

# Accident Analyses to Evaluate Necessity and Effectiveness of WWER-440 Protection System Modifications

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## 1. Introduction

At present eleven WWER-440/230 reactors and their modifications (V-270 in Armenia, V-179 in Russia, V-230 in Russia, Slovakia and Bulgaria) are under operation in several countries. The total time of these reactor operations amounts about 250 reactor-years and during this period there were no events leading to the radioactivity release out of the prescribed barriers. It demonstrates high safety of the indicated reactors reached due to effective engineering solutions, which were implemented at the design stage. The positive features of the first generation WWER-440 reactors from the safety point of view (low specific generation energy in the core, large water reserves in the primary and secondary circuits, etc.) were repeatedly noted by the IAEA experts during their missions on the relevant power plants.

At the same time, the first generation WWER-440 design was developed mainly according to the Russian safety standards, which were in force at the end of 60 - beginning of 70 years. The IAEA missions have determined a series of areas and aspects for which the WWER-440/230 design does not correspond to modern safety standards including standards accepted in the western countries. One of such areas is the accident analysis where the basic problem is the spectrum of the accidents considered in the design and adequacy of the analysis methodology. Taking that into account, two extensive projects on the accident analysis for first generation WWER-440 reactors were carried out in the frame of the TACIS program.

The specific purpose of these projects was the review of the available analyses and performance of the new accident analyses to obtain the systematic and modern set of the analyses according to the commonly accepted international practice. The accident analyses have been used to evaluate the real safety margins for existing NPPs, and also to evaluate necessity of the reactor protection system modification from the safety increase viewpoint. Novovoronezh NPP 3 and 4 units and Kola NPP 1 and 2 units were included in the projects.

All essential aspects for the accident analysis were determined on the basis of the modern Russian standards, IAEA standards and license requirements in France, Germany, Spain and Finland. The list of the initiating events has been developed. It included all events that have effect on the plant behavior and its safety performances. These events including operational disturbances (transients), design accidents, accidents with the reactor trip failure and other beyond design-basis accidents, were distributed on the classes. For each class of the initiating events the acceptance safety criteria were formulated and the analysis methodology was selected. Both Russian and modern western methodology and computer codes were used in the projects. For each initiating event the initial and boundary conditions were determined to ensure the conservative results with respect to safety acceptance criteria.

The accident analyses were performed using western codes RELAP5, CATHARE, WAVCO, HEXTIME/COBRA and Russian codes DINAMIKA, TECH-M. 60 initiating events were analyzed, for which more than 100 calculations of the emergency scenarios are performed during the projects. The analyses of the several selected cases were calculated using the different computer codes and/or by different users to ensure the representative results and the required quality of the analyses.

## 2. TACIS-91-1.3 Project

Different transients and accidents were analyzed in the frame of TACIS 91-1.3 project «Accident analysis for NPP with WWER-440/230». A wide spectrum of the initiating events was considered. These events result in the accidents of the following types:

- loss of coolant accidents (about 20 calculations were performed to cover leaks of the different size and locations, and also different operation of the emergency core cooling system);
- transients without loss of coolant (35 initiating events were investigated, and the analyses of the sensitivity and convergence of the result obtained using different codes are performed);
- beyond design-basis accidents;
- transients and accidents which require to consider the three-dimensional core physics;
- transients within containment;
- radioactivity release.

For loss of coolant accidents, the wide spectrum of the leaks from 32 mm up to 150 mm in the cold leg and 200 mm in the hot leg was analyzed. French approach for the analyses was adopted using CATHARE, TECH-M and RELAP5 codes. For these accidents the meeting of the acceptance criteria concerning to the peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, change of the core geometry and long-term core cooling has been considered. The analysis results have shown that all acceptance criteria are met for the leaks less than 120 mm in the cold leg and less than 150 mm in the hot leg. The existing reactor protection signals are sufficiently effective to trip the reactor and to activate the emergency core cooling system. For this type of the accidents, the dominant parameter is the emergency core cooling system flow rate, but not the protection signals. However, it was recommended to consider the necessity of the implementation of the additional reactor protection signals to improve the reliability of the reactor trip.

The analyses of the events without loss of coolant have been performed for the conditions, which envelop the main categories of the transients, accidents and beyond design-basis accidents. The existing acceptance criteria subject to test for these conditions were the peak cladding temperature, minimum departure from nucleate boiling ratio (DNBR), maximum pressure in the primary and secondary circuits. The RELAP5 and DINAMIKA codes were used for analysis.

Under all conditions, the margins of the DNBR and peak cladding temperature are sufficiently large to exclude the fuel rod damage. The margins for peak pressure in the primary and secondary circuits are also sufficiently large, except for some scenarios of the beyond design-basis accidents, where the primary pressure is rather close to the acceptable limit. The fuel melt temperature is not reached in any of the conditions considered.

As a whole, all the analyses have confirmed the conservatism of results, which were presented earlier in the design safety substantiation for WWER-440/230 reactors. The acceptance criteria are satisfied for all considered events using the western approaches and computer codes.

The analyses performed have allowed to propose a series of the measures for further safety improvement. In particular, it was found that the safety could be essentially improved, if some additional signals in the reactor protection system are implemented.

For example, for the “break of main feed water collector” case, the analysis results produced by DINAMIKA code have shown that the existing power control is sufficient to keep the plant parameters within the permissible limits. The calculations with RELAP5 have confirmed this conclusion, but shown some more drying of the steam generators (SG), before the reactor and turbine will stop. Therefore it was recommended to consider the effectiveness of the replacement of

the existing reactor protection signal of the third kind (EP3) on the reactor outlet temperature by the reactor protection signal of the first kind (EP1).

The accident analyses with three-dimensional core physics were performed according to the German methodology using HEXTIME code system. It was shown that the WWER-440/230 reactors have a large safety margin under all initiating events, so no modifications of the protection system are required.

Containment parameters under LOCA conditions were analyzed using WAVCO code. The analyses have shown that the safety criteria on the containment pressure are fulfilled under all the events considered in the project. However it was noted, the attention should be paid to the operability of the safety important equipment at high temperature appeared within containment. The interlock was also proposed to exclude unacceptable temperature increase in the boric acid emergency storage tank.

As a whole, on the basis of the analysis results it was recommended to consider the following additional protections and interlocks:

- low SG water level - reactor scram by EP1;
- high reactor outlet coolant temperature - reactor scram by EP1 instead of the existing EP3 signal;
- high pressurizer level - reactor scram by EP1;
- low primary pressure - reactor scram by EP1 instead of the existing protection signal of second kind (EP2);
- low pressurizer level - reactor scram by EP1;
- loss of all feed water pumps - closing of all turbine stop valves;
- low SG water level - activating of the emergency feed water pumps (EFWP);
- high temperature within boric acid emergency storage tank - activating of the spray system pumps for re-circulation.

The consideration of the possible effectiveness of these signals has been then performed in the frame of R2.02/96 TACIS project to evaluate necessity of their implementation on the operating units with WWER-440/230.

### **3. R2.02/96 Project**

In the R2.02/96 Project «Emergency protection signal effectiveness evaluation for WWER-230» the analyses have been performed taking into account the proposed signals to determine their influence on the safety margins concerning the acceptance criteria used. For each signal the list of design events was considered and those events were only selected for which these signals have the most essential influence. The choice of events to be analyzed has been based on the following:

- for event “loss all feed water pumps” the occurrence of low SG water level, high reactor outlet temperature, high pressurizer level and low primary pressure signals is assumed;
- increased margin, due to implementation of low SG level signal may ensure the desired additional requirement “SG feed water collector remains covered” after loss of all main feed water pumps (MFWP);
- DNBR margin for event “trip of two out of six main coolant pumps (MCP)” may be significantly increased by “high reactor outlet temperature” signal ;
- “high SG level” signal may be reached in the event “loss of all MFWP” and “trip 2/6 MCP”;
- “low primary pressure” signal is typical for event “pressurizer safety valve stuck open”;
- “low pressurizer level” signal is typical for loss of coolant accidents.

On the basis of the above consideration, the following list of events to be analyzed was defined:

- loss of main feed water pumps;
- trip of two out of six main coolant pumps;
- loss of turbine load;
- pressurizer safety valve stuck open;
- coolant leak of 100 mm diameter.

#### **4. Effectiveness of new signals**

For the initiating events mentioned above, the first reactor trip signal methodology was selected. The conservatism is ensured by the appropriate assumptions for the initial parameters uncertainties to the unfavorable side (for example: increased reactor power, decreased coolant flow rate, decreased primary pressure, decreased pressurizer level etc.). The calculations were performed using DINAMIKA, TECH-M, RELAP5 and CATHARE codes. The numerical values for new protection signals were assumed taking into account the measurement errors and other uncertainties.

##### *Loss of main feed water pumps*

The signals on low SG level for EFWP switching on and EP1 actuation have been investigated for “loss of main feed water pumps” event.

The EP1 signal on low SG level should increase the safety margin under loss of feed water flow rate. When analyzing this event, the effectiveness of this protection signal was evaluated on the increasing of DNBR margin and primary and secondary pressure margins.

The low SG level signal for EFWP actuation is purposed to reduce possibility of SG feed water collector uncover at the EFWP actuation time. This additional requirement is the criterion of evaluation of this signal effectiveness. The set point for EFWP actuation was selected as high as possible to prevent EFWP actuation during normal conditions including transients ones.

In the frame of project several scenarios of “loss of all MFWP” and also “loss of one MFWP were considered. Both existing signals and new additional signals were taken into account in these scenarios. With the conservative diversions of parameters, the failure of some systems are assumed in calculations (for example, emergency protection of the second, third and fourth kind on low SG level is not available; steam dump valve to the atmosphere and steam dump valve to turbine condenser are not available).

The calculation of “loss of all MFWP” event with new and all existing signals has shown, that new EP1 signal on low SG level was the first to appear and that the existing signal on high primary pressure was second. It was also found, that the automatic EFWP actuation slows down the SG depletion. One EFWP is also capable to compensate the flow rate through the SG safety valve, and also to remove residual heat during 20 minutes. After this time the level in steam generators begins to increase. From the minimum SG level point of view, reactor trip plays also an important role during the transient. For example, when the existing EP1 signal was assumed, SG collapsed level decreased up to 626 mm at the moment when the EFWP was manually actuated by operator (at 30 minutes from the beginning of the event). When automatic EFWP actuation was assumed, the collapsed level decreased up to 980 mm at the same time. When both new signals for EP1 and EFWP actuation were assumed, the collapsed level decreased only up to 1155 mm at EFWP actuation time. In the design, the upper point of SG feed water collector is at the level of 1120 mm from the SG bottom, therefore, new signals reduce the probability of the feed water collector uncover.

The results of “loss of one MFWP” analysis have shown, that the new EP1 signal on low SG level was not the first to come. Therefore it may be used as an additional signal, and the requirement “uncovery of the feed water collector” is fulfilled even without automatic EFWP actuation.

#### *Trip of two out of six main coolant pumps*

The EP1 signal on high reactor outlet temperature (or temperature in the hot legs) was investigated by analysis of the “trip of 2 out of 6 MCP” event. This signal was proposed instead of existing EP3 signal to increase DNBR margin and to avoid the primary circuit overheating. For this event, the increase of the primary and secondary pressure margins was also investigated.

To evaluate the effectiveness of the new signal, the calculations of this regime with existing protection signals and with new signals were performed. In the scenarios in which the existing protection signals were only taken into account, EP1 was initiated by the signal “high neutron flux”.

In the scenarios with the new EP1 signal “high reactor outlet temperature”, this signal was the first to appear and the reactor trip was initiated earlier approximately by 3 seconds. On the basis of the calculation results it was found, that the new signal increase the safety margins under loss of coolant flow conditions.

#### *Loss of turbine load*

The “loss of turbine load” event was analyzed to evaluate the effectiveness of two new EP1 actuation signals on high reactor outlet temperature and high pressurizer level. The EP1 signal on high pressurizer level was proposed for prevention of the coolant release through the pressurizer safety valves. For this event the calculations have been performed for scenarios taking into account existing signals and for scenarios with the new signals. In this case, the fulfilment of the acceptance criteria on the DNBR and primary and secondary pressure was considered.

For all the calculations, the first signal reached was the existing signal “high primary pressure”. The set points of new protection signals were not reached, so those new signals do not influence on the safety margins.

#### *Pressurizer safety valve stuck open*

For the “pressurizer safety valve stuck open” event the new signal for EP1 actuation on low primary pressure was proposed instead of existing EP2 signal. The conversion to the EP1 signal should improve reliability of the protection actuation during loss of coolant accidents.

For this event two scenarios were considered: (1) taking into account the new EP1 actuation signal on “low primary pressure”, and (2) taking into account the existing EP2 actuation signal on the same parameter. For both scenarios, the identical algorithm of the regime, assumptions and parameter uncertainties were assumed, except the set points for protection actuation. For the first scenario, it was accepted that EP1 will actuate when the primary pressure will decrease to 9,13 MPa. For the second scenario, the existing EP2 set point on low primary pressure 11,05 MPa (taking conservatively into account the uncertainty) was considered.

The calculation results have shown that the reactor trip was realized in the both cases and the non-crisis core cooling is ensured. The maximum value of the cladding temperature was 330 °C for both cases takes place in the steady-state condition. During the regime, the cladding temperature decreases. It was found that new EP1 actuation signal was reached approximately 50 seconds later than existing EP2 signal; however it activates the protection of the first kind, and it is more reliable for the reactor safety.

The analyses performed have shown that new signal of the reactor protection on low primary pressure does not increase the safety margins in the quantitative relation, but it improves the reliability of the reactor trip in case of accident.

#### *Coolant leak 100 mm diameter*

For the loss of coolant accidents, new EP1 actuation signal on low pressurizer level was proposed. For this purpose, the EP1 signal on the coincidence of low primary pressure and low pressurizer level is foreseen in the design. New EP1 signal on low primary pressure was proposed also. These new signals may increase the safety margins in loss of coolant accidents and improve reliability of the reactor trip functions.

For the evaluation of new signal effectiveness, the “coolant leak 100 mm diameter” event was analyzed. For this event three following scenarios were selected:

- with the existing EP1 signal (coincidence of low primary pressure and low pressurizer level);
- with new EP1 signal on low primary pressure;
- with new EP1 signal on low pressurizer level.

It was found that the maximum cladding temperature was 384 °C, and the minimum coolant inventory was 52 tons in the first scenario. In the second scenario, the reactor trip was realized on new signal “low primary pressure” 5 second later than in the previous case. However, no worsening of the temperature conditions in the core was observed in this case. In the third scenario, the reactor trip was realized on new signal “low pressurizer level” 1,5 seconds earlier than in the first case. As a whole, the transient was practically identical to the first scenario.

The analyses performed have shown that each proposed signal is sufficient to manage accident and to bring the plant to the safe state. New signals are practically equivalent to the existing signals in relation to the acceptance criteria meeting during the accident.

### **5. Implementation of the new signals**

For the loss of all feed water pumps event, new EP1 actuation signal on low steam generator level is reached the first. Earlier emergency protection actuation allows increase the safety margins in relation to the acceptance criteria. The signal on low SG level was not the first to come in case of loss of one feed water pump with the assumption of the control system failure. However this signal could be recommended as the only signal indicating this event. At the same time, new signal could back up the existing signals to reinforce the protection against loss of secondary heat removal. Therefore, the implementation of new EP1 signal on low steam generator level increases the NPP safety, so that its implementation at the operating reactors would be reasonable.

The implementation of the emergency feed water pumps actuation signal on low steam generator level has a positive impact on the events analyzed. This signal is reached earlier than EFWP actuation by operator in 30 minutes from the beginning of accident as it foreseen in the design. New signal makes the steam generator liquid level to decrease more slowly, and after approximately 20 minutes of the accident one EFWP is capable to remove the decay heat and restore SG level. The moment when the SG feed water collector has become uncovered is also delayed, and the additional requirement about feed water collector submerging is fulfilled at the moment of EFWP actuation. This will reduce the probability of the thermal shock. Therefore the implementation of this signal would be reasonable.

For the “trip of 2/6 MCP” event, the proposed EP1 signal on high reactor outlet temperature is reached earlier than the existing signals. The higher level of the reactor protection increases the

safety margins. Though the NPP safety is ensured without this signal, its use is obvious since this signal has preference in the signal formation if the reactor trip is necessary.

The EP1 signal on high pressurizer level has no effect on safety margin. In all considered cases, the existing EP1 signal “high primary pressure” appears earlier. New signal could be considered as duplicating one to improve reliability of the reactor trip. However there is a possibility of false actuation because of difficulty of the precise level measurement. This aspect should be taken into account when solving the question about implementation of the indicated signal at the NPP.

New proposed signal on low primary pressure is reached later than existing EP2 signal in case of pressurizer safety valve stuck open. The analyses performed have shown that both signals are capable to ensure the absence of departure from nucleate boiling and, therefore, to ensure the safety. Thus, new signal in this case does not increase the safety margins, but improves reliability of the reactor trip.

The analyses which have been performed for the coolant leak 100 mm have shown that both new EP1 signals on low primary pressure and low pressurizer level and existing signal are sufficient to ensure the safe accident course. However existing signal requires the coincidence of low primary pressure and low pressurizer level. New signals work on one parameter. New signal on low primary pressure improves reliability of the reactor trip for loss of coolant accidents, therefore it is reasonable to implement it at NPP. New signal on low pressurizer level has no significant advantages. This signal could be recommended as additional to back up the reactor trip for loss of coolant accidents. However it is necessary to take into account the difficulty of the precise level measurement when solving the question about the implementation of this signal.

Thus, the quantitative evaluation of the possible additional protection signals for WWER-440/230 was performed and the relevant recommendations about implementation of these signals at NPP were given. As a whole, modifications of the protection system realized at NPPs correspond to these recommendations. For example, new EP1 signals on low SG level, on high reactor outlet temperature, on low primary pressure, as well as EFWP actuation signal on low SG level are implemented at Novovoronezh NPP units 3 and 4. The same signals and also EP1 signal on low pressurizer level are introduced at Kola NPP units 1 and 2.

It could be also noted that all the considered signals except one EP1 signal on low SG level are implemented at Kozloduy NPP units 1-4. All six new signals indicated in section 4 are implemented at Bohunice NPP units 1 and 2. New EP1 signals on low SG level, on high pressurizer level, on low primary pressure and on low pressurizer level are realized at Metzamor NPP unit 2.

## **6. Conclusion**

In the area of the transient and accident analysis, the converging of the approaches used for WWER-440/230 and similar western reactors is continued. For the operating WWER reactors, the comparison of the existing accident analyses with modern western practice was performed with the IAEA assistance, and the relevant recommendations were developed.

A wide spectrum of transients and accidents was analyzed taking into account these recommendations to evaluate the safety of the operating WWER-440/230 units with the plant specific characteristics. The accident analysis was performed in the frame of 1.3 Project of TACIS-91 program using both Russian and western methodology and computer codes.

As a whole the analyses performed have confirmed the conservatism of results that have been represented earlier in the design safety assessment for WWER-440/230 reactors, and the sufficiency

of the existing reactor protection signals to ensure the safety of operating NPPs. However, it was also found that the set of protection signals and their set points could be improved. On the basis of the accident analysis results, the additional protections and interlocks have been recommended.

The quantitative evaluation of the effectiveness of proposed additional reactor protection signals was performed in the frame of the R2.02 Project of the TACIS-96 program. During the project, the necessary accident analyses were performed for identification and quantitative determination of the relevant benefits expected from the implementation of these signals. On the basis of the results obtained, the recommendations about necessity and possibility of the new protection signal implementation for operating reactors have been given. These recommendations were successfully used at the NPP units with WWER-440/230 reactor.