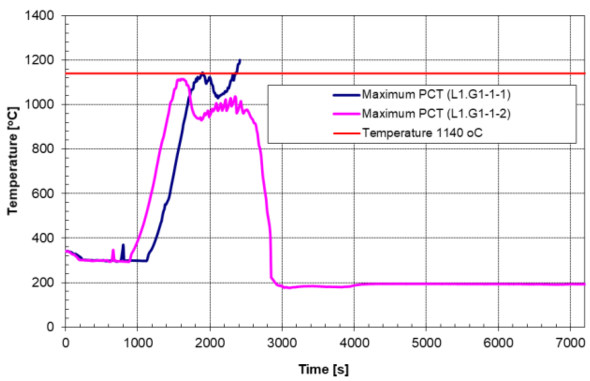
- Criterion 1: Maximum PCT is below 1200°C (considering uncertainty of 5% this value is changed to 1140°C).
- Criterion 2: The total local oxidation of the fuel does not exceed 17% of the initial cladding thickness before the oxidation onset.
- Criterion 3: The total mass of the reacted Zr is less than 1% of its total mass in the core. It is assumed that this criterion is met when Criterion 2 is fulfilled.

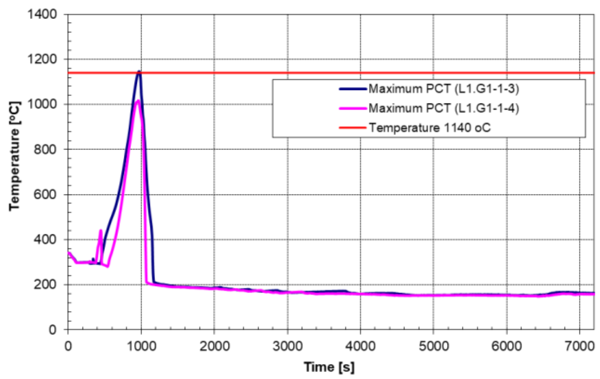
Separately, for the LB LOCA of Group 2 an additional criterion for the maximum PCT of 800°C is applied (considering uncertainty of 5% this value is changed to 760°C). The temperature of 800°C is chosen in compliance with [3]. When the maximum PCT exceeds 800°C (especially for long period of times) a deterioration of the core geometry is possible which in turn may lead to aggravation of the conditions for core cooling.

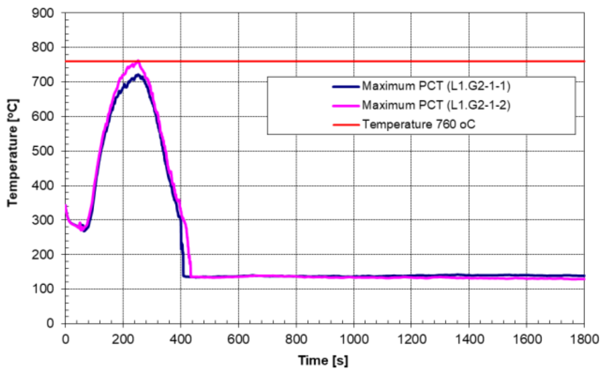
7 Conclusions

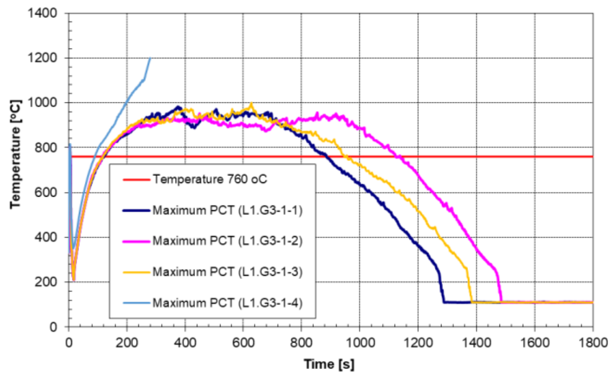
As a result of the performed thermohydraulic analyses the following conclusions are drawn:

- Group 1 of LB LOCA falls into the range with break sizes from DN 80 mm to DN 110 (115) mm. With DN 110 mm CD is reached, but with DN 115 mm a safe end state (OK state) is established. The lower limit for these leaks is considered in case of availability of 2 out of 4 HA and 1 out of 3 LPIPs, and the upper limit in case of failure of 4 out of 4 HA and availability of 1 out of 3 LPIP;









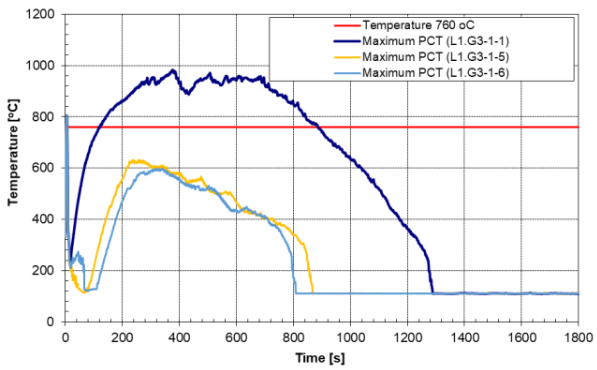


Figure 5. Results for the PCT behaviour for the considered scenarios.

Table 3. Assessment of the acceptance criteria

No	Acceptance criterion	Value	In the representative scenarios			
			L1.G1-1-2	L1.G1-1-4	L1.G2-1-1	L1.G3-1-1
1.	Maximum PCT is below 1200°C (considering uncertainty of 5% this value is changed to 1140°C). For Group 2 an additional criterion is applied	$\leq$ 1200°C (1140°C) $\leq$ 800°C (760°C)	1115.0°C	1017.0°C	722.0°C	983.0°C
2.	The total local oxidation of the fuel pellet does not exceed 17% of the initial cladding thickness before the oxidation onset	$\leq$ 17%	$\sim$ 6.0%	$\sim$ 2.0%	$\sim$ 0.21%	$\sim$ 2.8%
3.	The total mass of the reacted Zr is less than 1% of its total mass in the core.	$<$ 1%	$<$ 1%	$<$ 1%	$<$ 1%	$<$ 1%

- Group 2 of LB LOCA falls into the range with break sizes from DN 115 mm to DN 370 mm (DN 380 mm). For DN 370 mm the criterion for maximum PCT below 760°C is fulfilled, but for DN 380 mm this criterion is violated. In this group, failure of 4 out of 4 HA and availability of 1 out of 3 LPIP are considered.
- Group 3 of LB LOCA falls into the range with break sizes from DN 380 mm to DN 2×850 mm. For these leaks (especially for their upper limit) it is expected that the failure of 4 of 4 HA may lead to deterioration of the core geometry.

For detailed assessment of the possibility for core channels blockage and obstruction of the core cooling, thermo-mechanic analyses should be performed.

Assessment of the defined acceptance criteria is presented in Table 3. The maximum PCT behaviour for the representative scenarios is shown in Figure 5.

References

[1] Актуализация на ВАБ, ниво 1 за пълна мощност, за ниска мощност и за спрян реактор на блокове 5 и 6 на АЕЦ „Козлодуй“ и разширяване на обхвата му с отчитане на вътрешните и външните опасности, характерни за площадката на АЕЦ „Козлодуй и влиянието между блоковете. Инженерен наръчник. Описание, верификация и валидация на модела на РИ за RELAP5/MOD3.3, Договор № 208000019/23.01.2020 г. с АЕЦ „Козлодуй“ ЕАД.

[2] Актуализация на ВАБ, ниво 1 за пълна мощност, за ниска мощност и за спрян реактор на блокове 5 и 6 на АЕЦ „Козлодуй“ и разширяване на обхвата му с отчитане на вътрешните и външните опасности, характерни за площадката на АЕЦ „Козлодуй и влиянието между блоковете. Описание на резултатите от проведените термохидравлични анализи за целите на ВАБ ниво 1 за 5 и 6 блок в АЕЦ „Козлодуй” при номинална мощност на реакторната инсталация, Договор № 208000019/23.01.2020 г. с АЕЦ „Козлодуй“ ЕАД.

[3] Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions. State-of-the-art Report, OECD 2009, NEA No. 6846, ISBN 978-92-64-99091-3.