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The Variant Analysis for Assessment of the Core Condition of Reactor Type WWER-440 (V-230) in case of Loop Guillotine Rupture

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INTRODUCTION

This paper presents the result of an assessment of the status of the WWER-440 (B-230) reactor core, related to the thermal-hydraulic behavior and the probability of events going along with this accident. For this purpose a methodology for the study is elaborated, which comprises both realistic and conservative elements of the approach in analyzing such type of events (LOCA). This methodology complies with the requirements of BNSA Regulation №3 promulgated in state gazette No.27 of 24 Apr 1988. and PNAEG-024-90 of 01 Sep 1990 and with the recommendations of the IAEA set in IAEA-EBP-WWER-01. Based on these documents, the acceptance criteria applicable to the analyzed event are defined.

The analysis is presented by two basic scenarios of the accident progression:

1. ***Rupture of a primary loop without LPSI***
2. ***Rupture of a primary loop with LPSI***

Each scenario is also analyzed from the point of view of availability or loss of off-site power, applying a using single failure to an active element of the safety systems. A matrix of the studied variants is elaborated on the basis of which an evaluation of the core conditions is performed. Such evaluation is also done based on the probability for occurrence of such type of event, the probability of the applied single failure, as well as the probability of the postulated break location.

METHODOLOGY FOR CORE ASSESMENTS

Single failure [1] – in case of availability of off-site power, a failure of one LPIP is postulated, and in case of LOOP a failure of one DG is assumed.

Break location – it is assumed that the break is in the cold leg of the main circulation loop, between the primary isolation valve and the reactor inlet nozzle.

Off-site power – two possible events are analyzed: a case with and a case without LOOP.

Safety systems – a conservative assumption is made in the analyses that two LPSI pumps inject to loops with mechanical coast-down.

Fuel Campaign and Reactor Kinetics – according to IAEA [3] recommendations LB LOCA is analyzed for the beginning of the fuel campaign. Moderator density and Doppler feedback coefficients are used [4].

Boundary conditions – in the cases with conservative elements, the values of the main parameters of the units are shifted in a direction unfavorable for the overall accident progression.

Reactor scram – it is assumed that the reactor protection will be actuated by the first signal, which is initiated by the event [1]. Besides, it is taken into account that ECCS injects boron solution to the primary circuit [2].

Core hot channel – the model of the reactor core includes a hot cannel with a representative hot pin, defined according to the requirements of [2].

Code used for the analysis - RELAP5/3.2 code, used for the analysis belongs to the group of advanced best estimate thermal-hydraulic codes developed by NRC for this purpose [2].

Matrix for the analysis – for completeness of the core assessment for the considered event, the analyses are made according to an a-priory developed matrix covering all the possible cases required by [5].

Acceptance criteria – The acceptance criteria are defined based on the requirements of the existing regulations – BNSA Regulation №3, state gazette No.27/24.04.1988, ПНАЭГ-024-90/01.09.1990 and IAEA requirements specified in IAEA-EBP-WWER-01 [1].

WWER-440 (V-230) MODELING

The reactor model (Fig.1.1) is developed taking into consideration the new requirements regarding a realistic reactor modeling. The inlet nozzles and down-comer between the RPV and reactor shaft are three-dimensionally modeled in accordance with the number of circulation loops.

The lower plenum and the reactor bottom are split into five axial nodes. The upper plenum and reactor head volumes are represented by six axial components. The reactor outlet part is modeled similarly to the reactor inlet and down-comer.

The reactor core is represented as three channels – average channel, hot channel with a representative hot pin and core by-pass.

The fuel parts of the assemblies in the core are modeled as ten axial nodes. This applies to all components in the core: average channel, hot channel and by-pass.

The model of the primary circuit (Fig.1.3 and 1.4) of the reactor type WWER-440 (V-230) includes all six loops, each of them including a MCP, the SG primary side and the respective elements of the Safety System trains. The Pressurizer is connected to one of the loops.

The SG tube bundle (Fig.1.2) is split into five horizontal layers, taking into account the number and length of the tubes in the respective layer. In SG secondary side the free volumes in the SG tube bundle region, the down-comer around the tube bundle, the separator and the volumes around the separator are modeled.

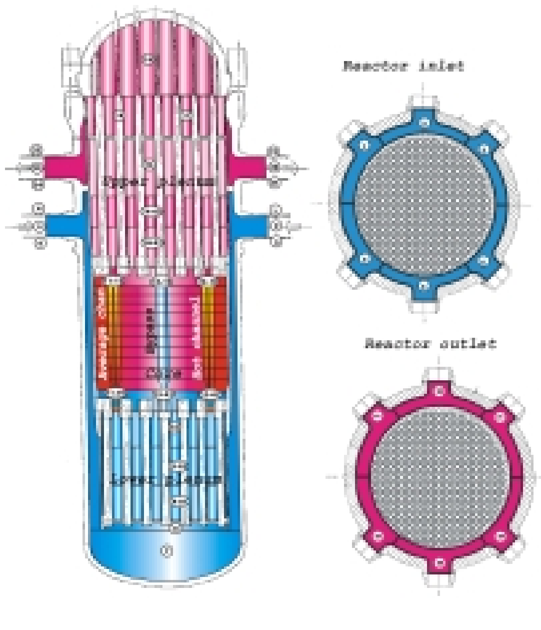

Fig.1.1

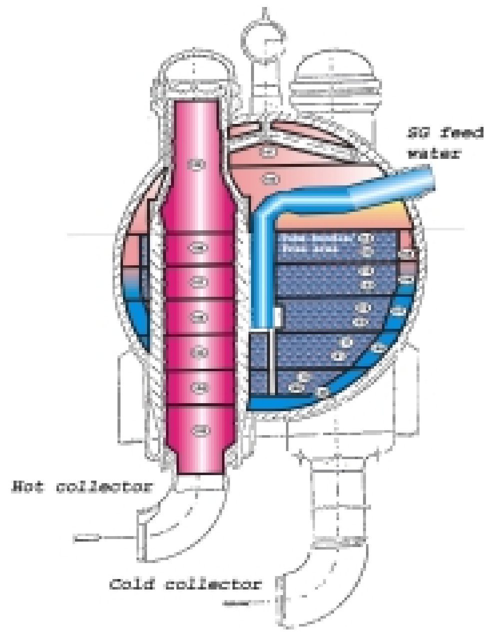

Fig.1.2

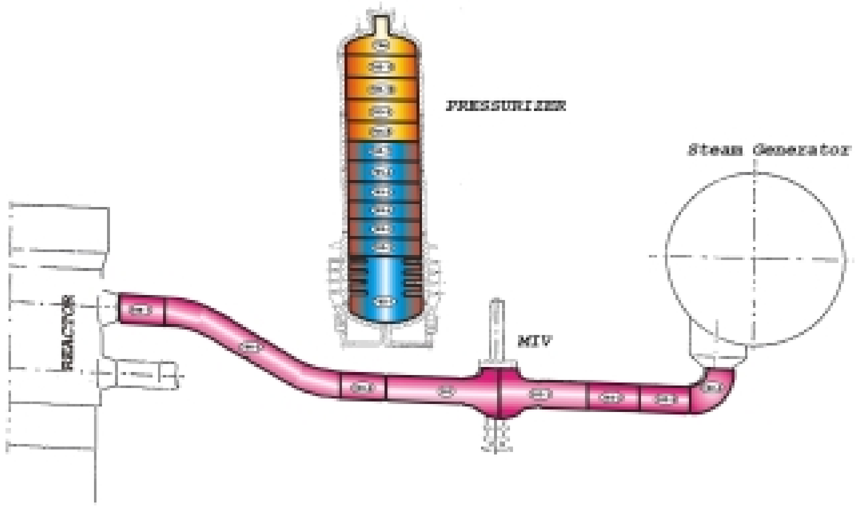

Fig.1.3

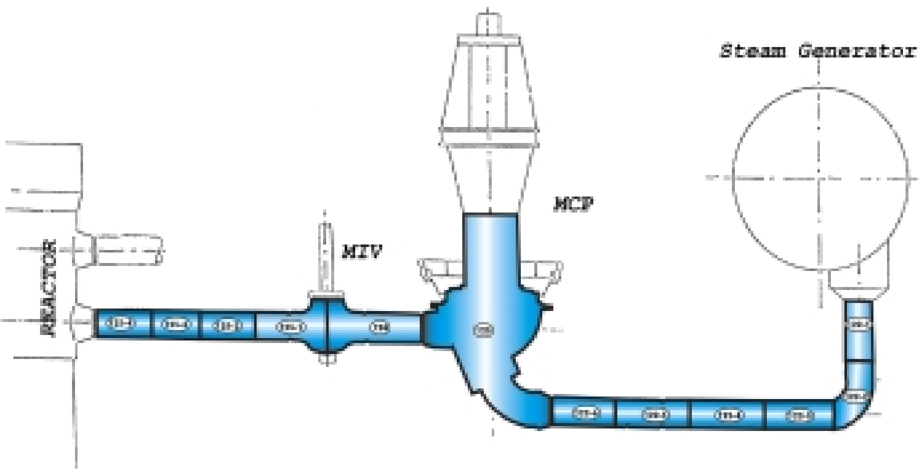

Fig.1.4

THERMO-HYDRAULIC PARAMETRIC ANALYSES OF THE PRIMARY CIRCIUT

ANALYSES MATRIX

According to the developed methodology aimed at fulfillment of the requirements of [5], a matrix for performing of the analyses is elaborated.

Matrix for the analyses of 2x100% LOCA at KNPP units III and IV Table 1.1

Parameters	Variants							
	1.1	**1.2**	**2.1**	**2.2**	**3.1**	**3.2**	**4.1**	**4.2**
Off-site power	Without LOOP	With LOOP	Without LOOP	With LOOP	Without LOOP	With LOOP	Without LOOP	With LOOP
Rupture location	Loop without LPIP		Loop without LPIP		Loop with LPIP		Loop with LPIP	
Single failure	One LPIP	One DG	One LPIP	One DG	One LPIP	One DG	One LPIP	One DG
Unit lifetime	15 cycle, BOC – III 13 cycle, BOC - IV		15 cycle, BOC – III 13 cycle, BOC - IV		15 cycle, BOC – III 13 cycle, BOC - IV		15 cycle, BOC – III 13 cycle, BOC - IV	
Main assumption:								
- reactor power, MW	1375.0		1430.0		1375.0		1430.0	
- primary pressure, MPa	12.36		12.56		12.36		12.56	
- Pressurizer level	nominal		nominal		nominal		nominal	
- ECCS signal, MPa	10.38		10.18		10.38		10.18	
- HPIP flow	nominal		-5%		nominal		-5%	
- LPIP flow	nominal		-5%		nominal		-5%	
- SCRAM actuation, MPa	11.38		11.18		11.38		11.18	
- decay heat	ANS79-1		ANS79-1		ANS79-1		ANS79-1	
- MCP turnover	nominal		nominal		nominal		nominal	
- reactor flow	nominal		nominal		nominal		nominal	
- feedback coefficients	assumed		assumed		assumed		assumed	

A SUMMARY OF THE ANALYSIS OF THE SCENARIO WITHOUT LOOP

During the first several seconds of the accident a rapid increase of the cladding temperature is observed. This is a result of the loss of level, boiling of the coolant and reverse flow through the reactor core as a consequence of the large flow through the break in the cold leg. A little later the break flow decreases and a part of the coolant is re-directed to the core. This causes a decrease of the cladding temperature. This process is also influenced by the change of the power, which begins slightly to increase in this moment, i.e. a positive reactivity insertion as a result of increased coolant density is observed. This process continues up to the moment of actuation of the reactor protection.

The pressure in the primary circuit quickly decreases. Set points for actuation of RP-1 and starting the ASSS program on “technological parameter” are reached. 10 s after scram actuation the turbine stop valves are closed. Due to decrease of the coolant level in the reactor below its outlet nozzles the circulation through it is stopped. This means that the secondary circuit no longer acts as a heat sink. All the energy generated in the core is removed with the break flow.

Conditions, when HPSI starts to inject in the primary circuit, are reached. The three HPIS pumps start to inject boron solution to the circuit. With a certain delay the LPSI pumps also start to inject boron solution, but with lower boron concentration. Taking into account the fact that as the single failure is assumed the failure of one LPSI pump two LPSI pumps start to inject effectively. In spite of this, the cladding temperature begins to increase. The gradient and the maximum of the peak cladding temperature depend entirely on the efficiency of the emergency core cooling system. This is well manifested in the analysis of the event with rupture of a primary loop with low pressure injection, when the overall ECCS injection is decreased (only one LPIS pump remains effective).

The sequence of events of the scenario without LOOP for units III and IV is given in Table 1.2, and the main parameters are presented in Fig.1.5-1.6 and in Fig.1.9 and 1.10.

Sequence of events for the analyses of Units III and IV Table 1.2

Event	Time, s							
	Variant 1.1		Variant 2.1		Variant 3.1		Variant 4.1	
	III	IV	III	IV	III	IV	III	IV
2x100% rupture of the cold leg of a primary loop	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Reactor scram signal (delay-1,5 s).	0.04	0.04	0.05	0.05	0.04	0.04	0.05	0.05
Signal for ASSS start (1,0 s)	0.06	0.06	0.06	0.06	0.06	0.06	0.06	0.06
Actuation of reactor scram	1.55	1.55	1.56	1.56	1.55	1.55	1.56	1.56
Turbine trip (10 s after scram actuation)	12.16	12.15	12.17	12.16	12.15	12.15	12.17	12.16
3 HPSI pumps - start; - begin to inject.	 6.06 21.06	 6.06 21.06	 6.06 21.06	 6.06 21.06	 6.06 21.06	 6.06 21.06	 6.06 21.06	 6.06 21.06
2 LPSI pumps: - start; - begin to inject.	 16.06 42.79	 16.06 43.00	 16.06 42.75	 16.06 43.00	 16.06 43.00	 16.06 43.00	 16.06 42.57	 16.06 43.00
Beginning of core heat-up	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0
Peak cladding temperature: - value, ^{o}C; - time of reaching.	 750.0 141.0	 727.0 147.0	 795.0 152.0	 754.0 157.0	 936.0 203.0	 888.0 207.0	 1023.6 223.0	 945.0 222.0
End of reflood of the core	230.0	230.0	240.0	247.0	368.0	373.0	402.0	405.0
End of calculation	600.0	600.0	600.0	600.0	600.0	600.0	600.0	600.0

A SUMMARY OF THE LOOP SCENARIO ANALYSIS

The sequence of events of the LOOP scenario for units III and IV is given in Table 1.3, and the main parameters are presented graphically in Fig.1.7-1.8 and in Fig.1.11 and 1.12.

Sequence of events for analysis of Units III and IV Table 1.3

Event	Time, s							
	Variant 1.2		Variant 2.2		Variant 3.2		Variant 4.2	
	III	IV	III	IV	III	IV	III	IV
2x100% rupture of the cold leg of a primary loop	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Reactor scram signal (delay-1,5 s).	0.04	0.04	0.05	0.05	0.04	0.04	0.05	0.05
Actuation of reactor scram	1.55	1.55	1.56	1.56	1.55	1.55	1.56	1.56
Turbine trip (10 s after scram actuation)	12.16	12.15	12.17	12.16	12.16	12.15	12.17	12.16
LOOP. Signal for ASSS start.	12.16	12.15	12.17	12.16	12.16	12.15	12.17	12.16
Two MCPs coast-down mechanically, rest four MCPs coast-down electro-mechanically.	12.16	12.15	12.17	12.16	12.16	12.15	12.17	12.16
Two DGs start.	13.17	13.15	13.18	13.16	13.17	13.15	13.18	13.16
Two DGs are ready.	48.17	48.15	48.18	48.16	48.17	48.15	48.18	48.16
ASSS consequence: 2 HPSI pumps: - start; - begin to inject. Umps: - start; - begin to inject.	 53.17 68.17 63.17 78.17	 53.15 68.15 63.15 78.16	 53.18 68.18 63.18 78.18	 53.16 68.16 63.16 78.17	 53.17 68.17 63.17 78.17	 53.15 68.16 63.15 78.16	 53.18 68.18 63.18 78.18	 53.16 68.17 63.16 78.17
Beginning of core heat-up.	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0
Peak cladding temperature: - value, ^{o}C; - time of reaching.	 977.0 220.0	 889.0 224.0	 1052.0 236.0	 922.0 231.0	 1240.0 306.0	 1132.0 311.0	 1469.0 341.0	 1152.0 329.0
End of reflood of the core	320.0	320.0	353.0	339.0	500.0	479.0	538.0	514.0
End of calculation	700.0	600.0	700.0	600.0	700.0	600.0	700.0	600.0

The thermal-hydraulic processes in the primary circuit in this case are similar to the previous scenario until the occurrence of LOOP. The sole exemption is the starting of the ASSS program, which, in case of LOOP is shifted in the time.

Loss of off-site power is assumed as a consequence of turbine trip. Two MCPs coast-down mechanically, while the rest four pumps coast-down electro-mechanically, with the house-loads generator. DGs are started by loss of power signal and their start-up time delays ECCS starting and beginning of injection into primary circuit. This leads to higher cladding temperatures and longer time necessary for core re-flooding (in case of rupture of a cold leg with LPSI), respectively core cooling. Another significant factor influencing the said parameter, is the assumed single failure – failure of one DG. The result is loss of one ECCS train (one HPSI pump and one LPSI pump). Two HPIP and two LPIP remain effective. This HPSI and LPSI configuration is valid in case of rupture of a loop without LPIS, but in case of rupture of a loop with LPIS the efficiency of an additional LPSI pump is lost. The values of the peak cladding temperature are highest in this case.

MAIN RESULTS CONCERNING THE ACCEPTANCE CRITERIA

The results of the variant calculations of double-sided guillotine rupture of the cold leg of the main circulation loop for KNPP units III and IV are presented in Table 1.4.

The important parameters for evaluation of the fulfillment of the acceptance criteria are the following:

- ***cladding temperature lower than 1200 °C;***
- ***local depth of cladding oxidation less than 18 % of its original depth;***
- ***fraction of reacted zirconium less than 1% of the total zirconium mass in the core (or according to [1] the hydrogen mass generated by the steam-zirconium reaction shall be less than 1% (5,12 kg) of the hypothetical hydrogen mass, which would be generated if all zirconium in the core reacts with the steam).***

Table 1.4

Double-sided guillotine rupture of a loop without low pressure safety injection												
Without loss of off-site power						**With loss of off-site power**						
Var.№/ Unit №		**T_{clad}, °C**	**T_{fuel}, °C**	**%clad oxid.^{max}**	**H2 gen., kg**	**Var.№/ Unit №**		**T_{clad}, °C**	**T_{fuel}, °C**	**%clad oxid.^{max}**	**H2 gen., kg**	
1.1	III	750.	773.	0.0062	0.0011	1.2	III	977.	1000.	0.1780	0.0292	
	IV	727.	746.	0.004	0.00054		IV	889.	909.	0.0606	0.0020	
2.1	III	795.	820.	0.0131	0.0023	2.2	III	1052.	1075.	0.4273	0.0954	
	IV	754.	775.	0.006	0.00089		IV	922.	939.	0.0970	0.0075	
Double-sided guillotine rupture of a loop with low pressure safety injection												
Without loss of off-site power						**With loss of off-site power**						
Var.№/ Unit №		**T_{clad}, °C**	**T_{fuel}, °C**	**%clad oxid.^{max}**	**H2 gen., kg**	**Var.№/ Unit №**		**T_{clad}, °C**	**T_{fuel}, °C**	**%clad oxid.^{max}**	**H2 gen., kg**	
3.1	III	936.	960.	0.1358	0.0344	3.2	III	1240.	1255.	1.8300	0.4984	
	IV	888.	910.	0.0722	0.01327		IV	1132.	1149.	0.9152	0.1264	
4.1	III	1023.	1045.	0.3343	0.0968	4.2	III	1469.	1477.	5.7220	2.4246	
	IV	945.	969.	0.1428	0.03279		IV	1152.	1161.	1.2585	0.2512	

ASSESMENT OF THE OVERALL PROBABILITY OF THE EVENTS

For the assessment of the consequences from a double-sided guillotine rupture of a primary loop DN 500 mm for the core it is necessary to evaluate the probability of the sequences of events, arisen during the accident. The results of this assessment are presented in Table 1.5.

Probability of events sequence Table 1.5

Event	Probability			
	Unit III		Unit IV	
	Without loss off-site power	**With loss off-site power**	**Without loss off-site power**	**With loss off-site power**
2x100% rupture of a loop without LPSI	$2.0e^{-6}$ (Var.1.1÷2.1)	$2.0e^{-6}$ (Var.1.2÷2.2)	$3.0e^{-6}$ (Var.1.1÷2.1)	$3.0e^{-6}$ (Var.1.2÷2.2)
2x100% rupture of loop with LPSI	$2.0e^{-6}$ (Var.3.1÷4.1)	$2.0e^{-6}$ (Var.3.2÷4.2)	$3.0e^{-6}$ (Var.3.1÷4.1)	$3.0e^{-6}$ (Var.3.2÷4.2)
Failure of LPIP	$4.23e^{-5}$ (Var.1.1÷4.1)	-	$4.23e^{-5}$ (Var.1.1÷4.1)	-
Failure of DG	-	$5.13e^{-5}$ (Var.1.2÷4.2)	-	$5.13e^{-5}$ (Var.1.2÷4.2)
Rupture of a loop without LPSI and failure of LPIP	$2.0e^{-6}x4.23e^{-5}= 8.46e^{-11}$ (Var.1.1÷2.1)	$2.0e^{-6}x5.13e^{-5}= 1.026e^{-10}$ (Var.1.2÷2.2)	$3.0e^{-6}x4.23e^{-5}= 1.269e^{-10}$ (Var.1.1÷2.1)	$3.0e^{-6}x5.13e^{-5}= 1.539e^{-10}$ (Var.1.2÷2.2)
Rupture of a loop with LPSI and failure of LPIP	$2.0e^{-6}x4.23e^{-5}= 8.46e^{-11}$ (Var.3.1÷4.1)	$2.0e^{-6}x5.13e^{-5}= 1.026e^{-10}$ (Var.3.2÷4.2)	$3.0e^{-6}x4.23e^{-5}= 1.269e^{-10}$ (Var.3.1÷4.1)	$3.0e^{-6}x5.13e^{-5}= 1.539e^{-10}$ (Var.3.2÷4.2)

Without LOOP – Unit III
Single failure – one LPIP

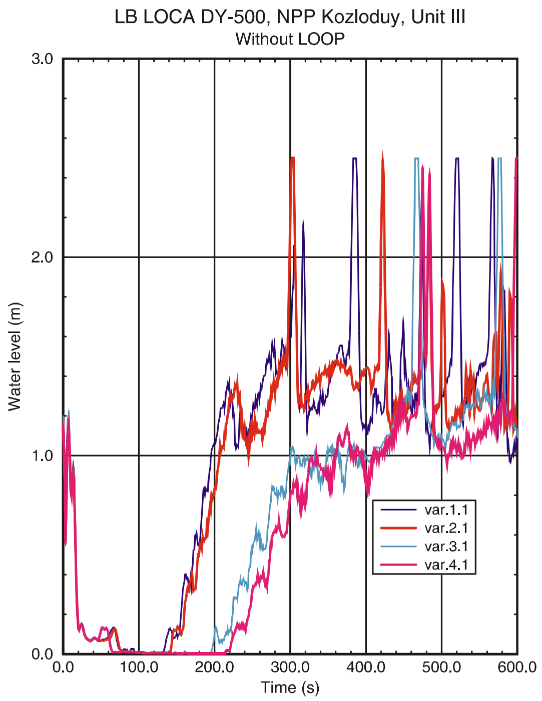

Fig.1.5 Reactor core water level (collapsed).

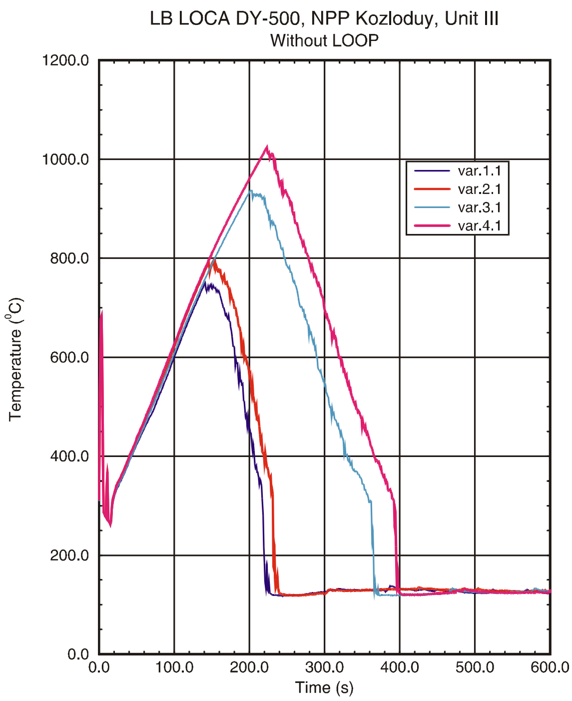

Fig.1.6 Maximal fuel cladding temperatures.

With LOOP – Unit III
Single failure – one DG

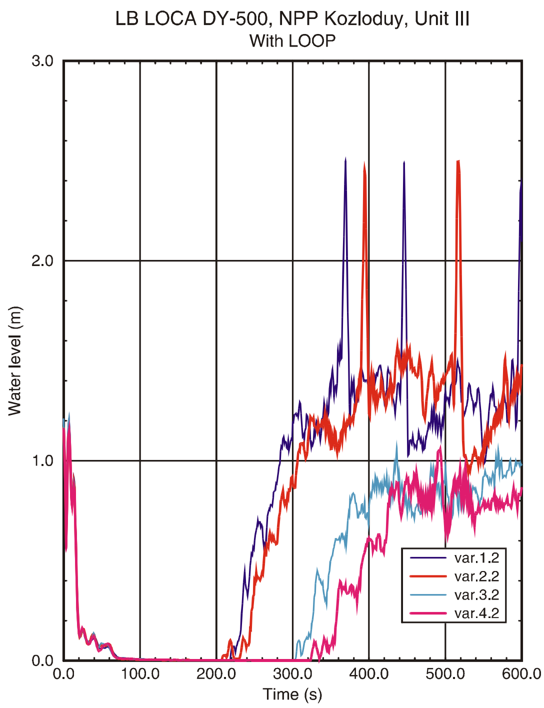

Fig.1.7 Reactor core water level (collapsed).

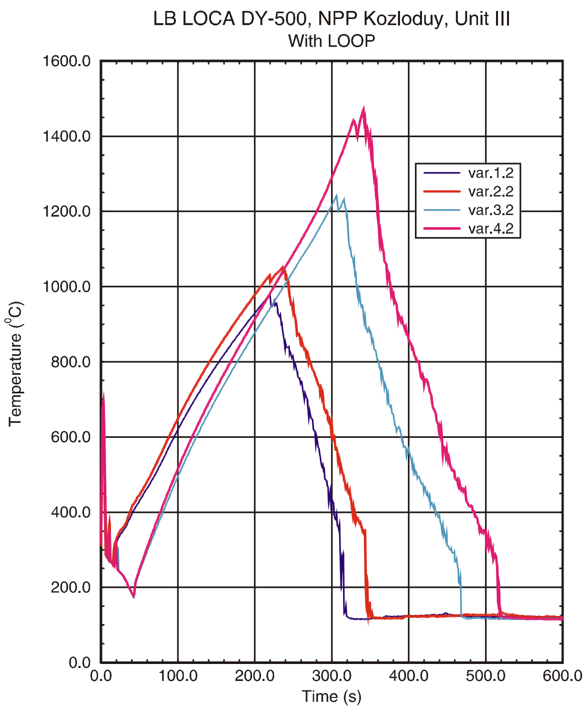

Fig.1.8 Maximal fuel cladding temperatures.

Without LOOP – Unit IV
Single failure – one LPIP

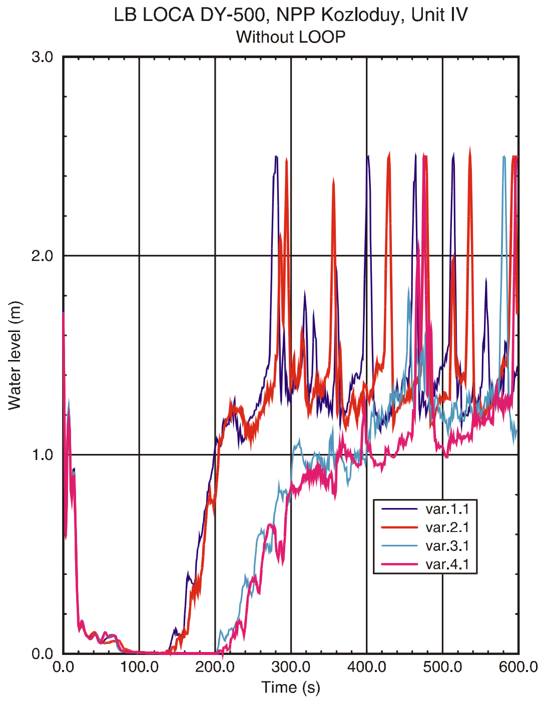

Fig.1.9 Reactor core water level (collapsed)

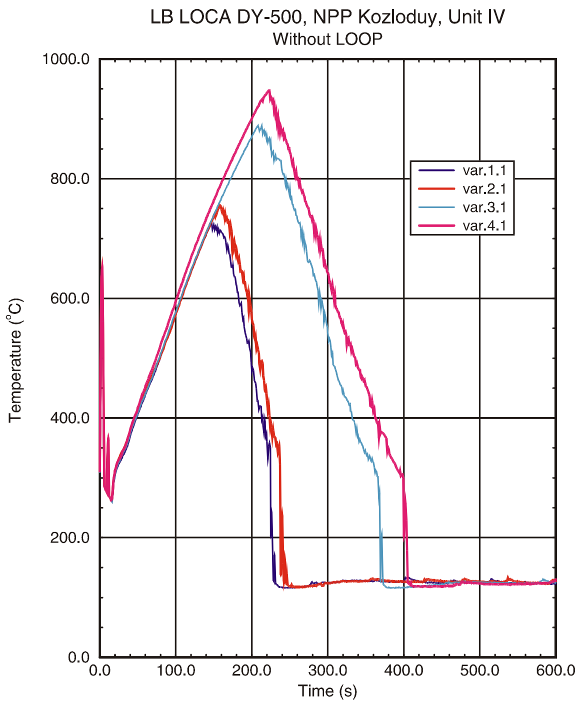

Fig.1.10 Maximal fuel cladding temperatures

With LOOP – Unit IV
Single failure – one DG

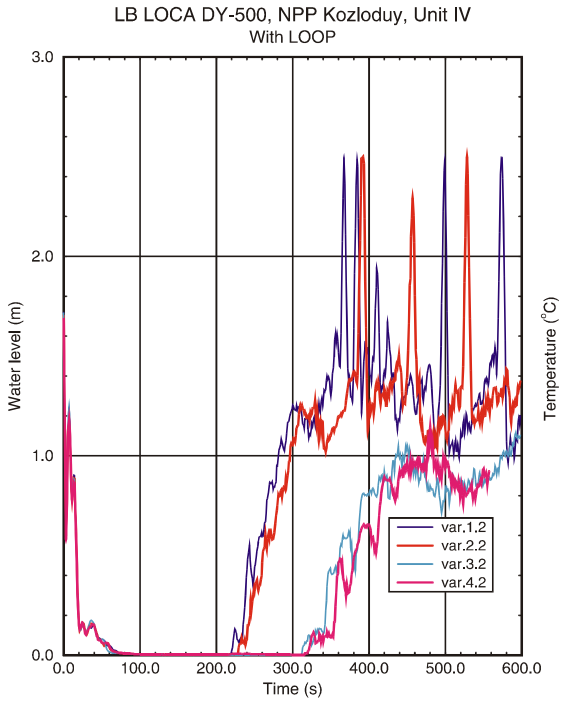

Fig.1.11 Reactor core water level (collapsed)

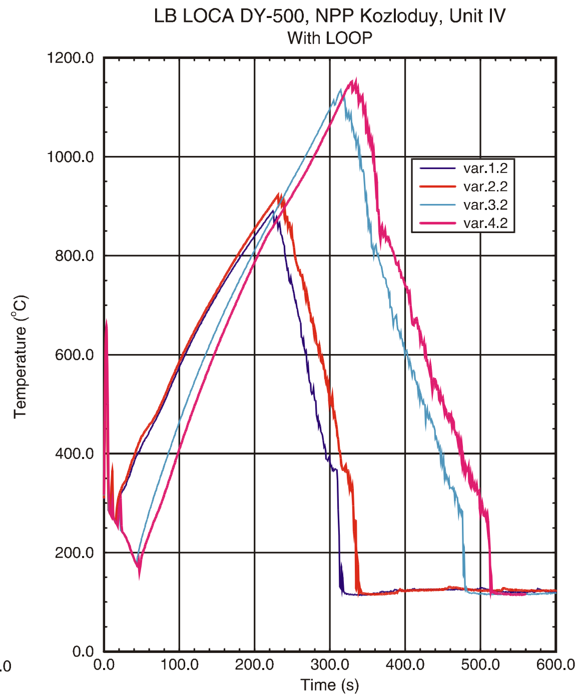

Fig.1.12 Maximal fuel cladding temperatures

CONCLUSIONS

The results of the analyses of the most severe cases for both units are summarized in Table 1.6.

Table 1.6

Acceptance criteria	Maximal parameters			
	Unit III		Unit IV	
	Without loss off-site power (Var.4.1)	With loss off-site power (Var.4.2)	Without loss off-site power (Var.4.1)	With loss off-site power (Var.4.2)
Peak cladding temperature < 1200 ^{o}C	1024. ^{o}C	1469. ^{o}C	945. ^{o}C	1152. ^{o}C
Maximum cladding oxidation depth < 18 %	0.334 %	5.72 %	0.143 %	1.2585 %
Mass of H_2 generation < 1 % (5,12 kg)	0.097 kg	2.425 kg	0.033 kg	0.2512 kg

For KNPP unit III higher maximum values of the parameters are reached due to the higher core peaking factors.

SCENARIOS WITHOUT LOOP

According to the results of the variants without LOOP we can draw the conclusion that one effective LPIP and three effective HPIP are sufficient for prevent the violation of the acceptance criteria for loss of coolant accidents.

SCENARIOS WITH LOOP

In case of LOOP more time for core re-flooding is necessary. As a result, higher peak cladding temperatures are reached in comparison with the variant without LOOP.

The assumed conservatism and the failure of one DG in the analysis of unit III, result in a violation of the acceptance criteria, which as a whole depends on the location of the ECCS injection points and the break position. To prevent the violation of the accepted criteria it is necessary at least two LPSI pumps to inject effectively to the primary circuit.

According to the results of the reviewed variants for unit IV the conclusion is that one effective LPIP and two effective HPIP are sufficient to prevent the violation of the acceptance criteria. Taking into account the application of realistic approach with conservative elements, it is necessary to point out the minimum safety margin to fulfillment of the acceptance criterion on the peak cladding temperature. Therefore the statement for necessity of two effective LPIP is valid.

PROBABILISTIC ASSESSMENT

The probabilistic assessment of the events and scenarios shows that the combination of the initiating event and assumed independent failure place the considered accident in the category of events with low probability.

LIST OF ABREVIATIONS

ASSS	Automatic sequential start of the safety systems
BNSA	Bulgarian Nuclear Safety Authority
BOC	Beginning of Campaign
DG	Diesel-Generator
ECCS	Emergency core cooling system
HPIP	High Pressure Injection Pump
HPSI	High Pressure System Injection
IAEA	International Atomic Energy Agency
KNPP	Kozloduy Nuclear Power Plant
LB LOCA	Large Break Loss of Coolant Accident
LOOP	Loss Of Off-site Power
LPIP	Low Pressure Injection Pump
LPSI	Low Pressure System Injection
LPSI	low pressure safety injection system
MCP	Main Coolant Pump
MIV	Main Isolation Valve
NRC	Nuclear Regulatory Commission
RP-1	First order of Reactor Protection (reactor scram)
RPV	Reactor Pressure Vessel
SG	Steam Generator
SS	Safety system
WWER	Water-cooled water-moderated energetic reactor

REFERENCES

[1] Guidelines for accident analyses of WWER NPP, IAEA EBP-WWER-01, OEC. 1995
[2] Best-estimate calculation of Emergency Core Cooling System performance, Regularity Guide 1.157, (Task RS 701-4)
[3] Guidance for Accident Analyses of Commercial Nuclear Power Plants, Appendix A – Specific Information Related to Light Water Reactors, Vienna, Austria, 1999
[4] Lifetime input data for Units III and IV, EP-1, KNPP (in bulgarian)
[5] Terms of reference TR-32/17.10.2000, from PRG’97, TZ.M.2.2/3 (in bulgarian)
[6] LB LOCA probability and equipment failures input data for unit III and IV, EP-1, KNPP (in bulgarian)