Effect of passive safety systems on typical beyond-design accidents for WWER-1000/V-392 reactor plant

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1. Introduction

The Russian regulatory documents for nuclear power plant safety contain the requirement on the necessity of the beyond-design-basis accidents (BDBA) consideration as the events and scenarios participating in the formation of the relevant safety systems design basis. In particular, the list of such accidents have to be composed, the acceptance criteria are to be formulated and the realistic analysis of BDBAs have to be made. The designer should tend to the estimated probability of the limiting radioactivity release less than 10^{-7} per reactor-year, and the estimated probability of severe core damage derived on the PSA basis should not exceed 10^{-5} per reactor-year [1].

Such approach results in necessity to provide for the special engineered features dedicated to mitigate the BDBA consequences and to prevent the development of BDBAs to severe accidents. Effect of additional passive safety systems (SPOT, HA-2) on typical beyond-design accidents for WWER-1000/V-392 reactor plant is briefly considered in the present paper.

2. Main characteristics of power unit

Advanced 1000 MWe power unit design with WWER-1000/V-392 reactor plant is developed in the frame of the State program “Ecologically clean power production”. The pilot unit (NVAES-2) is planned for commissioning at Novovoronezh NPP site.

V-392 reactor plant design has been developed on the basis of standard power unit with reactor plant V-320. These units are in operation for a long time at nuclear power plants in Russia, Ukraine, Czech Republic and Bulgaria. The basic characteristics of the reactor plant V-392 are given in Table 1. Reactor plant is placed in the double containment that prevents the radioactivity release to the environment and protects reactor plant against the external impacts. In particular, the reactor plant is capable to withstand to the earthquake of magnitude of 8 as per MSK-64 scale, and to the air shock wave with front pressure up to 30kPa, etc. The technological safety systems are located next to the containment that provides high reliability of fulfillment of their functions.

Table 1

Parameter	Value
Core rated power, MW	3000
Coolant pressure at core outlet, MPa	15,7
Coolant flow rate through reactor, m^3/h	86000
Steam pressure at steam generator outlet, MPa	6,27

The NVAES-2 design is based on the evolutionary approach, i.e. mainly the equipment and processes that have proved their operability at operating nuclear power plants are applied. At the same time, the application of the advanced equipment and additional passive safety systems are implemented to reach higher level of the plant safety.

Examples of such advancements in the safety increasing area are as follows:

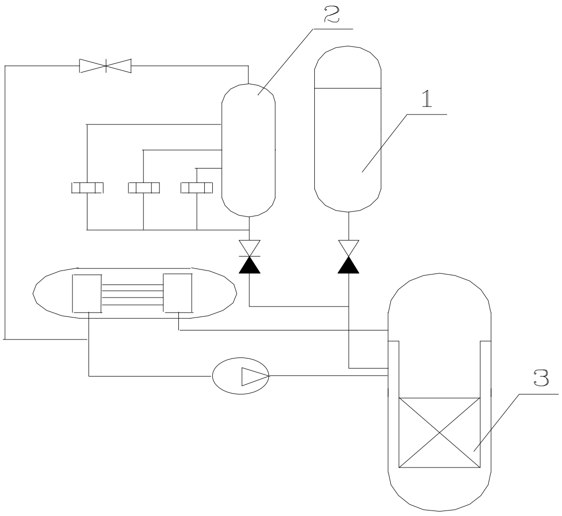
- advanced reactor WWER-1000;
- passive system of residual heat removal (SPOT);

- passive system for core flooding under loss-of-coolant accidents (HA-2);
- passive system of quick boron supply to reactor;
- primary coolant pump preventing coolant leak under long-term station blackout;
- passive system to create the rarefied atmosphere and to clean the media in the inter-containment gap under accident conditions.

3. Brief description of new passive systems

In NVAES-2 design with V-392 reactor plant, new passive systems are provided to prevent the BDBA transition to severe accident phase with beyond-design core damage up to the core melting. The effect of the passive system for core flooding under loss-of-coolant accidents (HA-2 system) and the passive system for residual heat removal (SPOT system) on typical BDBA for WWER-1000/V-392 reactor plant is considered in present paper.

The HA-2 system (JNG50-80) is intended to supply the boron solution to reactor with the purpose of long term (up to 24 h) cooling of the fuel during LOCAs of different size with active ECCS failure (for example, in case of LB LOCA with station blackout). The HA-2 system consists of four groups (four channels) of the tanks with 16 g/kg boron solution being under atmospheric pressure. The simplified flow diagram of one HA-2 channel is shown in Fig. 1.



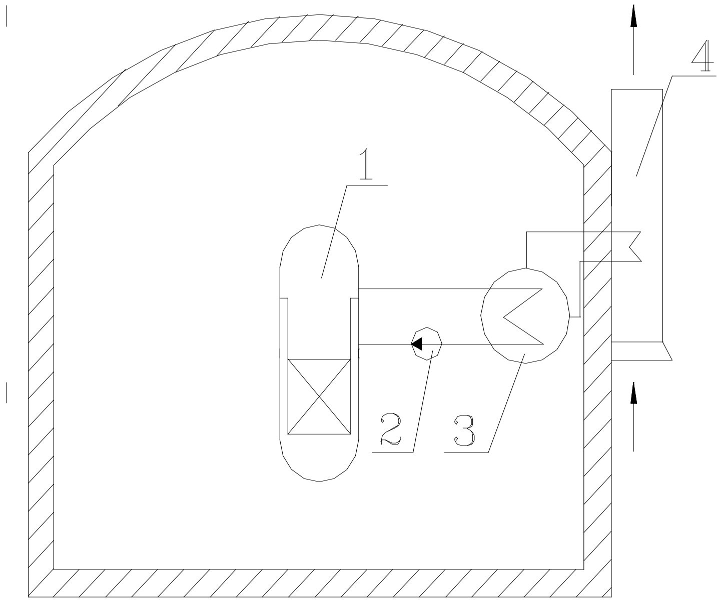
1 - ECCS hydroaccumulator (HA-1)
2 - HA-2 tank (2 pcs.)
3 - reactor

Fig. 1. Flow diagram of HA-2 channel

The HA-2 supply line is connected to pipeline supplying water from the ECCS hydroaccumulator (HA-1 system). The isolation valves are provided on the HA-2 supply line to cut off the HA-2 from the primary circuit if the repair or maintenance is needed. Special check valve will exclude the pressure increase in HA-2 in “wait”. This valve will passively open in case of LOCA after primary pressure decreases to 1,5 MPa.

Top of the HA-2 tank is connected via special valve to cold leg near SG collector. This valve is adjusted to open after primary pressure decreases to 1,5 MPa; after that the pressure in HA-2 increases up to pressure in the primary circuit, and the water flows down to the primary circuit under hydrostatic head. The time dependent profiling of the boric solution flowrate from HA-2 is used to maintain the necessary supply according to decay heat decreasing; this is ensured by appropriate selection of restrictive devices located on the supply lines.

The passive heat removal system (JNB50-80) is intended for the long term residual heat removal under condition with complete loss of feedwater supply to SG in case of intact primary circuit (for example, in case of station blackout). This system can also facilitate to the residual heat removal under certain scenarios of a loss of coolant accident. The simplified diagram of one SPOT channel is shown in Fig. 2.



1 - reactor; 2 - MCP
3 – steam generator; 4 – air heat exchanger
Fig. 2 - Flow diagram of SPOT channel

The system consists of four independent circuits of natural circulation, one for each reactor plant loop. Each circuit includes four heat exchange modules, pipelines for steam and condensate with the valves, air duct path for air suction and discharge. Two isolation valves are installed at heat exchanging modules inlet and outlet to switch off the damaged exchanger from the system for reparation.

During the most unfavorable environmental conditions (temperature of outdoor air +50 ^{0}C), three SPOT channels are capable to remove not less than 2 % of nominal reactor power (heat removal above 2 % at the initial accident stage is ensured by partial evaporation of SG water into the atmosphere through SG dumping devices).

4. Beyond-design accidents without new passive systems

The following typical beyond-design accidents that essentially determine the design basis of the above passive systemsare considered in this paper :

- LB LOCA (double-ended cold leg break 850 mm diameter) with 24 h station blackout;
- station blackout with intact primary circuit.

The HA-2 and SPOT systems operation during these accidents are considered in present paper. For the analysis of these accidents in the framework of this paper, the design structural, thermal and neutron-physics characteristics of the reactor plant were used as they are given in the design documentation.

The “best estimate” approach is used when performing the calculations, namely:

- NPP initial conditions correspond to normal plant operation without consideration of possible deviations and uncertainties of parameters, values of setpoints, etc;
- core characteristics (for example, reactivity coefficients and power peaking factors) are assumed according to neutron-physics analysis without uncertainties and calculational errors;
- equipment failures and operator errors are not assumed.

The domestic codes DINAMIKA-97 [2] and TECH-M-97 [3] developed by EDO “Gidropress” and certificated by GAN RF were used for the analyses. Besides, some comparative calculations with the western codes RELAP5/MOD3.2 [4] and ATHLET 1.2A [5] have been performed as well.

Station blackout

The results of calculation of the station blackout accident without operation of the new passive systems are shown in Figs. 3 - 6. Calculation is performed using DINAMIKA-97, RELAP5/MOD3.2 and ATHLET 1.2A codes.

The initiating event – station blackout:

- closing of turbine generator stop valves;
- disconnection of all RCP;
- disconnection of the primary circuit makeup-blowdown system;
- disconnection of BRU-K;
- disconnection of power supply to PRZ systems;
- disconnection of the main and auxiliary feedwater systems of the secondary circuit.

Besides, all active safety systems do not work due to all diesel-generators failure.

After scram by the fact of trip of three and more RCPs the reactor power decreases to the level of decay heat. After RCP coastdown, the primary coolant natural circulation is settled. The heat from the primary circuit is being removed due to BRU-A operation (during the first period) and then through SG PSD.

The heat removal from the primary circuit decreases due to irreplaceable loss of boiler water from SG through the secondary dump devices, and this results in the primary pressure increasing up to control PRZ SV opening. Later on, the gradual loss of primary coolant through PRZ SV takes place. Decreasing of the water level in upper plenum below then reactor outlet nozzles results in termination of the natural circulation. Continued decreasing of the coolant level in reactor results in the core upper part uncovering, fuel rod temperature increasing and beyond-design damage of the core.

Table 2 presents the chronological sequence of events occurred during this accident.

Table 2

Event	Time, s		
	DINAMIKA-97	RELAP5/ MOD3.2	ATHLET 1.2A
Beginning of the PRZ SV operation	1920	2550	2240
Steam generators emptying	7500	6400	6200
Beginning of the upper plenum boiling	4830	6600	5900
Termination of the natural circulation	6000	7200	6600
The maximum cladding temperature reached 1200 °C	8280	9500	8680

The above results show that the accident behavior predicated by different codes as a whole is practically the same. Exceeding of the maximum design limit of fuel rod damage happens already 2-2,5 h after initiating event, so the development of special engineered features is required for prevention given BDBA progression into severe accident.

Main coolant pipeline break at reactor inlet (2x100% CL LOCA) with station blackout

The results of calculation of the 2x100% CL LOCA with station blackout and without HA-2 operation are shown in Figs. 7, 8. Calculation is performed using TECH-M-97 code.

It is necessary to note that the very initial stage of the accident is practically the same as the design path of its progression. The essential differences starts after ECCS hydroaccumulators are emptied.

After decreasing of primary pressure to 5,88 MPa (at 8,5 s of the accident process) the ECCS hydroaccumulators (HA-1) injection starts, limiting the rate of the reactor emptying and providing filling of the lower plenum to moment of HA-1 emptying at 57,5 s. Due to

HA-1 operation, the primary coolant inventory increases, improving the cooling of the core during some period of time. Then the decreasing of the water level in the reactor (due to release through the leak and evaporation of the water) results in deterioration of the core cooling and its subsequent heating up after 100 s. At 285 s of the accident, the maximum fuel temperature exceeds 1200 ^{o}C (Fig. 8).

The same calculation was performed using RELAP5/MOD3.2 code. The results of calculation are shown in Figs 9 and 10. As one can see from these plots, the character of the transient is similar to that obtained above by TECH-M-97 code. The fuel rod heat up also begins at ~100 s, and the maximum fuel rod temperature exceeded 1200 ^{o}C at 280 s (Fig. 10).

Thus for prevention of given BDBA progression into severe accident the development of special engineered features is needed.

5 Calculations of typical BDBAs with operation of new passive systems

The results of the typical BDBA considered above indicate the necessity to provide for additional engineered features, intended to prevent the progression of a BDBA into severe accident. In the present chapter, the calculation results of the same typical BDBAs, but with new passive systems (HA-2, SPOT) operation are shown. It was assumed that all four channels of this systems in operation (SPOT during the first period works in the control mode, and after 1800 s is switched over by operator to cooldown mode).

Station blackout

The results of calculation of the station blackout accident with operation of the new passive systems are shown in Figs. 11 - 14. Calculation is performed using DINAMIKA-97, RELAP5/MOD3.2 and ATHLET 1.2A codes.

At the first stage, the behavior of the accident is similar to considered in chapter 4. However as a result of SPOT operation some part of the heat from primary circuit is being removed to the environment, and other part releasing through BRU-A (the loss of the boiler water from SG is continued). After the relevant decreasing of decay heat, the secondary circuit dumping devices are closed and loss of boiler water from SG is stopped. The heat removal from the primary circuit takes place due to SPOT operation in the closed scheme (the vapor from SG is condensed in heat exchanging modules, and the condensate returns back to SG). The plant parameters are decreasing, the reliable cooling of the core is assured.

So, the calculation results have shown that the SPOT operation in considered BDBA prevents any damage of the core.

Main coolant pipeline break at reactor inlet (2x100% CL LOCA) with station blackout

The results of calculation of the 2x100% CL LOCA with 24 h station blackout and with the new passive systems (HA-2 and SPOT) operation are shown in Figs. 15-18. Calculation is performed using TECH-M-97 code. The optimized (taking into account the pre-determined containment pressure change) dependence of the water flowrate from HA-2 was used in calculation.

It was assumed for the SPOT operation that the operator switches it in cooldown mode after 1800 s, and the steam generators cooldown by the secondary circuit will begin from this moment. Duration of the SG cooldown phase amounts 4800 s (Fig. 15). After temperature in the secondary circuit decreases lower than the temperature in the primary circuit, the steam generators pass in mode of the primary circuit steam condensation returning the condensate to primary loops.

The makeup from HA-2 keeps the water inventory in the primary circuit at about 50 tons, and it ensures the cooling down condition in the core. At this, the major part of the coolant is located in the reactor. After HA-2 transition to the final stage of the water supply (at 30000 s), the primary coolant inventory decreases (approximately up to 32 tons), however the core heating up is not observed.

Thus, the results of calculation have shown that during analyzed 24 hours of accident the joint work of SPOT and HA-2 provides the acceptable temperature conditions of the core (without exceeding the maximum design limit of fuel rod damage).

The calculation of the accident with double ended break of main coolant pipeline with 24 h station blackout and with the new passive systems (HA-2 and SPOT) operation was performed also using RELAP5/MOD3.2 code. The calculation results presented in Figs. 19 and 20 have shown that the acceptable cooling of the core is guaranteed during all analyzed time of the accident.

6 Conclusion

Operation of the new passive systems (SPOT and HA-2) in considered beyond-design accidents provides a possibility of effective cooling of the core during required 24 hours of accident. This ensures the essentially decreased probability of severe core degradation.

SPOT operation during beyond-design accident with station blackout in case of intact primary circuit allows to stop a loss of boiler water from SG and to provide for residual heat removal from the core.

In some scenarios of beyond-design accidents with leaks from the primary circuit, the SPOT system provides for additional makeup to the primary circuit by the condensate which is generated in SG tubes. This improves the temperature conditions of the core. The value of back pressure in the containment has the important effect on SPOT effectiveness. The greater value of the containment pressure corresponds to the larger power of the SPOT and, accordingly, to the larger amount of the condensate being returned to loops.

References

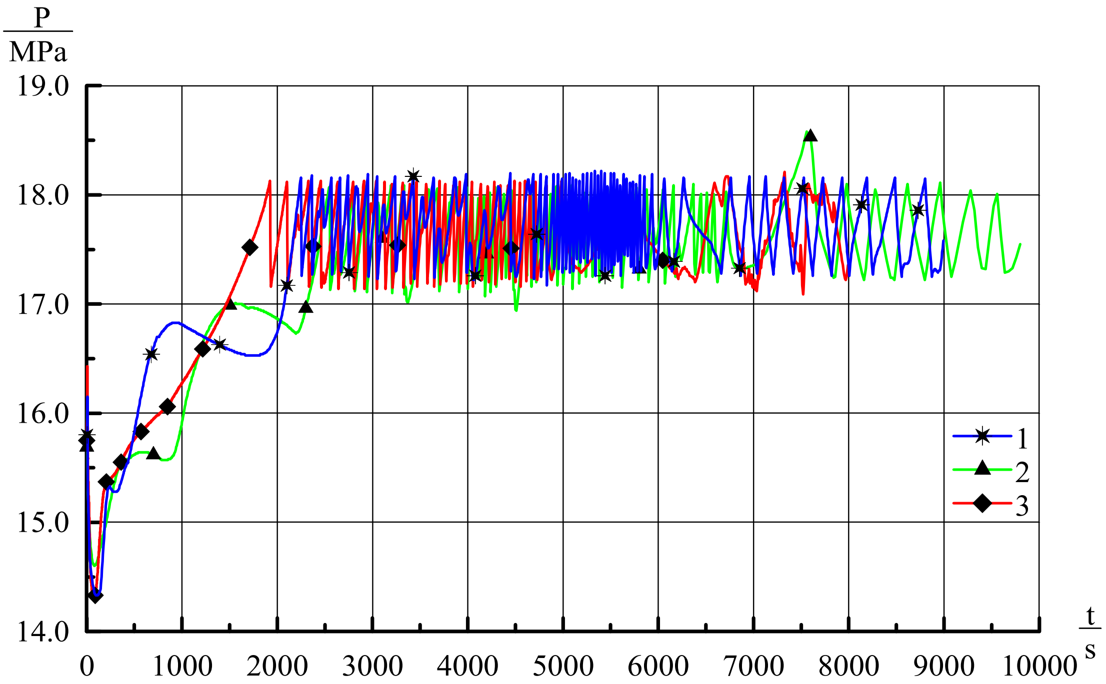
1 General provisions for NPP safety (OPB-88/97). PNAE G-01-011-97 (in Russian).

2 Программа для ЭВМ. Расчет нестационарных режимов энергетических установок с ВВЭР «Динамика-97». Методика расчета. 8624607.00467-019001. ОКБ «Гидропресс», 1998 (Паспорт аттестации ГАН России №110 от 02.09.99).

3 Программа для ЭВМ. Расчет параметров первого контура при разрывах трубопроводов «ТЕЧЬ-М-97». Методика расчета. 8624607.00466-019001. ОКБ «Гидропресс», 1998 (Паспорт аттестации ГАН России №112 от 02.09.99).

4 RELAP5/MOD3 code manual. The RELAP5 Code Development Team. Idaho National Engineering Laboratory, June 1995, NUREG/CR-5535.

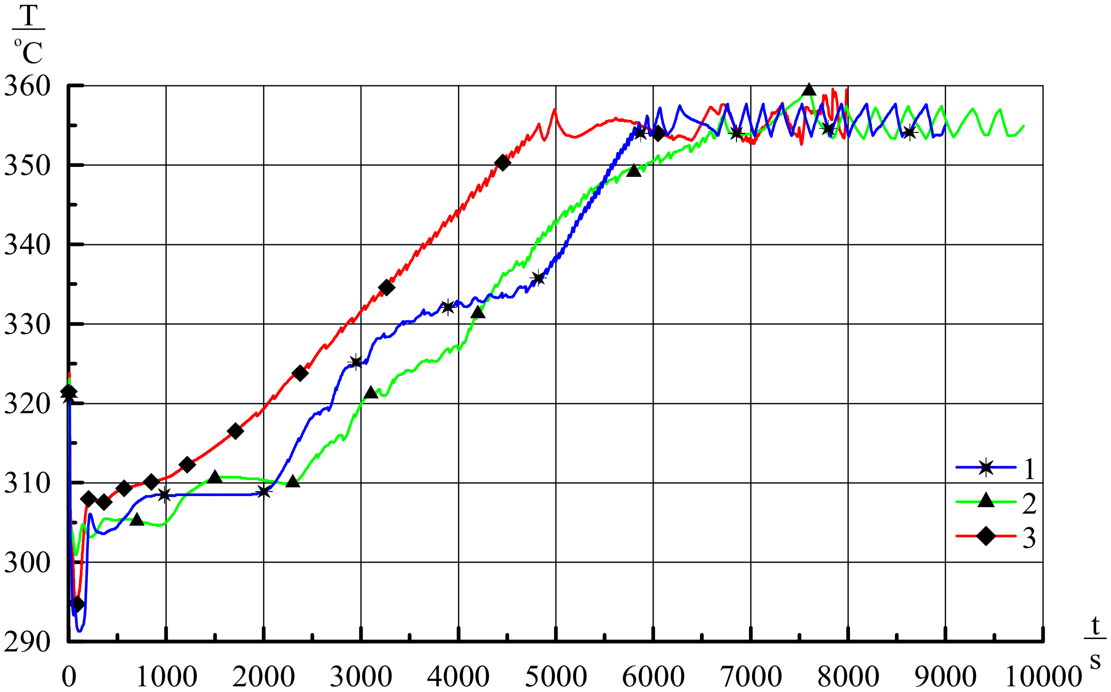
5 ATHLET Mod 1.2 Cycle A. User’s Manual. G. Lerchl, H. Austregesilo, March 1998, GRS-P-1 / Vol. 1, Rev. 1.



Pressure at the core outlet:

1 – ATHLET 1.2A
2 – RELAP5/MOD3.2
3 – DINAMIKA-97

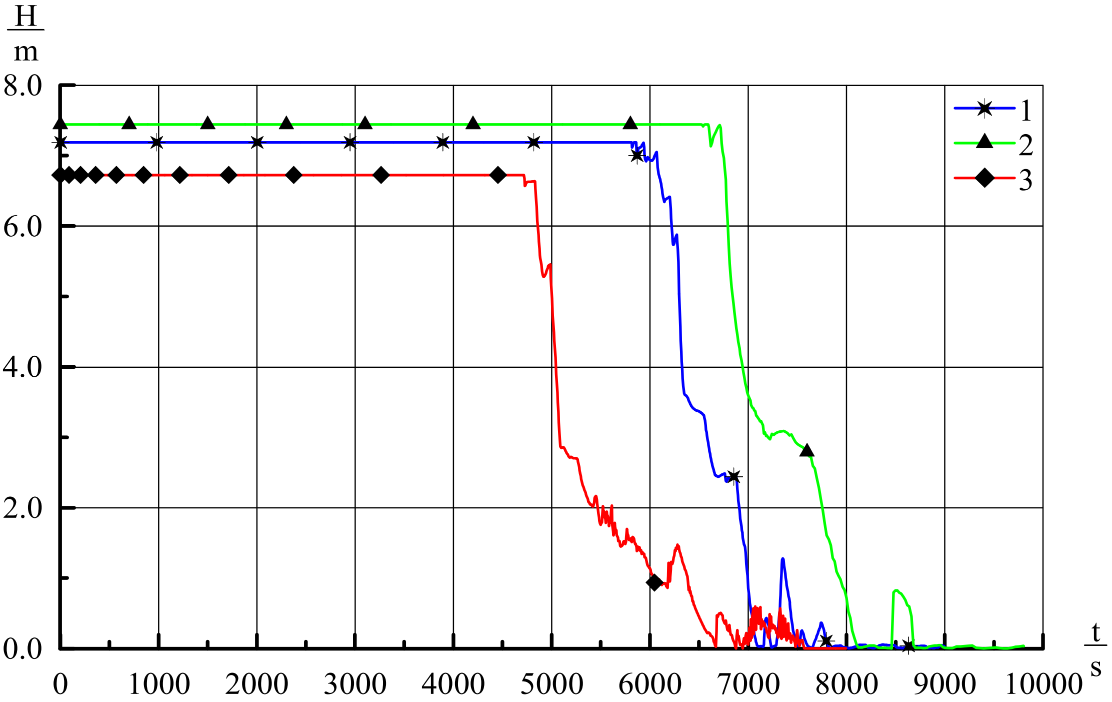
Fig. 3 – Station blackout (without SPOT)



Coolant temperature at the reactor outlet:

1 – ATHLET 1.2A
2 – RELAP5/MOD3.2
3 – DINAMIKA-97

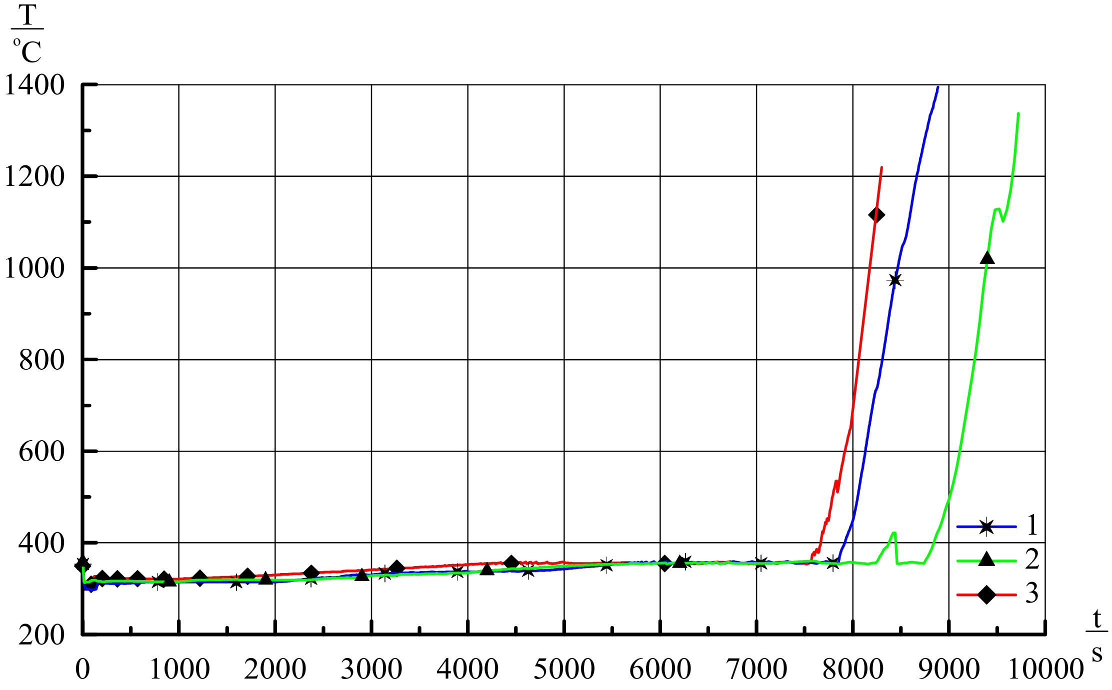
Fig. 4 – Station blackout (without SPOT)



Collapsed level in upper plenum:

1 – ATHLET 1.2A
2 – RELAP5/MOD3.2
3 – DINAMIKA-97

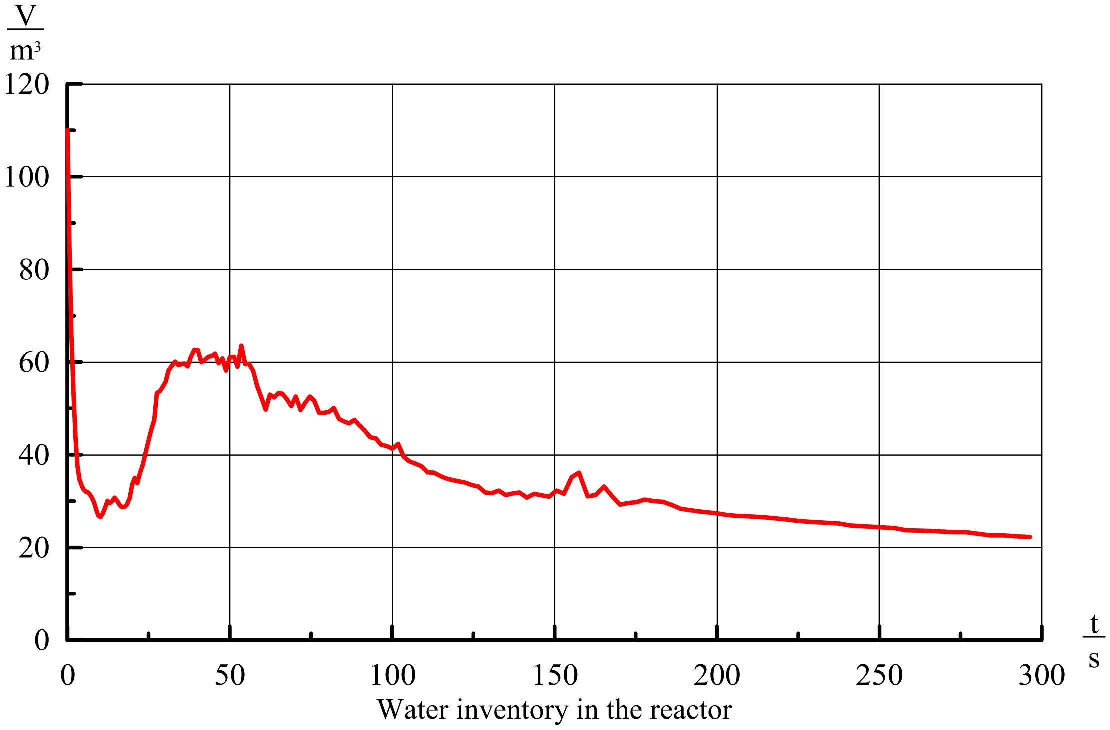
Fig. 5 – Station blackout (without SPOT)



Maximum temperature of fuel rod claddings:

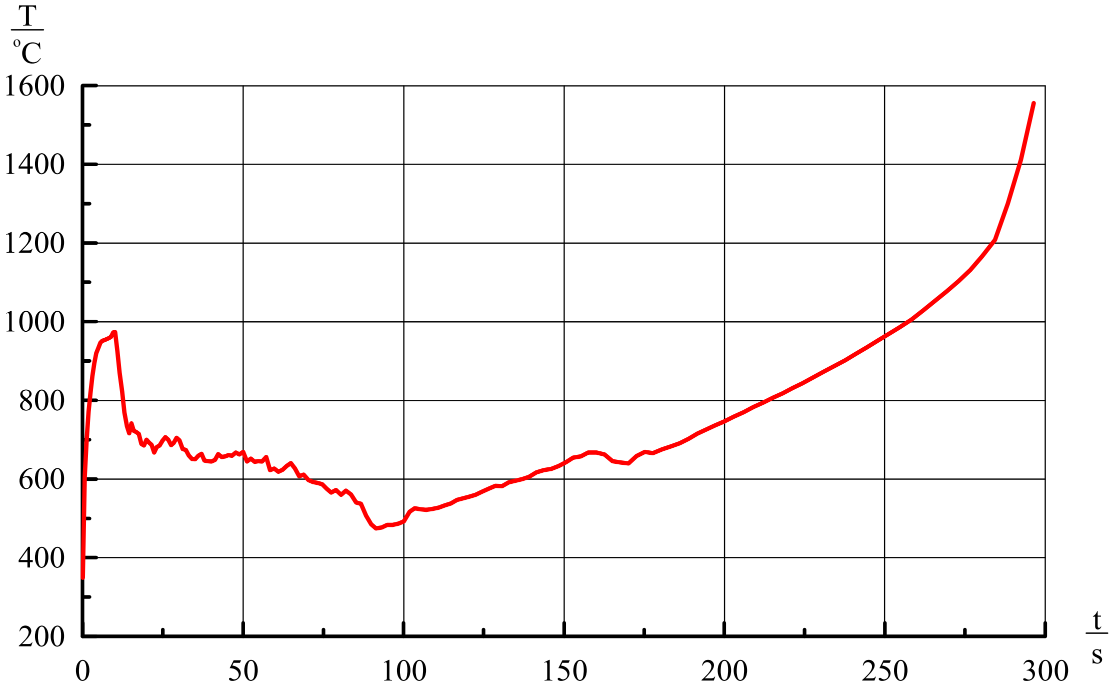
1 – ATHLET 1.2A
2 – RELAP5/MOD3.2
3 – DINAMIKA-97

Fig. 6 – Station blackout (without SPOT)



Water inventory in the reactor

Fig. 7 – 2x100% CL LOCA with station blackout (without HA-2, TECH-M-97)



Maximum temperature of fuel rod claddings

Fig. 8 – 2x100% CL LOCA with station blackout (without HA-2, TECH-M-97)

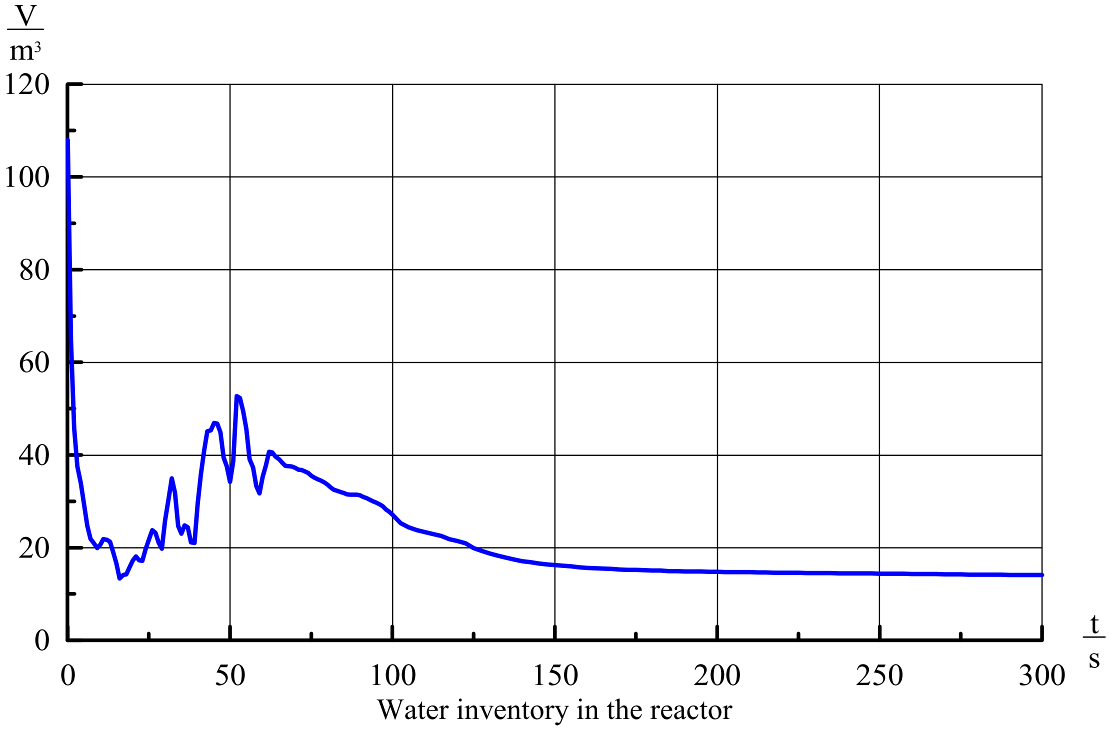


Fig. 9 – 2x100% CL LOCA with station blackout (without HA-2, RELAP5/MOD3.2)

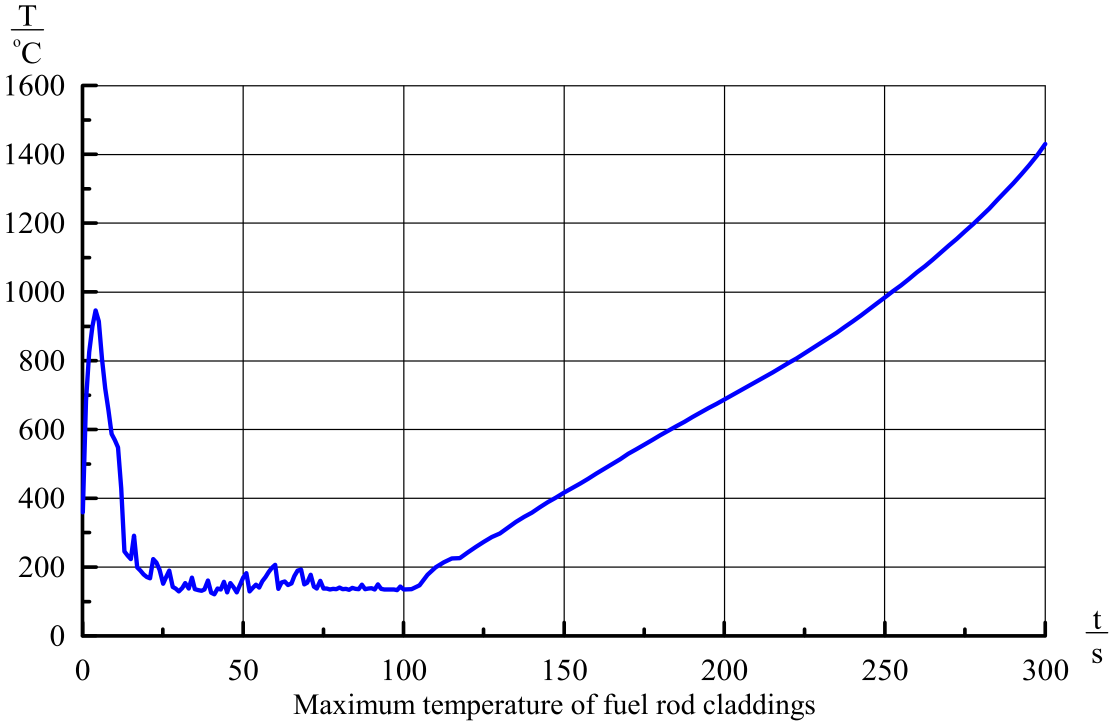
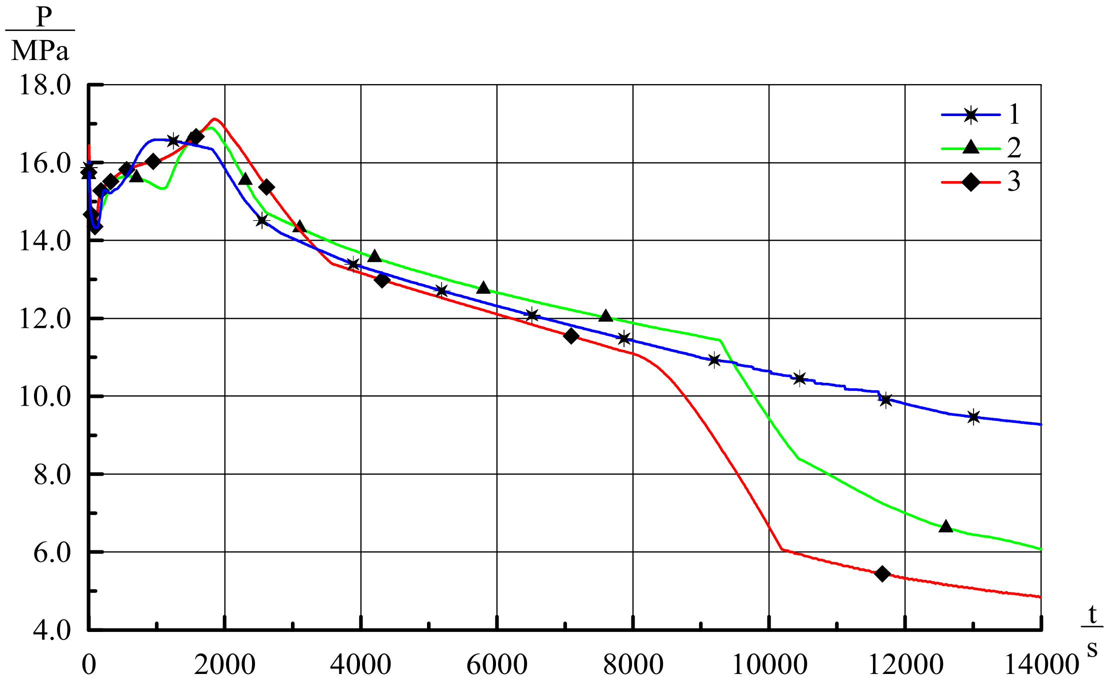


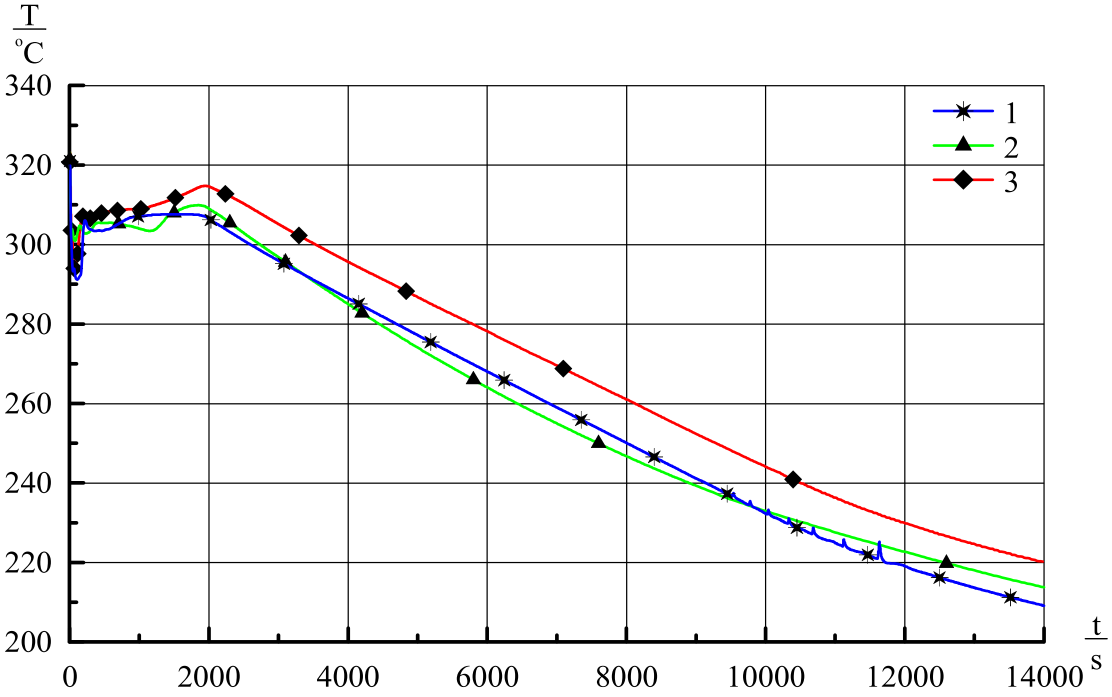
Fig. 10 – 2x100% CL LOCA with station blackout (without HA-2, RELAP5/MOD3.2)



Pressure at the core outlet:

1 – ATHLET 1.2A
2 – RELAP5/MOD3.2
3 – DINAMIKA-97

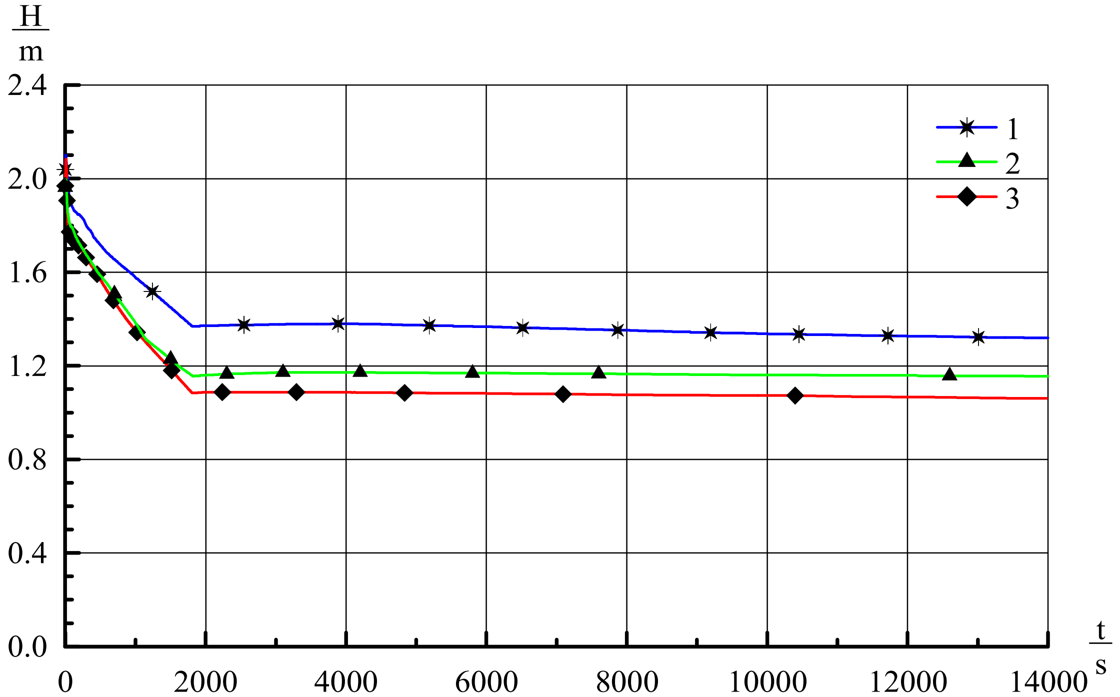
Fig. 11 – Station blackout (with SPOT)



Coolant temperature at the reactor outlet:

1 – ATHLET 1.2A
2 – RELAP5/MOD3.2
3 – DINAMIKA-97

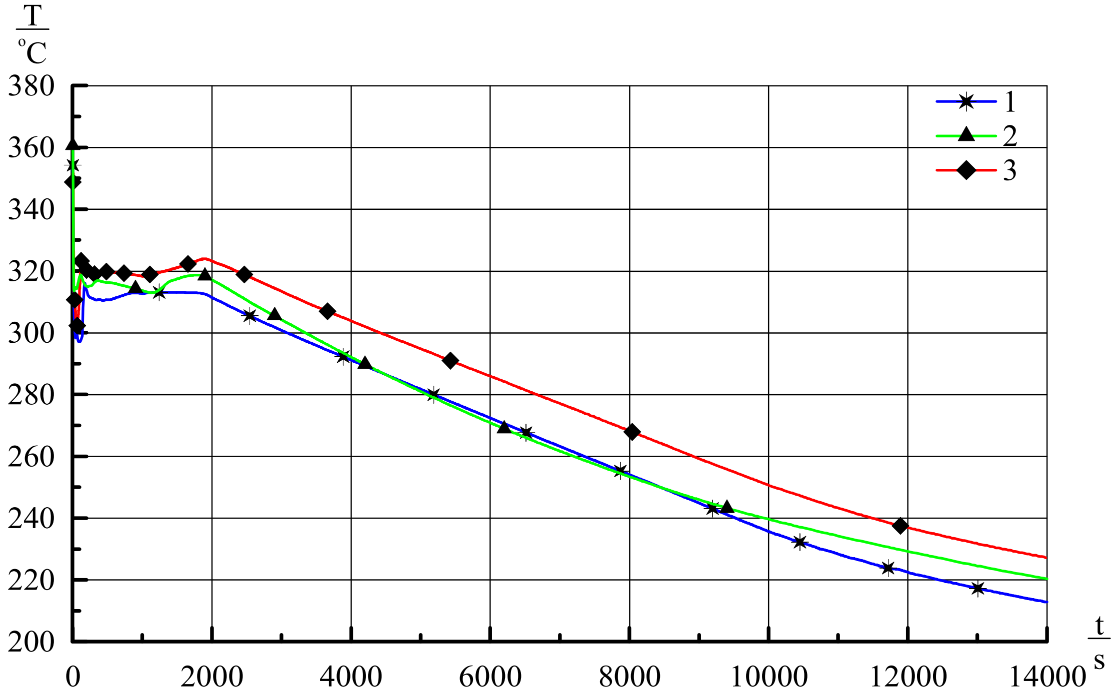
Fig. 12 – Station blackout (with SPOT)



Collapsed level in SG:

1 – ATHLET 1.2A
2 – RELAP5/MOD3.2
3 – DINAMIKA-97

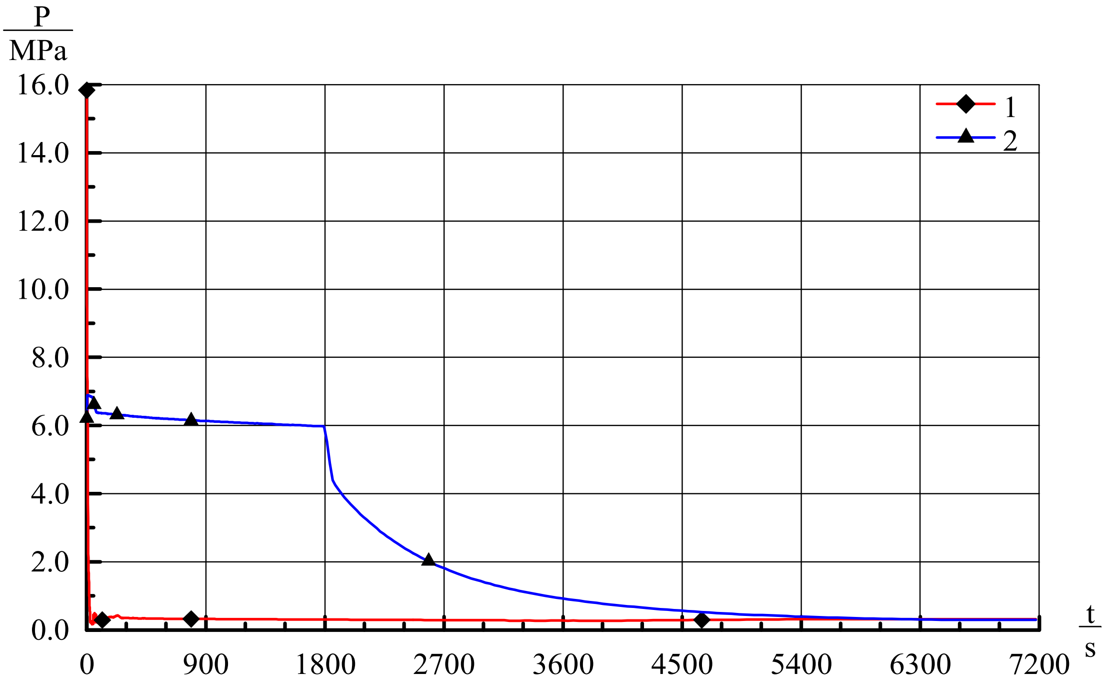
Fig. 13 – Station blackout (with SPOT)



Maximum temperature of fuel rod claddings:

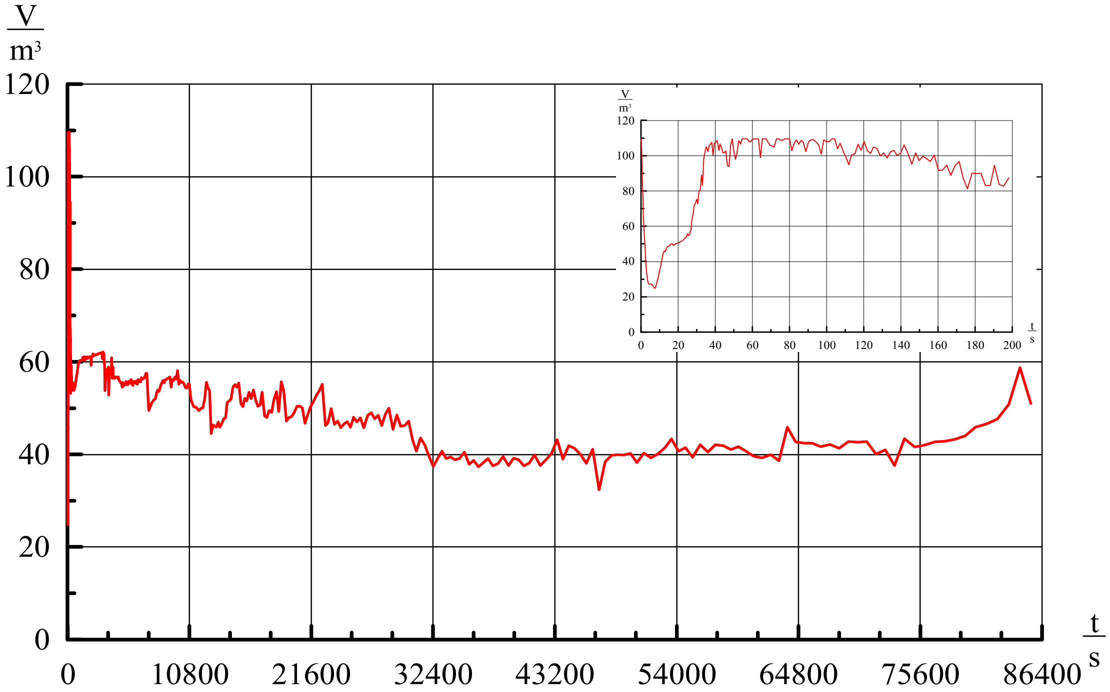
1 – ATHLET 1.2A
2 – RELAP5/MOD3.2
3 – DINAMIKA-97

Fig. 14 – Station blackout (with SPOT)



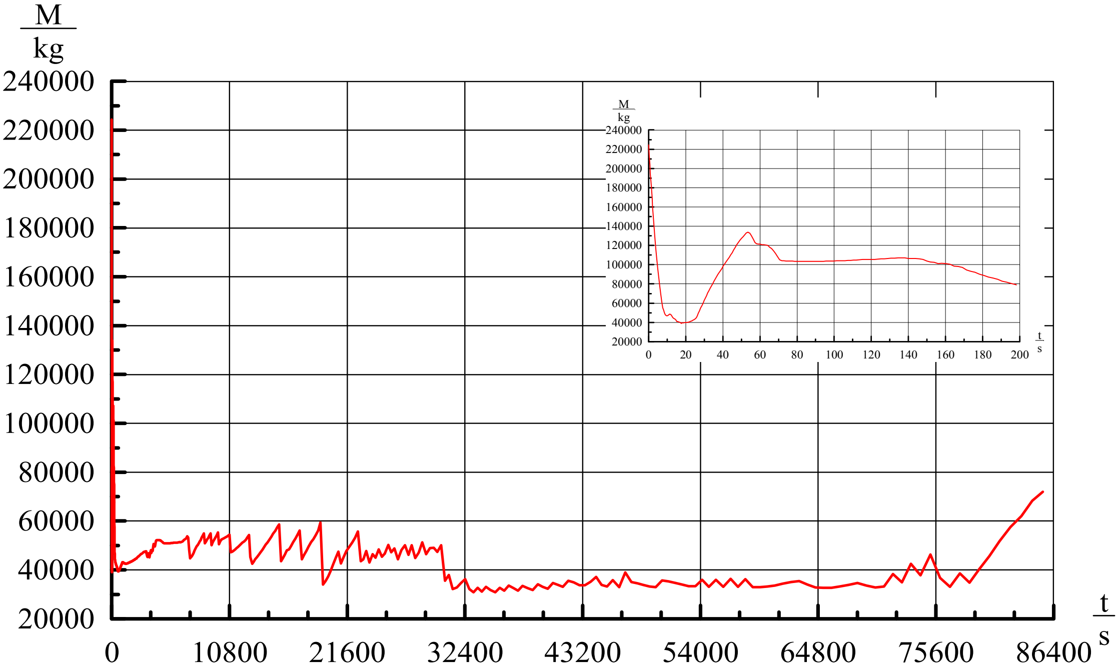
1 – pressure at the core outlet
2 – pressure in SG

Fig. 15 – 2x100% CL LOCA with station blackout (with HA-2 and SPOT, TECH-M-97)



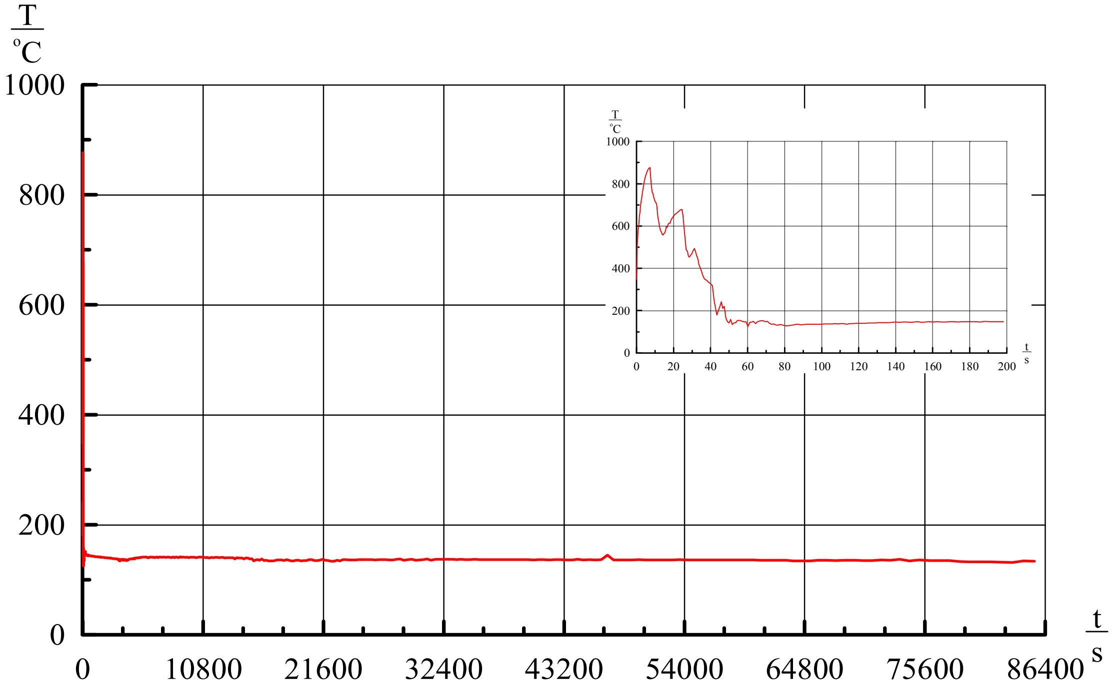
Water inventory in the reactor

Fig. 16 – 2x100% CL LOCA with station blackout (with HA-2 and SPOT, TECH-M-97)



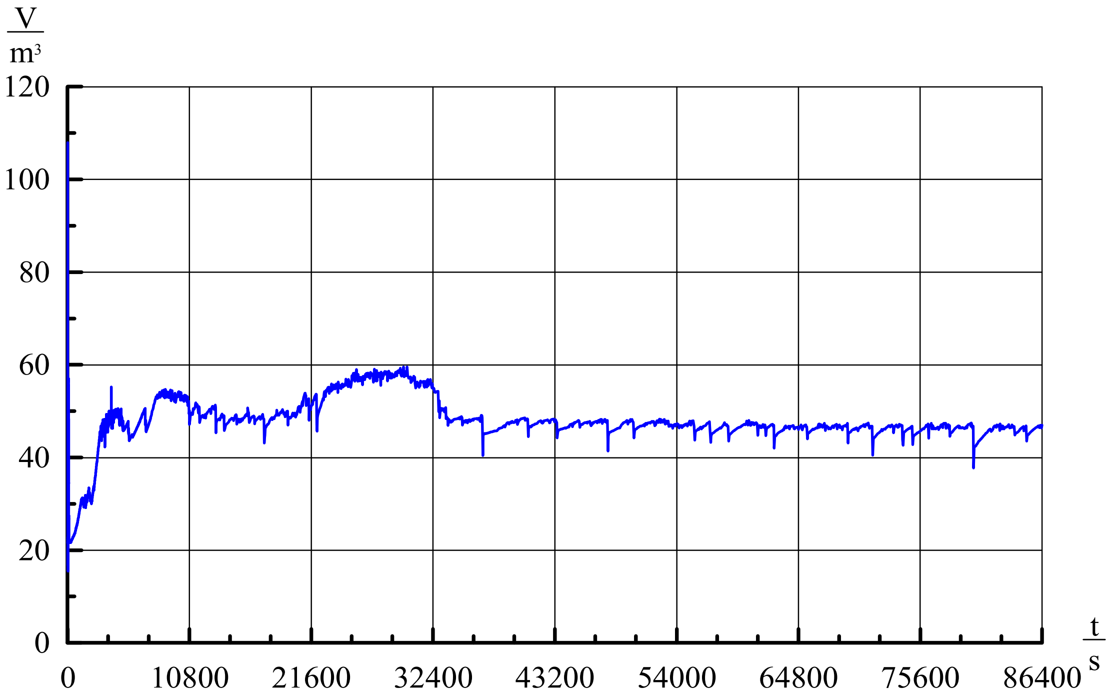
Mass of the primary coolant

Fig. 17 – 2x100% CL LOCA with station blackout (with HA-2 and SPOT, TECH-M-97)



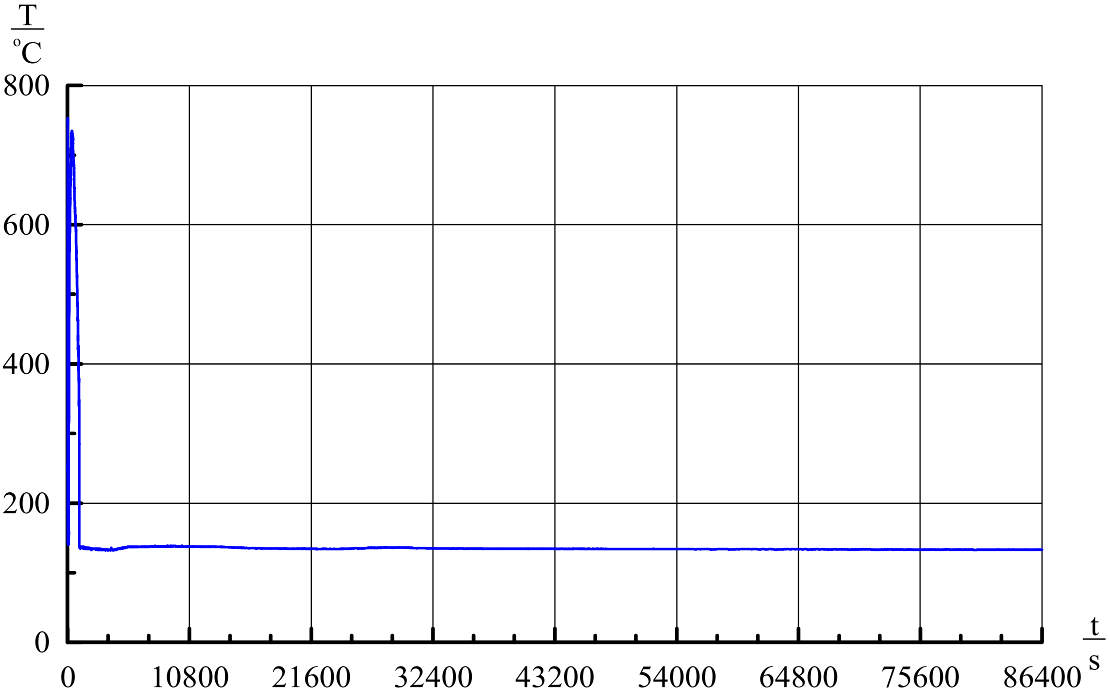
Maximum temperature of fuel rod claddings

Fig. 18 – 2x100% CL LOCA with station blackout (with HA-2 and SPOT, TECH-M-97)



Water inventory in the reactor

Fig. 19 – 2x100% CL LOCA with station blackout
(with HA-2 and SPOT, RELAP5/MOD3.2)



Maximum temperature of fuel rod claddings

Fig. 20 – 2x100% CL LOCA with station blackout
(with HA-2 and SPOT, RELAP5/MOD3.2)