RELAP5/MOD3.2 Investigation of Loss of In-house Supply Power for VVER 1000/320V

by

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ABSTRACT

This paper discusses the results of the thermal-hydraulic investigations of the "Loss of In-house supply power" accident at Unit 6, Kozloduy NPP. The RELAP5/MOD3.2 computer code has been used to simulate the Loss of In-house supply power accident in a VVER 1000 Nuclear Power Plant model. This model was developed at the Institute for Nuclear Research and Nuclear Energy – Bulgarian Academy of Sciences for analyses of operational occurrences, abnormal events, and design basis scenarios. It will provide a significant analytical capability for the Bulgarian technical specialists located at the Kozloduy NPP. The criteria used in selecting transient are: importance to safety, availability and suitability of data followed by suitability for RELAP5 code validation. The investigation of "Loss of normal and reverse AC power" is a process that compares the analytical results obtained by RELAP5/MOD3.2 model of the VVER 1000 against experimental transient data received from Kozloduy NPP Unit 6. The comparisons between the RELAP5 results and the test data indicate good agreement.

1. Introduction

The reference power plant for this analysis is Unit 6 at Kozloduy site. This plant is VVER 1000 model V320 pressurized water reactor that produce 3000 MW thermal power and generates 1000 MW electric power. Main reason for the development of this report is to get confidence into results obtained by developed plant model for VVER1000/V320. The RELAP5 model configuration provides a detailed representation of the primary, secondary and safety systems in the Kozloduy NPP. The primary system has been modeled using four coolant loops, each including one main coolant pump and horizontal steam generator. The secondary system has been modeled using four steam lines and four horizontal steam generators.

Nodalization and input structure of primary and secondary systems was developed at Institute for Nuclear Research and Nuclear Energy.

Data and information for the modeling these systems were obtained from Kozloduy documentation and from the power plant staff.

2. Description of the model

The Baseline input deck for VVER1000/320V characteristics of all major systems and equipment of the Kozloduy NPP includes: reactor core, reactor vessel, MCPs, MIVs, SGs, steam line system and main steam header, emergency protection system, pressure control system of the primary circuit, makeup system, safety injection system, steam dumping devises (BRU-Ks, BRU-As, SG safety valves, PRZ safety valves), main feedwater system, emergency feedwater system.

3. Methods

The process of "Loss of In-house supply power" accident investigation by RELAP5 VVER1000 plant model starts with establishing steady state conditions and comparison of steady state conditions against the plant data.

The objective of the investigation is getting confidence which can be seen when the thermal hidroulic model predict the real plant parameters. These parameters will serve as "indicator" for determing whether RELAP5 code provides satisfactory results. For thermal-hidraulic models these parameters may include: temperatures, pressures, mass, flow rates, water levels in Pressurizer and Steam Generators, etc. In any case there are specific output information and specific output quantities of interest that provide the reason for performing the calculation.

4. Description of the transient

This section contains a description of the accident and the operator's actions manual and automatic, which occurs during the recovery.

Loss of In-house supply power (loss of normal and reserve AC power) appears due to failure in electrical grid and as a result of this the Unit loses 400 kV and 220 kV. In this case we have the following events: actuation of automatic step by step loading system (AASSL), start of DGs and all safety systems (HPPs, LPPs etc.). Turbine Stop Valves of the turbine closes and Turbine Generator switches off. In case of Loss of In-house supply power all four MCPs are switched off due to loss of AC power. Due to pump rundown there is a significant flow rate in next 3-4 minutes through the reactor core. This allows residual heat removing from the reactor core. In this time the operator has to start injection of boron concentration into the primary circuit. After reaching a safety boron concentration, the operator could start to cool down the reactor system by feed and bleed (in this case BRU-Ks are not available, too). Cooldown the reactor system by feed and bleed is possible after depressurization of primary circuit by YR system (gas removing system) down to 110 kgf/cm^2. This is also a condition for supporting the controlled Pressurizer water level. After performing of all these conditions cooldown continues by natural circulation.

In case of Loss of In-house supply power event the third group systems (which are not connected directly to the nuclear safety) remain without el. power. Such systems are the main coolant pumps, the auxiliary pumps, the condenser pumps etc. The second group of systems, which function is connected with nuclear safety, is supplied with power by the diesel generators. In case of losing of in- house supply power (on site power) the diesel generators start automatically and in 10 sec. (up to 15 sec. by instruction) the second group of systems is supplied with power. AASSL program starts.

Scenario:

The transient time was approximately 600 sec. The scenario is as follows:

1. The transient starts at 98% reactor power because of failure in the electrical grid.
2. 0.37 sec after the beginning of transient Turbine Stop Valves (TSVs) close - the Turbine stops.

3. Fast Load Coastdown System (FLCS) and Warning Protection (WP-1) switch on. It causes a reducing of Reactor Power down to 46%.
4. Main Coolant Pump #1 (MCP #1) and Make up pumps switch off. The AASSL program starts (see above).
5. MCPs #2, #3, and #4 switch off.
6. Main Feed Water Pumps switch off.
7. BRU-As open.
8. Reactor SCRAM by Frequency of electricity f < 46 HZ.
9. Stabilization on Natural Circulation.
10. High - High Pressure Pumps start injection of boron concentration.
11. Startting on the Emergency Feed Water Pumps by Steam Generator Water Levels signal.
12. All SGs are isolated by BZOK.
13. Depressurization of Primary side by YR line and YP line.
14. High pressure pump HPP - TQ23 starts.

5. Results

The plant transient event considered in this report can be categorized as of class of transients resulting from power plant equipment failure and perturbing the reactor core cooling due to losing of all four main coolant pumps.

The transient calculations are compared with the plant transient event data in Figure 1 through 7. An important parameter is the pressure in the primary circuit, since this parameter is input to many reactor control systems. Figure 1. presents measured primary pressure during the plant transient event and calculated one. As shown, the calculated parameter is almost identical to the measured one. Maximum core exit pressure of 15.88 MPa was reached at 4.0 sec during the plant event and in the same time in RELAP5 calculation - 15.90 MPa. Minimum pressure of 14.3 MPa was reached at 104 sec. in RELAP5 calculation and the same value is reached according to the plant data at 120 sec. At the end of transient time both pressures (the measured and the calculated one) are stabilized at approximately 15.24 MPa.

Comparison of secondary side pressure is presented in Figure 2. In plant data, maximum pressure of 7.44 MPa is reached at 12.6 sec. Approximately the same value 7.4 MPa is reached at 11.6 sec in RELAP5 calculation. After approximately 300 sec. the both curves are stabilized at level of 6.3 MPa. In RELAP5 calculation this became more smoothly.

Other very important characteristic is the water level in the Pressurizer. Comparison of this parameter is presented in Figure 3. In the first 100.0 sec the behavior of pressurizer water level curves are very close, almost identical. At 200.0 sec experimental results are 0.18 m higher (see Figure 3.). After 350 sec. transient time RELAP5 calculated Pressurizer water level is stabilized at level 6.15 m while transient plant data is 6.7 m. This difference could come due to RELAP5 calculated collapsed water level.

Figure 4 provides a comparison of Reactor Vessel Inlet / Outlet Pressure Difference.

The main coolant pump head curve is also an important parameter for the quality of the model predictions. The results from the experiment and RELAP5 calculations are compared in Figure 5.

Another important characteristic is the coolant temperature in the cold and hot legs. As it is seen from Figure 6., the calculation results closely follow the results obtained from the experiment. RELAP5 calculated hot leg temperature became minimum – 563 K at 92.0 sec and almost in the same time at 90.0 sec. hot leg temperature of plant data is minimum – 566.0 K. At the end of the transient time the difference between the calculated and measured hot leg temperature is less than 2 degrees. In case of RELAP5 calculation the hot leg temperature became 571.3 K and in the plant data is 573.0 K. The behavior of cold leg temperature is the same. At 12.7 sec. the calculated cold leg temperature reaches maximum of 566 K. Plant data cold leg temperature reaches its maximum of 564 K in 15.5 sec. At the end of the calculation the difference is less than 3 degrees. Figure 7 shows the steam generator secondary side RELAP5 calculated collapsed water level. These results are compared with the plant data. There are two curves of plant data concerning SG measured water level: L19 and L11. The figure indicates good agreement between the plant data of L19 for the first 200.0 sec and agreement with trend with L11 after 200.0 sec.

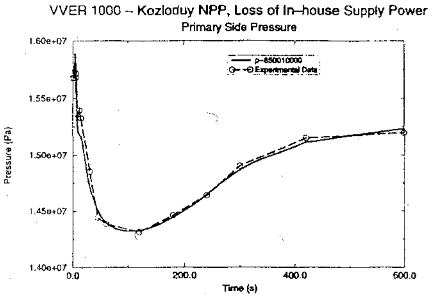

Figure 1.: Comparison of primary side pressure

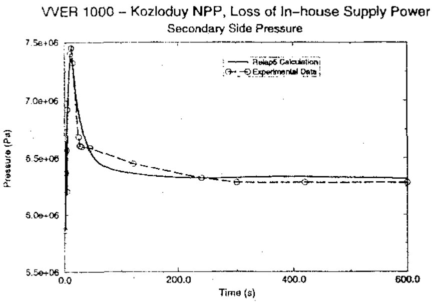

Figure 2.: Comparison of secondary side pressure

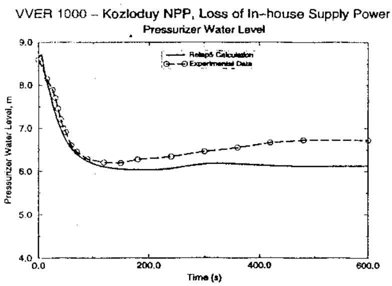

Figure 3.: Comparison of Pressurizer Water Level

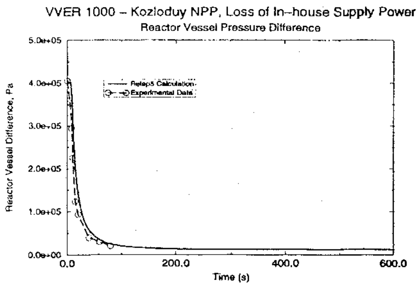

Figure 4.: Comparison of Reactor Vessel Pressure Difference