ANALYSIS OF THE NATURAL CIRCULATION STANDARD PROBLEM FOR KOZLODUY NPP UNIT 6

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ABSTRACT

The present article provides the brief description of the results of independent analysis of the Kozloduy NPP, Unit 6 Natural Circulation test standard problem.

“Investigation of the Natural Circulation” is a test done at Kozloduy NPP Unit 6 (with VVER-1000/320 reactor type) as important to safety for the NPP. The Investigation of Natural circulation calculations will be used for definition of a RELAP5 validation benchmark problem based on operational data from Kozloduy NPP.

The VVER-1000 RELAP5 model was used for the analysis of the natural circulation transient with RELAP5/MOD3.2 code.

The comparison of the test data versus calculated data demonstrates a reasonable agreement between the experimental data from the test and the RELAP5 results for all parameters analyzed.

1 INTRODUCTION

The purpose of this article is to describe briefly the main results of analysis of benchmark problems for application to VVER-1000 reactors with RELAP5 code. Operational data for the test “Investigation of the Natural Circulation” on Unit 6 from Kozloduy NPP are available for the purpose of assessing how the RELAP5 model compares against plant data. The criteria used in selecting this transient include importance to safety, availability and suitability of data, followed by suitability for RELAP5 code validation (according to a previously developed RELAP5 Code Validation Methodology). The benchmark problem selected will be defined and analyzed. The results from the RELAP5 analysis, based on a baseline model for VVER-1000/320, developed by Kiev Taras Shevchenko National University specialists will be compared against plant data.

The VVER-1000 RELAP5/MOD3.2 model has been developed for Kozloduy NPP Unit 6. The model was developed based on Zaporizhzhya NPP Unit 5 VVER 1000/320 RELAP5 model. Zaporizhzhya NPP Unit 5 is similar to Kozloduy Unit 6 on general design, set of main operation equipment as well as safety and control systems. The model being disposed was tested taking into account actual Kozloduy Unit 6 Data. As results of comparison shown, no essential differences have been detected.

Data and information for the modeling of VVER1000, Units 6 KNPP systems and components as well as natural circulation transient analysis were obtained from the reports:

- “Data Base for VVER1000, Safety Analysis Capability Improvement of KNPP (SACI of

KNPP) in the field of Thermal Hydraulic Analysis, BOA 278065-A-R4, INRNE-BAS, Sofia";
- "Kozloduy Nuclear Power Plant Safety Analysis Capability – Transient Analysis Code Assessment for VVER Reactors Project. Kozloduy NPP VVER-1000 Thermal-Hydraulics Standard Problem. Definition Report. BOA 278065-A-R4. Sofia, 2001".

The Kozloduy (Bulgaria) NPP Unit 6 VVER-1000 reactor was put into operation in 1991.

The VVER-1000 (V-320) reactor type is a pressurized light water reactor (PWR) with a thermal rating of 3000 MW and electrical output of 1000 MW. The Unit under consideration in this analysis is a typical V-320 model with four circulation loops, each including a main circulation pump and a horizontal steam generator. The steam generators are fed by means of two separate turbine-driven main feedwater pumps. Feedwater is delivered from pumps into each steam generator through feedwater collector and feedwater lines. All elements of the primary circuit are located in a steel-lined, cylindrical, prestressed concrete containment building.

In the VVER-1000 primary system, coolant enters into the reactor vessel though four inlet pipes associated with the four primary loops. The flow then passes into the downcomer between the reactor vessel and the inner vessel. The flow enters the lower plenum of the reactor vessel and passes through orifices in the bottom of inner vessel and then enters slots in the fuel support structures that lead directly to the fuel assemblies. The flow passes through the open bundles of the core. The fuel assemblies are in the configuration of a hexagon with each containing 312 fuel rods. There are 163 fuel assemblies of which 61 have control rods. After the reactor core, the flow moves into the upper plenum, which contains the shielding block, and then out to the hot legs of each of the four primary loops in the system.

2 VVER1000 RELAP5 MODEL GENERAL DESCRIPTION

The following systems are included into the model boundaries:
- reactor coolant system:
  - reactor VVER-1000 (and its internals);
  - main circulation pipelines;
  - main coolant pumps;
  - steam generators (i.e., manifolds, tube bundle and vessel volumes);
  - pressurizer system (i.e., the pressurizer itself, surge line, spray lines, including spray valves, pressurizer heaters, and safety valves);
- chemical and volume control system (i.e., make-up and let-down systems, including deaerator, pumps, and piping);
- main steam lines system (i.e., main steam lines and main steam header, turbine stop/control valves, fast acting steam isolation valves, cut-off valves, BRU-K);
- main feedwater system (i.e., pumps, main feedwater collector and main feedwater pipelines, MFW control valves, and cut-off/check valves);
- auxiliary feedwater system (i.e., pumps, auxiliary feedwater collector, piping, and valves);
- control and protection systems (i.e. reactor power controller ARM, reactor power limiter ROM, reactor partial trip, reactor protection system, pressurizer level controller, start-up/main/emergency FW controllers, BRU-K controller, electrical-hydraulic turbine control

system, as well as primary and secondary circuits setpoints and interlocks).

The following basic principles have been applied to the input model development:

- all major flow paths and heat structures of primary and secondary circuits should be modeled;
- components of the modeled facility with complex 3-D flow structure are represented by parallel channels with cross-connections between them to take into account flow mixing;
- the model should reflect the real geometry of the reactor system elements;
- a simplified approach is used to select technological systems of primary and secondary circuits for the modeling. As it is mentioned above, the extent to which a plant system is simulated depends on the problem being studied. Therefore, the set of systems such as Emergency Core Cooling System (ECCS), Steam Dump Valves to Atmosphere (BRU-A), Steam Generator Safety Valves (SG SV), Emergency Feedwater system is not presented in the model.

2.1 SYSTEM MODELING

The reactor vessel model (see Figure 2.1) includes 58 volumes: 9 in the merged annular downcomer (010); 2 in the lower plenum (020 and 022), including the lower core support region; 20 in the core heated length, including 10 in the single “hot” fuel assembly (032) and 10 in the rest of the core (030); 11 in the core bypass, including 5 in the flow paths through the baffle cooling channels and the gap between baffle and barrel (034), and 6 in the flow paths through the assembly central and guide tubes (036 and 044); 9 in the upper plenum (including the upper core unheated length) and the upper head (040, 042, 050, 052, 054, 062, 064, 090 and 092); 5 in the upper plenum bypass (080, 084 and 094), including the flow paths through the protective tubes of the control rods, temperature and neutron flux measurement channels; and 2 in the outlet nozzle region (072 and 074).

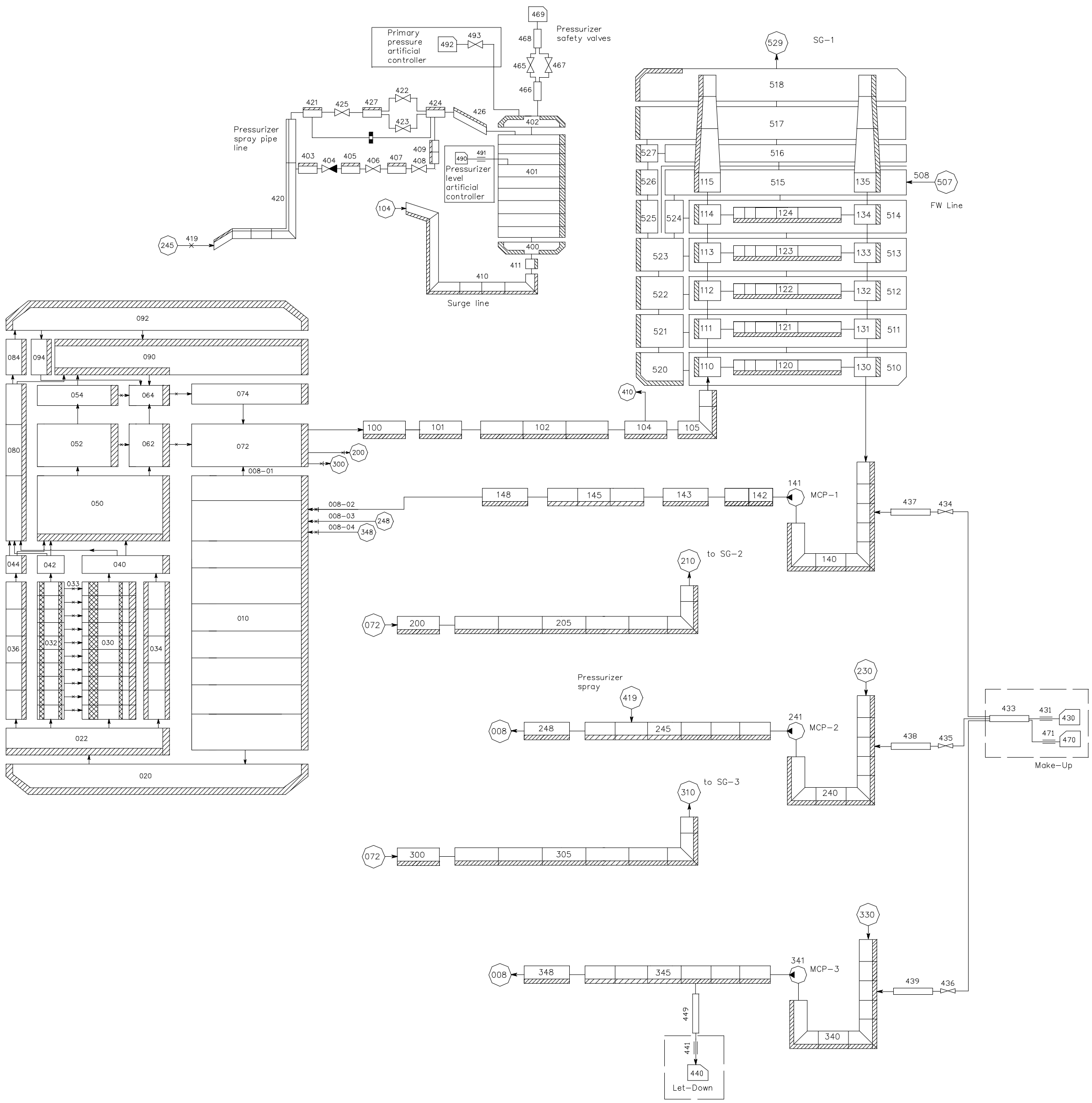

Figure 2.1 Nodalization Scheme of the Reactor VVER-1000 Primary Side

The core is modeled by two parallel “channels” weighted as 1:162 (the 1 (component 032) represents ‘’hot’’ fuel assembly and the 162 (component 030) represents rest of the 162 fuel assemblies). The model takes into account the flow mixing between the “channels” using the cross-flow junctions 033.

The middle of the upper plenum is modeled by two parallel “channels”. One of those “channels” (i.e., 052 and 054) simulates the internal volume, bounded by the perforated shell of the protective tube unit, while another one (i.e., 062 and 064) represents the external (annular) volume, bounded by the perforated shell and by the core barrel. The flow area of cross-flow junctions between above-mentioned channels is equal to actual area of the openings in the perforated shell.

The outlet nozzle region represents the vertical annular “channel”, bounded by the core barrel and pressure vessel. In the model this region is a set of the branch-type elements. The junction between control volumes 010-01 and 072 presents the flow path through the gap between the core barrel and the ring separating the downcomer and the upper plenum. The heat losses to the environment are modeled by corresponding heat structures.

Actual 4-loop reactor coolant system is modeled with 3-loop input deck (see Figure 2.1): two single loops representing, respectively, loop No.1 and loop No.2, and one double-capacity loop, which is used to simulate two remaining loops. Each model loop consists of the hot leg and cold leg. The heat losses to the environment are modeled by corresponding heat structures.

The model of Pressurizer system includes (see Figure 2.1):

- pressurizer;
- surge line;
- spray lines, including spray valves;
- pressurizer heaters.

The tube bundle and collectors of the horizontal steam generator (SG) are divided into the 5 horizontal layers arranging vertically (see Figure 2.1). Hot and cold collectors are represented, respectively, by elements X10…X15[1] and X30…X35 (here X=1,2,3 for loop No.1, loop No.2 and double capacity loop, respectively). Each layer of the heat exchange tubes X20…X24 was divided for the positive flow direction into 5 cells with a ratio 1:1:2:2:4. In vertical direction, each layer consists of 21 rows of the horizontal U-tubes except of lowest one (i.e., X20), which uses 26 rows.

The secondary side of SG (see Figure 2.2) was divided into four zones:

- the “hot” zone (X10...X14, where X=5,6,7 for, respectively, loop No.1, loop No.2 and double capacity SG, according to primary loops identification), including the collectors and tube bundle;
- the “downcomer” region (X20...X27), gathering different zones, where downflow is assumed;
- the volume X15 between the tube bundle and submerged perforated sheet, and region just above submerged perforated sheet X16, where location of the mixture level is assumed;
- the steam dome (X17 and X18).

Besides the U-tubes themselves and collectors, heat slabs representing the SG external walls are included in the model.

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[1] Here and below the sign "X" means 1st or 2nd (single), or 3rd (double capacity) loop, respectively.

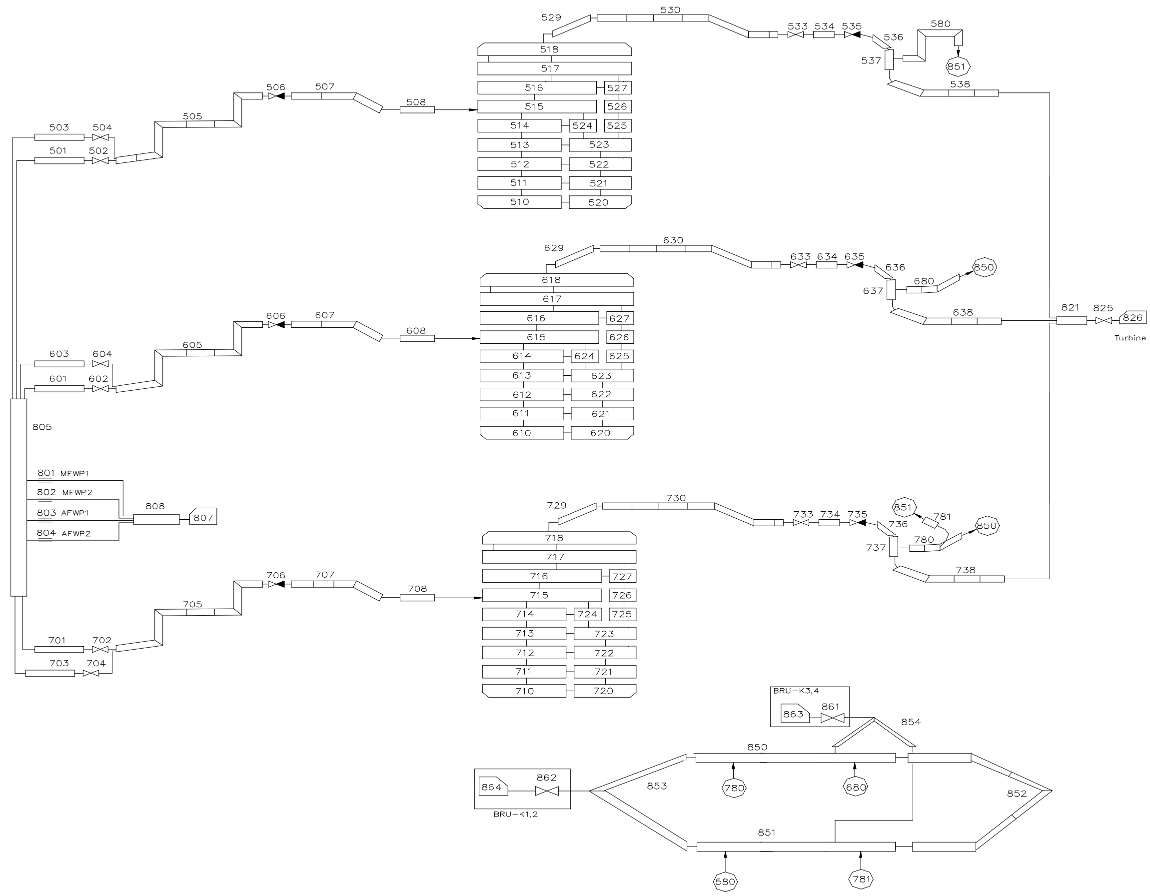

Figure 2.2 Nodalization Scheme of the Reactor VVER-1000 Secondary Side

The chemical and volume control system is simulated with two independent trains, providing make-up and let-down for the primary system (see Figure 2.1). The model components are described below.

Each train includes “time-dependent volume” 430, 440; “time-dependent junction” 431, 441; collector (“branch”) 433, 449; piping (“branch”) 437, 438, 439; control valves (“mtrvlv”) 434, 435, 436. The logic for the system operation is based on the pressurizer level maintenance.

The steam line system model, shown in Figure 2.2, represents the region from the outlet of the steam generators to the turbine stop-and-control valves. The model includes:

- piping of the main steam lines;
- fast-acting steam isolating valves (FASIV), and check valves;
- main steam header, and turbine bypass to condenser (BRU-K).

In accordance with primary loop nodalization, four real main steam lines at the plant are represented by 3-line input model: one single-capacity line (529…538) from loop No.1 steam generator (SG-1), one single-capacity line (629…638) from loop No.2 steam generator (SG-2), and one double-capacity line (729…738) from double capacity loop steam generator (SG-3).

Fast-acting steam isolating valve is modeled with “motor”-type valve X33 (where

X=5,6,7 for No.1, No.2 and double capacity steam line, respectively).

To simplify the turbine stop-and-control valves modeling, the steam lines are merged into single volume 821. The turbine stop-and-control valves themselves are modeled with “servo”-type valve 825. The turbine is represented by “time-dependent volume” 826.

The model components 850-854 represent the main steam header, including BRU-K surge lines. The BRU-K valves are modeled with double-capacity “motor”-typevalves (861, 862).

The model of the feedwater system from main and auxiliary feedwater pumps (MFWP, AFWP) through feedwater collectors and feedwater piping to steam generators is shown in Figure 2.2. Emergency Feedwater System was not presented in model because it was not used during natural circulation test.

The model includes:

- pumps;
- feedwater collectors (main and auxiliary);
- feed lines with control valves and check valves.

According to primary loop nodalization, four feed lines at the plant are represented by 3-line input model. The piping of each line is modeled with “pipe”-type elements X05...X08 (where X=5,6,7 for loop No.1, loop No.2 and double capacity loop feed line, respectively). The check valve is modeled with “check”-type valve X06 controlled by static pressure difference across the valve and flow rate.

The main feed controllers and the start-and-stop feed controllers are modeled with “motor”-type valve X02 and X04, respectively. The logic of the controllers is based on SG levels maintenance. In addition, a cut-off function has been assigned to the valves.

The modeling of the feedwater collector has been simplified to single volume 805. The feedwater pumps are modeled with “time dependent junctions”. Two main feedwater pumps (801, 802), representing two turbine-driven pumps, and two auxiliary feedwater pumps (803, 804), representing motor-driven pumps, supply the steam generators through the main (805) and AFW collectors and piping. The pump performance is represented by flow tables, which depend upon feedwater collector pressure.

The point kinetics model with reactivity feedback is used to calculate heat power generation in the fuel. To specify the feedback type, the “separable” option is applied. The fission product decay type is used.

The table of delayed neutron yield fractions versus decay constants (6 neutron groups) for fission of U^{235} by “thermal” neutrons is used. The tables of Doppler reactivity, moderator density reactivity and volume weighting factors are applied for the beginning of the 1st fuel cycle.

The following main models, representing plant controllers, have been developed and implemented:

- reactor power control and protection system model;
- model of make-up and let-down system;
- feedwater controllers’ models;
- model of electrical-hydraulic turbine control system;
- BRU-K controller model.

3 NATURAL CIRCULATION RUN RESULTS ANALYSIS

All available plant data on Natural circulation test results were used for analysis. The data (reactor unit parameters versus time) taken from the plant test plots, which have been used as basis for the analysis, are as follows:

- Primary Side Pressure;
- Cold Leg Temperature;
- Hot Leg Temperature;
- Pressurizer Water Level;
- SG No.1 Water Level;
- SG No.3 Water Level;
- MCP No.4 Pump Head;
- SG Primary Side Pressure Drop.

Plant data for all listed parameters, which have been registered during natural circulation transient, were provided within 600 seconds time period.

The sequence of events was modeled with the VVER-1000 RELAP Model for Kozloduy NPP Unit6 according to the plant data.

The calculation was performed during the 10 min (600 sec) transient time period. Before running the investigated transient the RELAP5 model was run with the real plant equilibrium conditions at 5% power. All main parameters of model have been stabilized very close to the test data (see Table 3.1).

3.1 TEST INITIAL CONDITION

Before the beginning of The Natural circulation test the Reactor Unit conditions were characterized as steady state with 5% Power.

The 5% power level was reached due to artificial reactor power controller operation under 5% power setpoint. In a 500 s (i.e. after stabilization of reactor parameters) artificial controller was switched off.

Measurement and calculation data characterized Reactor Unit initial conditions at the 5% Reactor Power (before the beginning of the natural circulation transient) are shown in Table 3.1.

Table 3.1. Reactor Unit Initial conditions at the 5% Reactor Power

Parameters	Plant data	RELAP5 calculation
Reactor power, MW	151	151
Primary side pressure (Core Exit), MPa	15.5	15.5
MSH Pressure, MPa	6.13	6.14
Pump Head, MPa	0.62	0.62
Pressurizer water level, m	5.2	5.2
Cold legs temperature, K	555.0	552.0

Parameters	Plant data	RELAP5 calculation
Hot legs temperature, K	556.5	553.4
Pressurizer steam temperature, K	617.2	617.5
SG water levels (4 m span), m	2.45	2.46
Temperature of Main Feed Water, K	434.8	434.8
Make-up/Let-down flow rates, m^3/h	30/30	30/30
Emergency feed water flow rates, m^3/h	160/131	160/131
SG pressure, MPa	6.13	6.14
Surge Line Fluid Temperature, K	559.5	559
Exit of the Assembly No. 09-32, K	555,3	554
Primary Side Boron Concentration, g/l	7.4	7.4
Elevation of Control Bank No. 8, cm (No. 9 and No. 10 are in the bottom)	100	100

As it follows from the table, the initial value of the hot leg and cold leg coolant temperatures in the test data is about 3 K higher than the initial value of the hot and the cold leg temperatures obtained in RELAP5 steady state run. All other parameters are in acceptable correspondence.

3.2 COMPARATIVE ANALYSIS OF NATURAL CIRCULATION RESULTS

The Data presented in Table 3.1 were used as initial conditions for the beginning of Natural Circulation transient.

The following parameters (which are available from the plant data) were used for comparison between plant measurements and RELAP5 code calculations:

- Primary side pressure;
- Hot and cold leg temperature;
- Pressurizer water level;
- SG water levels;
- MCP heads;
- SG primary side pressure drop.

The run results are compared with the Plant data in Figure 3.1 through Figure 3.6.

The measured primary pressure and the calculated one are represented in Figure 3.1. As a whole, the calculated parameter is in good agreement with the measured one. All characteristics of measured pressure variation are present on the calculated curve. Pressurizer spray valve opening (after the cold leg spray set point reaching at about 150.0 sec) resulted in delay of primary pressure increase. The low efficiency of the spray injection is explained by the cut-off of MCPs. The maximum pressure of 16.43 MPa was reached at 265.0 sec in the RELAP5 calculation versus 16.58 MPa at 270.0 sec of an experimental data. In a 500 sec (approximately) after the beginning of the transient the pressure is stabilized at a new range of 16.0 MPa in both cases the RELAP5 calculation and the test.

It is possible to explain some differences in maximum pressure by differences in maintaining of a 5% reactor power and primary Make-up operation.

The similarity in a behavior of primary pressure (calculated and measured values) is very important, since this parameter is input to reactor control and safety systems.

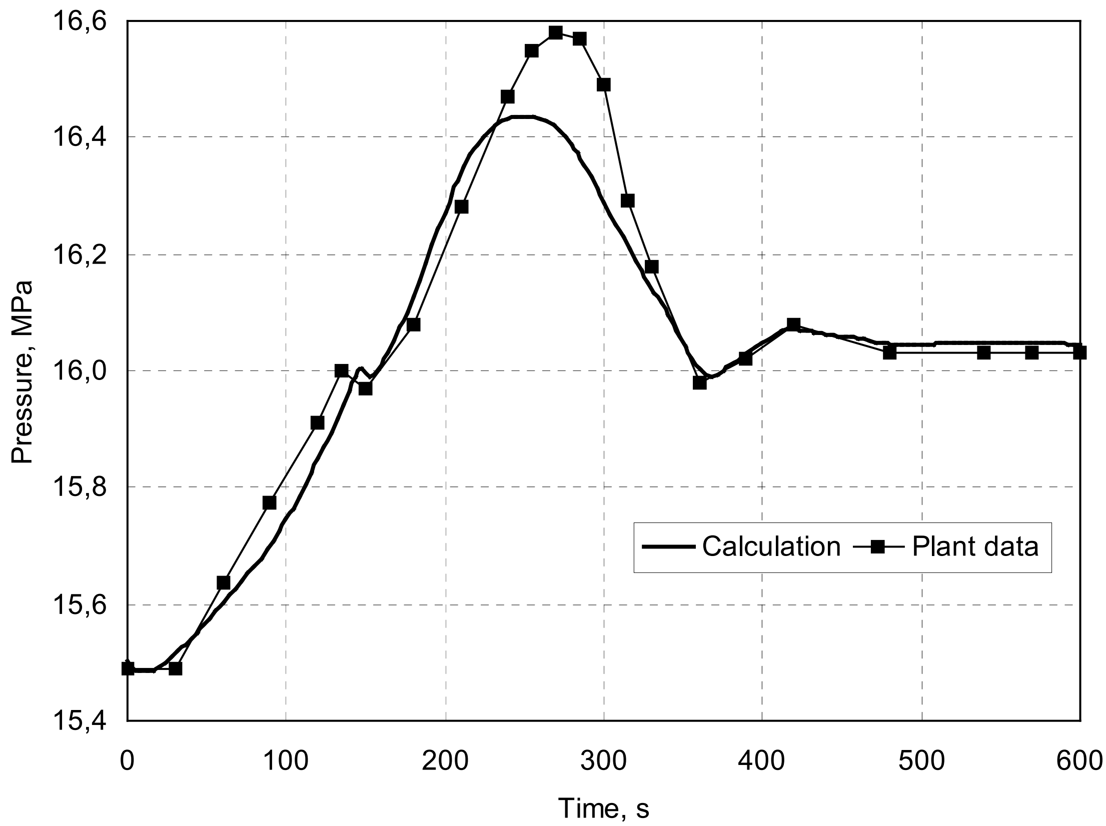

Figure 3.1 Primary Pressure (cntrlvar 3202)

The hot leg and cold leg coolant temperatures are presented in Figure 3.2.

As it follows from the figure, in whole, qualitative and quantitative similarity of temperature curves is presented. Some differences in hot leg temperatures within time interval 150-300 sec can be explained by differences in maintaining of a 5% reactor power.

As it was mentioned above, the initial value of the hot leg and cold leg coolant temperatures in the test data is about 3 K higher than the initial value of the hot and the cold leg temperatures obtained in RELAP5 run. As additional runs shown, it is impossible to increase primary temperature by increasing of secondary pressure (within acceptable ranges) or by changing of any other parameters because of low sensitivity. At the same time, the deviations of measured and calculated results are within 1.5% of the design range.

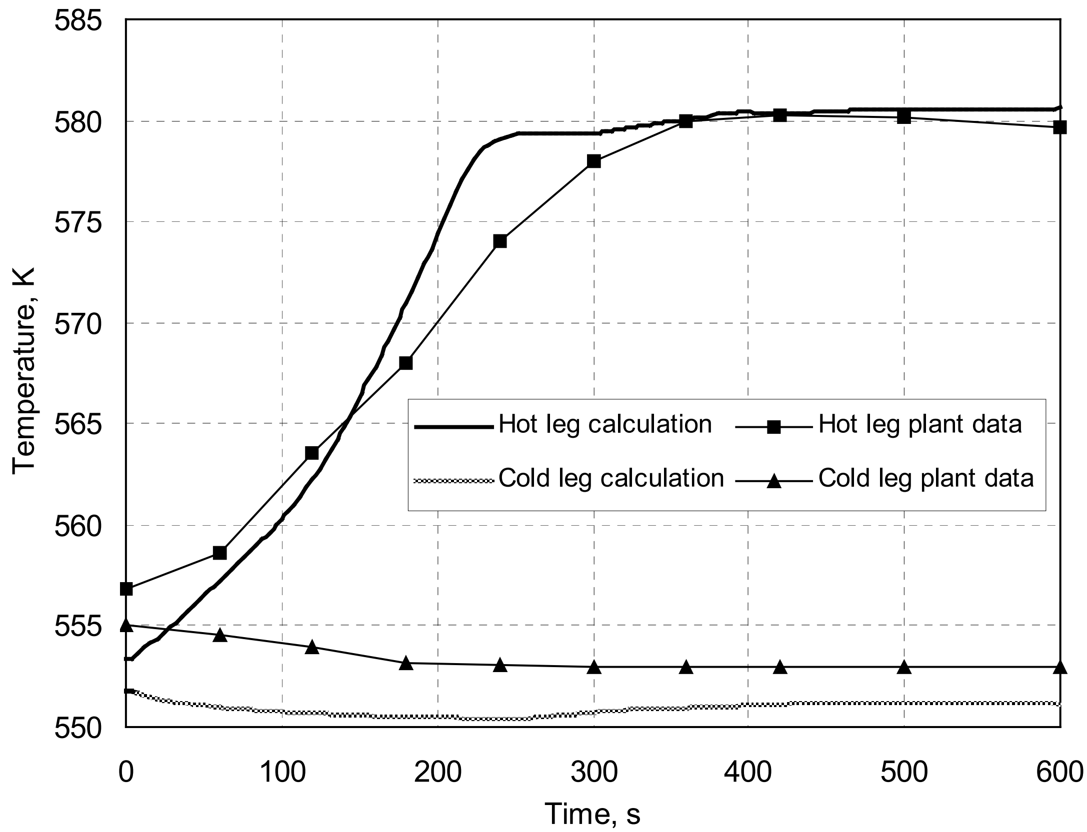

Figure 3.2 Hot and Cold Leg Temperature (cntrlvar-6243, 6246)

The pressurizer level (RELAP calculation and Plant Data) is shown in Figure 3.3. As it follows from Figure, the trends (time history of increase and decrease) of calculated and measured level are almost identical. Some difference in the values can be explained by differences in maintaining of primary Make-up.

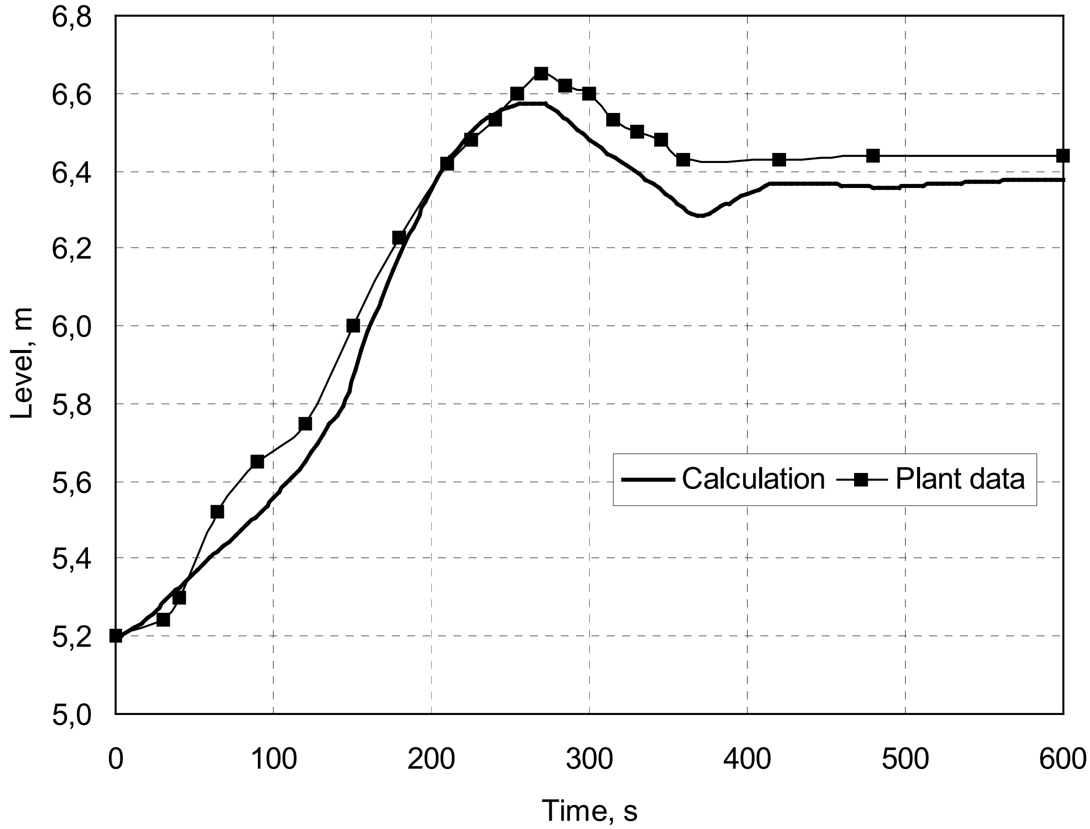

Figure 3.3 Pressurizer Level (cntrlvar-2405)

The steam generator water level (RELAP calculation and Plant Data) is compared in Figure 3.4. There is a good agreement between the plant data and the RELAP5 calculation results. Calculated curves are very close to the curve of SG-1 level. Some difference in the shape of curves can be explained by differences in steam lines trace.

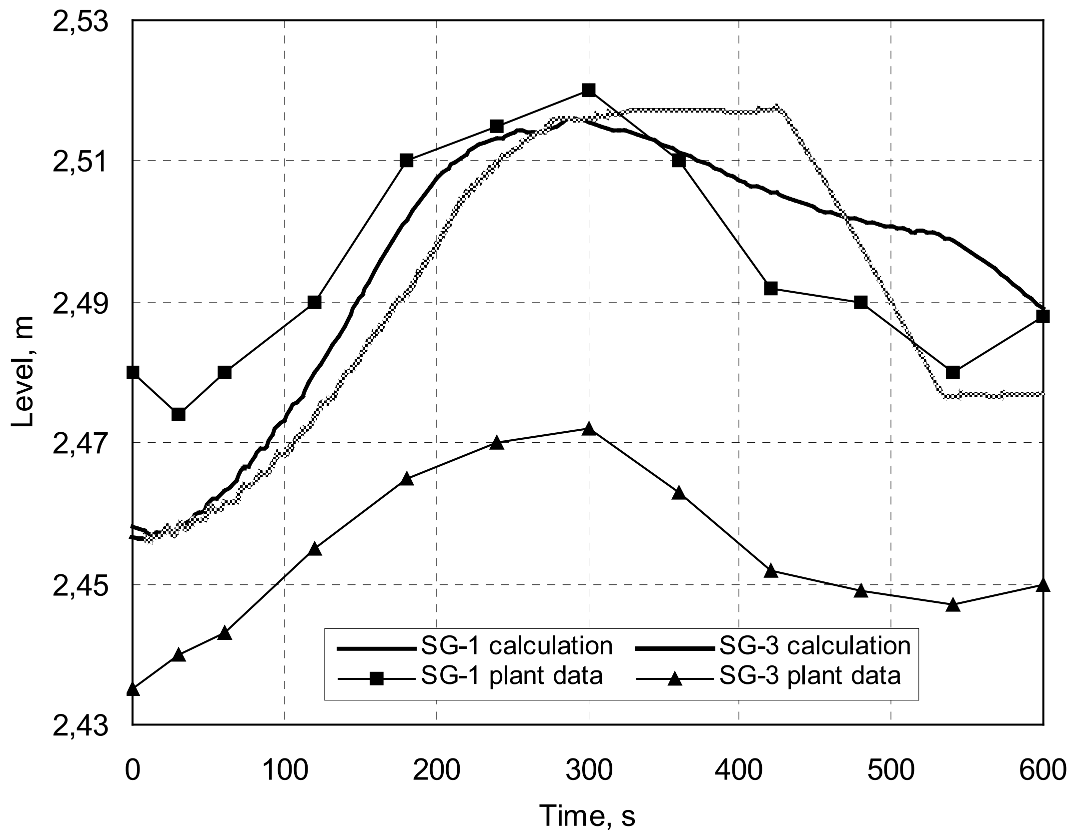

Figure 3.4 Steam Generator Levels (cntrlvar-2521, 2721)

The main coolant pump cost down happens according to design homologous characteristics. The results of the Plant test and RELAP5 calculation are presented in Figure 3.5. As it follows from Figure, there is a good agreement between the plant and the calculation MCP head.

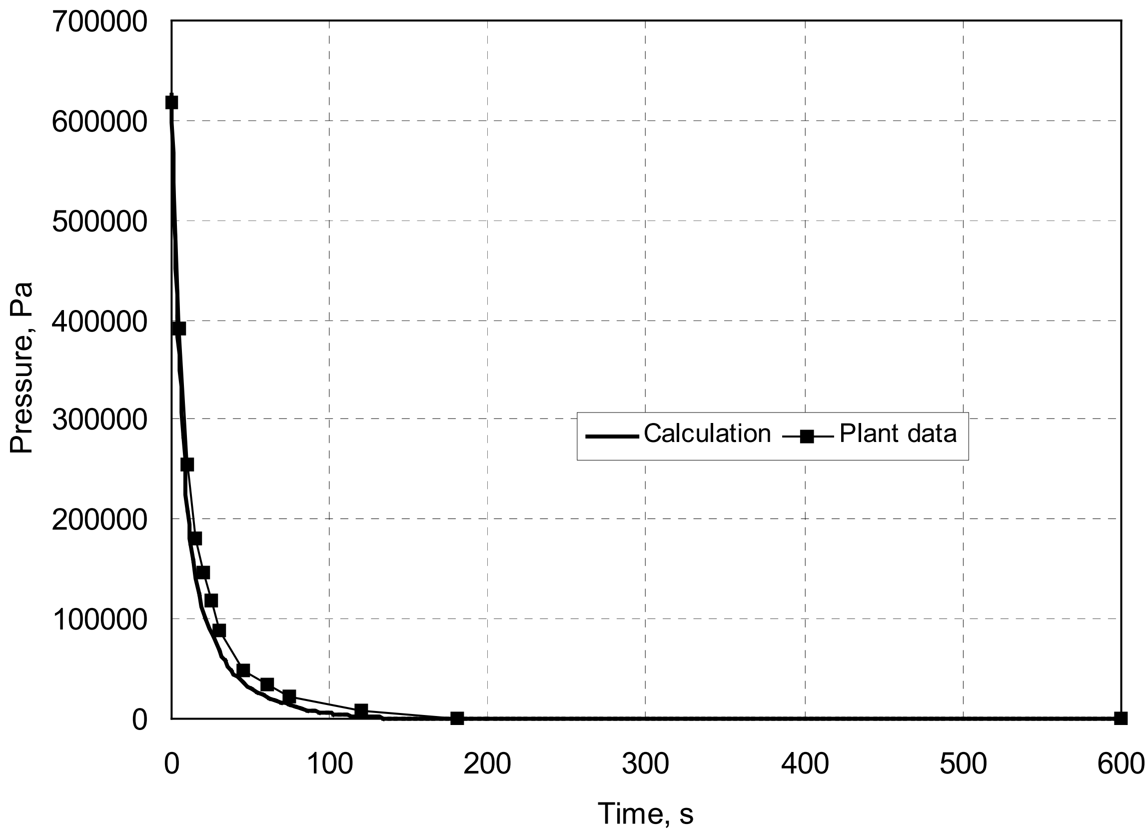

Figure 3.5 Main Coolant Pump Head (cntrlvar-7180)

Figure 3.6 provides a time history of SG Primary Side Pressure Drop. The curves presented in this Figure demonstrates a good agreement between the plant data and the RELAP5 calculation.

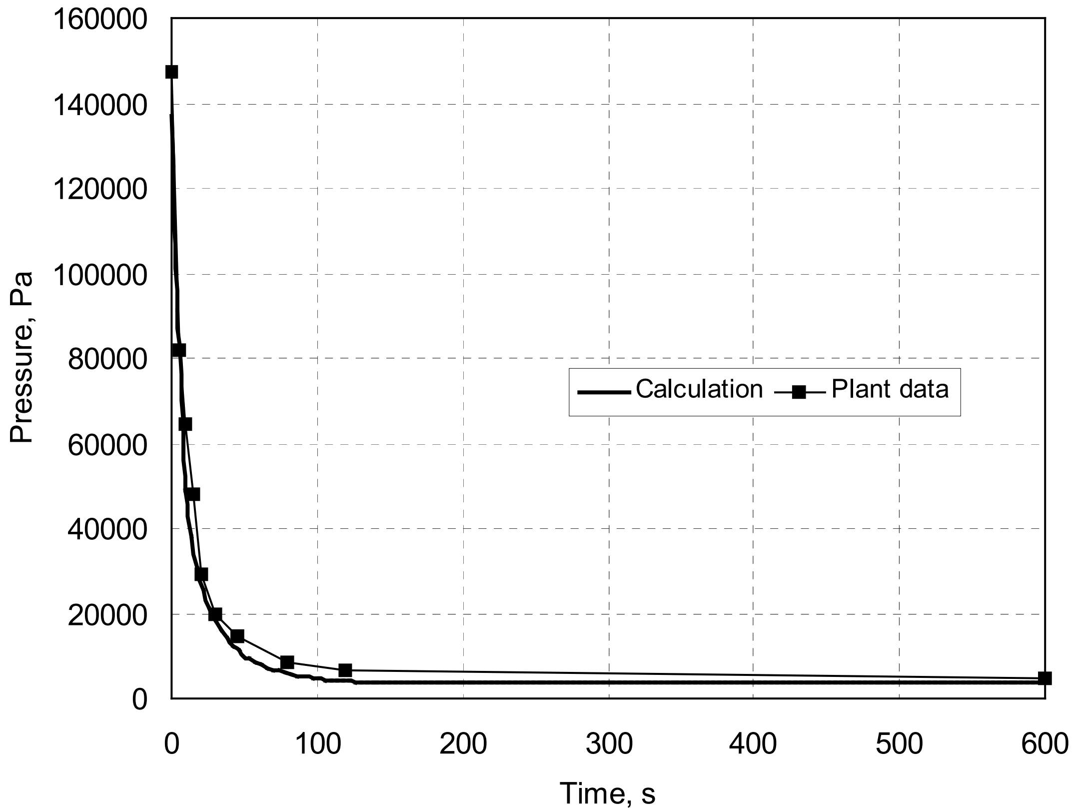

Figure 3.6 Steam Generator Primary Pressure Drop (cntrlvar-7122)

4 CONCLUSIONS

The RELAP5 model developed for the transient analysis of the performance of VVER-1000 Reactor Unit has been used to predict the results obtained during the Kozloduy NPP, Unit 6 Natural Circulation test.

Generally, the comparative analysis for the Kozloduy NPP, Unit 6 natural circulation test demonstrates a reasonable agreement between the RELAP5 calculation results and the Plant test data.

All characteristics of measured parameters variation (increase/decrease time history, maximum/minimum location) are present on calculated curves. The deviations of measured and calculated results are within design ranges.

Some difference in the shape of curves as well as maximum and minimum values of parameters (primary pressure, pressurizer level, etc.) can be explained by differences (uncertainties) in maintaining of a 5% reactor power and primary Make-up operation.

At the same time it is necessary to note that the initial value of the hot leg and cold leg coolant temperatures in the test data is about 3 K higher than the initial value of the hot and the cold leg temperatures obtained in RELAP5 run. Additional runs shown that it is impossible to increase primary temperature by means of other parameters changing (including a secondary pressure increasing) within acceptable ranges because of low sensitivity. The necessary increasing of primary coolant temperature can be obtained by means of primary-to-secondary heat transfer deterioration by a decreasing of heat transfer surface of SG tubes bundle or by use of fouling factor for SG tubes. Obviously, that both approaches are unacceptable because we deal with the beginning of Unit commissioning. It means that all steam generators were new, i.e. we can’t involve the decreasing of heat transfer surface of SG tube bundle as result of closing (by means of stubs) of some part of SG tubing during annual scheduled repair (up to 5% from total is permitted in case of primary-to-secondary small leak). For the same reason we cannot consider possible fouling of SG tubing surface because all heat transfer surfaces are clean at the beginning of the SG life.

As it follows from above, there is some redundant efficiency of primary-to-secondary heat transfer under natural circulation with low reactor power.

The results of comparative analysis are an important part of the validation of the RELAP5/MOD3.2 code.

The comparison demonstrates that RELAP5 predicts the test results with acceptable accuracy, i.e. within design ranges of uncertainty.

The overall conclusion is that RELAP5/MOD3.2 is adequate to simulate the transient phenomena occurring in a VVER-1000 under Natural Circulation conditions.