TRACE VVER-440/V-213 Model Cross-Code Validation

Prepared by:
S. Iegan*, Y. Vorobyov**, O. Zhabin**, S. Yanovskiy**

*State Nuclear Regulatory Inspectorate of Ukraine
9/11 Arsenalna Str.
Kyiv, 01011 Ukraine

**State Scientific and Technical Center for Nuclear and Radiation Safety of Ukraine
35-37 V. Stusa Str.
Kyiv, 03142 Ukraine

K. Tien, NRC Project Manager

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
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ABSTRACT

This report is developed by the State Nuclear Regulatory Inspectorate of Ukraine (SNRIU) and its technical support organization, the State Scientific and Technical Center for Nuclear and Radiation Safety of Ukraine (SSTC NRS), under Implementing Agreement On Thermal-Hydraulic Code Applications And Maintenance Between The United States Nuclear Regulatory Commission and State Nuclear Regulatory Inspectorate of Ukraine (signed in 2014) in accordance with Article III, Section C, of the Agreement.

The report provides results of the comparison calculations conducted with application of SSTC NRS model of VVER-440/V-213 for TRACE and RELAP5 computer codes. The calculation scenarios analyzed include design basis accidents and transients from several initiating event groups usually evaluated in safety analysis reports.
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EXECUTIVE SUMMARY

This report is developed in the framework of the Implementing Agreement on Thermal-Hydraulic Code Applications and Maintenance between United States Nuclear Regulatory Commission and the State Nuclear Regulatory Inspectorate of Ukraine.

At the previous stages of these activities existing RELAP5 model for VVER-440 was converted to TRACE code format and set of validation calculations based on actual incidents were performed.

This work is aimed at the comparative TRACE and RELAP calculations for selected design basis accident scenarios. Thus, this report contains numeric analyses results of the following initiating events:

- guillotine break of the main steam header;
- loss of vacuum in the condenser of one of the turbines;
- trip of 4 out of 6 reactor coolant pumps;
- uncontrolled withdrawal of control group of control assemblies from the reactor core with a normal operating speed of 20 mm/s;
- break of the pressurizer surge line.

Comparison of the results obtained with TRACE and RELAP5 models indicates some differences in calculated parameters. In particular, the differences in the primary circuit pressure that were observed in some of the scenarios are caused by different mathematical models and correlations for steam condensation, which are used in the special PRESSURIZER model in TRACE, as compared to the pressurizer modelling in RELAP5. Cladding temperature differences are related to the specifics of the heat structure modeling approach and the absence of TRACE correlation options, which does not allow more precise model adjustment to ensure complete convergence with the relevant RELAP5 models.

The results of cross-code validation calculations demonstrate that developed VVER-440/V213 thermal-hydraulic model for TRACE code is able to reproduce adequately the plant response to transients and accidents without core melt that were calculated previously in safety analysis reports using the RELAP5 model. For the majority of plant parameters good correspondence between TRACE and RELAP5 results is obtained.
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<th>Description</th>
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<tr>
<td>AFW</td>
<td>Auxiliary Feedwater System</td>
</tr>
<tr>
<td>ARM</td>
<td>Reactor Power Controller, Russian designation</td>
</tr>
<tr>
<td>BRU-A</td>
<td>Steam Dump Valve to Atmosphere</td>
</tr>
<tr>
<td>BRU-K</td>
<td>Turbine Bypass to Condenser</td>
</tr>
<tr>
<td>CAMP</td>
<td>Code Maintenance and Assessment Program</td>
</tr>
<tr>
<td>DBA</td>
<td>Design Basis Accident</td>
</tr>
<tr>
<td>DG</td>
<td>Emergency Power Supply Diesel-Generator</td>
</tr>
<tr>
<td>ECCS</td>
<td>Emergency Core Cooling System</td>
</tr>
<tr>
<td>EFW</td>
<td>Emergency Feedwater System</td>
</tr>
<tr>
<td>FASIV</td>
<td>Fast-acting Steam Isolation Valve</td>
</tr>
<tr>
<td>HA</td>
<td>Hydroaccumulators</td>
</tr>
<tr>
<td>HPIS</td>
<td>High Pressure Injection System</td>
</tr>
<tr>
<td>IE</td>
<td>Initiating Event</td>
</tr>
<tr>
<td>LOCA</td>
<td>Loss of Coolant Accident</td>
</tr>
<tr>
<td>LPIS</td>
<td>Low Pressure Injection System</td>
</tr>
<tr>
<td>MFW</td>
<td>Main Feedwater System</td>
</tr>
<tr>
<td>MSH</td>
<td>Main Steam Header</td>
</tr>
<tr>
<td>MSIV</td>
<td>Main Steam Isolation Valve</td>
</tr>
<tr>
<td>MSL</td>
<td>Main Steam Line</td>
</tr>
<tr>
<td>NPP</td>
<td>Nuclear Power Plant</td>
</tr>
<tr>
<td>PRZ</td>
<td>Pressurizer</td>
</tr>
<tr>
<td>RCP</td>
<td>Reactor Coolant Pump</td>
</tr>
<tr>
<td>RCS</td>
<td>Reactor Coolant System</td>
</tr>
<tr>
<td>RNPP</td>
<td>Rivne Nuclear Power Plant</td>
</tr>
<tr>
<td>RPL</td>
<td>Reactor Power Limiter</td>
</tr>
<tr>
<td>SG</td>
<td>Steam Generator</td>
</tr>
<tr>
<td>SLP</td>
<td>Sequential DG Loading Program</td>
</tr>
<tr>
<td>SNRIU</td>
<td>State Nuclear Regulatory Inspectorate of Ukraine</td>
</tr>
<tr>
<td>SRV</td>
<td>Safety Relief Valve</td>
</tr>
<tr>
<td>SSTC NRS</td>
<td>State Scientific and Technical Center for Nuclear and Radiation Safety</td>
</tr>
<tr>
<td>USNRC</td>
<td>United States Nuclear Regulatory Commission</td>
</tr>
<tr>
<td>VVER</td>
<td>Pressurized Water Reactor, Russian design</td>
</tr>
</tbody>
</table>
1 INTRODUCTION

At the end of 2014 the United States Nuclear Regulatory Commission (USNRC) and the State Nuclear Regulatory Inspectorate of Ukraine (SNRIU) signed Implementing Agreement On Thermal-Hydraulic Code Applications And Maintenance (CAMP). In accordance with Article III, Section C, of the Agreement, SNRIU shall submit to the USNRC the in-kind contribution reports providing the code assessment results or other activities results of equivalent value.

In the framework of the Agreement SNRIU and SSTC NRS obtained the state-of the-art TRACE code [1], [2] which provides advanced capabilities for modeling thermal-hydraulic processes and components, control systems and allows coupling with PARCS neutron kinetics code. In 2015 SSTC NRS initiated activities on TRACE code application for evaluation of the results of safety assessments performed for Ukrainian NPPs.

As the first step of these activities, the existing SNRIU/SSTC NRS RELAP5 model for VVER-440 was converted to TRACE code format.

In order to justify capabilities of VVER-440 model for TRACE code to simulate adequately the plant response during transients, calculations of several events that had actually occurred at Ukrainian NPPs were conducted. Results of TRACE simulations (validation) of these events in comparison to the plant measured data are provided in NUREG/IA-0485 [3]. Since these events do not cover all the phenomena expected to occur during accidents, the VVER-440 input model validation activities were extended by performing the comparative calculations of selected design basis transient and accident scenarios with RELAP5 and TRACE model. This report provides the results of comparative calculations.

Section 2 of the report briefly describes the validation process and refers to the description of the main primary and secondary systems of Rivne NPP (RNPP) unit 1 (VVER-440/V-213 design) which are important for development of thermal-hydraulic model, as well as to a description of RNPP Unit 1 model.

The results of comparative TRACE and RELAP calculations for selected design basis accident (DBA) scenarios are provided in Section 0 of the report.

For each scenario the following information is provided:

- brief description of the scenario;
- initial and boundary conditions selected for calculation;
- sequence of events;
- description of calculation results;
- plots of the main primary and secondary circuit parameters.
2 BRIEF DESCRIPTION OF MODEL VALIDATION PROCESS

After preparation of RNPP unit 1 model for TRACE code and adjustment of steady state calculation several transient calculations were performed simulating the actual incidents that had actually occurred at Rivne NPP and the results of calculations were compared with the plant measurement data. In particular the following incidents were simulated:

- reactor scram caused by concrete slab drop to the connection lines of house loads power supply transformer;
- reactor scram transient initiated by 6 kV switch short circuit;
- inadvertent reactor scram.

The results of these validation calculations, as well as a brief description of the model and of the main VVER-440/V-213 design features are presented in NUREG/IA-0485 report [3]. In general, the results demonstrate that calculated behavior of the main primary and secondary circuit parameters is in good agreement with the plant measured data.

However, since simulated incidents do not cover all phenomena that are important for accident analysis and allow to check correctness of modelling for the limited number of plant systems only it was decided to extend TRACE model validation by performing comparative calculations of several scenarios with TRACE and RELAP codes. For this purpose the following DBA scenarios at full power operation are simulated:

- main steam header (MSH) break;
- loss of turbine condenser vacuum;
- trip of 4 out of 6 reactor coolant pumps (RCP);
- uncontrolled withdrawal of control rods group;
- pressurizer (PRZ) surge line break.

These scenarios cover the majority of DBA initiating events groups, including loss of coolant accidents and secondary circuit breaks that can not be evaluated otherwise due to lack of correspondent plant incidents or reliable measured data.

For each scenario the identical initial and boundary conditions were specified for TRACE and RELAP calculations, and the results obtained with these models were compared.
3 RESULTS OF COMPARATIVE CALCULATIONS

3.1 MSH Break

3.1.1 Brief Description of Initiating Event

This initiating event (IE) assumes postulated guillotine break of the main steam header with a diameter of 465×16 mm, that leads to a sharp increase of heat removal by the secondary circuit. According to the expected frequency of occurrence the IE is categorized as a design basis accident.

3.1.2 Initial Conditions

The initial conditions selected for calculation correspond to normal full power plant operation (taking into account allowances due to plant control systems operation) and are presented in Table 3-1.

Table 3-1 Results of Steady State Calculation

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Units</th>
<th>Design value</th>
<th>Calculated value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core thermal power</td>
<td>MWt</td>
<td>1375±27</td>
<td>1402</td>
</tr>
<tr>
<td>Reactor outlet pressure</td>
<td>kgf/cm²</td>
<td>125</td>
<td>124</td>
</tr>
<tr>
<td>Reactor inlet temperature</td>
<td>°C</td>
<td>267</td>
<td>267</td>
</tr>
<tr>
<td>Coolant temperature at reactor outlet</td>
<td>°C</td>
<td>297.9</td>
<td>297</td>
</tr>
<tr>
<td>Coolant heating in the reactor</td>
<td>°C</td>
<td>30.9</td>
<td>30</td>
</tr>
<tr>
<td>Maximum temperature of fuel cladding external surface</td>
<td>°C</td>
<td>335</td>
<td>327</td>
</tr>
<tr>
<td>Reactor coolant flow rate</td>
<td>m³/h</td>
<td>40600±400</td>
<td>40200</td>
</tr>
<tr>
<td>PRZ level</td>
<td>m</td>
<td>5.96</td>
<td>5.97</td>
</tr>
<tr>
<td>SG pressure</td>
<td>kgf/cm²</td>
<td>47</td>
<td>45.9-46.5</td>
</tr>
<tr>
<td>SG level</td>
<td>m</td>
<td>2.105</td>
<td>2.1-2.11</td>
</tr>
<tr>
<td>SG steam production</td>
<td>t/h</td>
<td>450</td>
<td>444 - 467</td>
</tr>
<tr>
<td>SG feedwater temperature</td>
<td>°C</td>
<td>223</td>
<td>223</td>
</tr>
<tr>
<td>Water temperature in ECCS tanks</td>
<td>°C</td>
<td>55-60</td>
<td>60.0</td>
</tr>
<tr>
<td>Water temperature in ECCS HA</td>
<td>°C</td>
<td>60.0</td>
<td>60.0</td>
</tr>
</tbody>
</table>
3.1.3 Boundary Conditions

The following assumptions on systems availability and configuration are selected in the analysis.

The break occurs at MSH semi-header connected to turbine No.1 and is located in close proximity to fast-acting steam isolation valves (FASIV) separating MSH into two semi-headers. Instant opening of the valves simulating MSH break is modeled.

No operator actions are considered.

Operation of reactor power controller (ARM) and reactor power limiter (RPL) to decrease power is not accounted. At the moment of IE occurrence ARM operates in "T" mode (MSH pressure maintenance), and automatic switching to "N" mode (neutron power maintenance) is not considered.

Conservatively, loss of normal power supply is postulated to occur simultaneously with reactor scram. Operation of Level 4, Level 3 and Level 2 emergency reactor protection is not taken into account.

A failure of one out of three emergency feedwater (EWF) pumps (namely, EFWP-1) to start is postulated as a single failure.

3.1.4 Calculation Results

Sequence of events for this accident is presented in Table 3-2.
Table 3-2  Sequence of Events for MSH Break Accident

<table>
<thead>
<tr>
<th>TRACE Time, s</th>
<th>RELAP5 Time, s</th>
<th>Event</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0</td>
<td>0.0</td>
<td>Guillotine break of MSH semi-header No.1</td>
<td>Start of double-ended steam discharge from MSH</td>
</tr>
<tr>
<td>0.07</td>
<td>0.07</td>
<td>Maximal cladding temperature 330 °C (TRACE) and 334 °C (RELAP5)</td>
<td>MSH-1 pressure decrease to 38 kgf/cm²</td>
</tr>
<tr>
<td>0.1</td>
<td>0.1</td>
<td>Closure of turbine no.1 stop valves</td>
<td>MSH-2 pressure decrease to 40 kgf/cm²</td>
</tr>
<tr>
<td>0.2</td>
<td>0.2</td>
<td>Closure of turbine no.2 stop valves</td>
<td>Due to closure of the stop valves of last operating turbine</td>
</tr>
<tr>
<td>0.70</td>
<td>0.74</td>
<td>Scram signal</td>
<td>Due to loss of normal power supply</td>
</tr>
<tr>
<td>0.70</td>
<td>0.74</td>
<td>Loss of normal power supply</td>
<td>Boundary condition</td>
</tr>
<tr>
<td>1.70</td>
<td>1.74</td>
<td>Start of control assemblies drop due to reactor scram</td>
<td>Delay of 1 s from scram signal is assumed</td>
</tr>
<tr>
<td>2.70</td>
<td>2.74</td>
<td>Actuation of ECCS safeguard and start of emergency power supply diesel-generators (DG)</td>
<td>Due to loss of normal power supply</td>
</tr>
<tr>
<td>5.70</td>
<td>5.74</td>
<td>Trip of all RCPs Switching-off of all groups of PRZ electric heaters</td>
<td>Due to loss of normal power supply</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Trip of main feedwater (MFW) pumps</td>
<td></td>
</tr>
<tr>
<td>5.70</td>
<td>5.74</td>
<td>Signal to close main steam isolation valves (MSIV) dividing MSH and MSIV at the main steam lines (MSL)</td>
<td>Due to signal &quot;Increase of pressure drop rate in each MSH semi-header to 0.7 kgf/cm²/s at cold legs temperature &gt;150 °C in 2 out of 6 loops holding for 1 s&quot;</td>
</tr>
<tr>
<td>10.70</td>
<td>10.74</td>
<td>Full closure of all MSIVs</td>
<td>Closure time is 5 s</td>
</tr>
<tr>
<td>17.70</td>
<td>17.74</td>
<td>Start of DG sequential loading program (SLP)</td>
<td>DG start-up time of 15 s is assumed</td>
</tr>
<tr>
<td>35.0</td>
<td>34.0</td>
<td>Minimum pressure at the reactor outlet (115.2/116.5 kgf/cm²)</td>
<td></td>
</tr>
<tr>
<td>35.0</td>
<td>34.0</td>
<td>Minimum PRZ level is 4.7 m</td>
<td></td>
</tr>
<tr>
<td>45.0</td>
<td>42.74</td>
<td>Start of boric acid supply to the reactor coolant system (RCS) by high pressure injection system (HPIS)</td>
<td>Due to ECCS signal taking into account a transport delay</td>
</tr>
<tr>
<td>180.0</td>
<td>170.0</td>
<td>Maximum pressure at reactor outlet is 136 kgf/cm²</td>
<td></td>
</tr>
<tr>
<td>201.0</td>
<td>200.0</td>
<td>End of RCP coast-down</td>
<td></td>
</tr>
<tr>
<td>400.0-2000.0</td>
<td>400.0-2000.0</td>
<td>Periodic actuation of SG safety relief valves (SRV)</td>
<td>Secondary pressure change within 48-56 kgf/cm²</td>
</tr>
<tr>
<td>2000.0</td>
<td>2000.0</td>
<td>End of calculation</td>
<td>Stabilization of parameters</td>
</tr>
</tbody>
</table>
After MSH break, abrupt pressure drop occurs in the main steam header (Figure 3-32 and Figure 3-33). Within 0.3 s due to decrease of pressure in MSH semi-headers the turbines’ stop valves are closed, both turbines are tripped and due to a closure of 2/4 stop valves of last operating turbine the reactor scram signal is generated at 0.74 s of the accident (Figure 3-40).

According to the scenario the loss of normal power supply is postulated simultaneously with the scram that leads to a trip of all RCPs, MFW pumps and PRZ electric heaters.

At 1.03 s (with a delay of 1.0 s) a “MSH break” signal is formed due to increase of pressure drop rate in MSH, and then a signal to open cutoff valves at HPIS charging lines, as well as a signal to close all MSIV (at MSH and MSLs) are actuated.

At 45.0/42.7 s of the calculation, after postulated delays for DG and HPIS start-up (after loss of normal power supply) and HPIS transport delay, boric acid injection to the primary circuit by HPIS pumps is initiated (Figure 3-45 – Figure 3-47). Due to HPIS injection RCS pressure and PRZ level recover and injection rate decreases.

Up to 200.0 s, RCP coast-down ends and natural circulation in all RCS loops is established (Figure 3-16 – Figure 3-21).

After closure of all MSIVs, the secondary coolant loss is terminated. Due to HPIS injection and RCS coolant heat-up the primary pressure increases up to 136.0 kgf/cm² at 180.0 s of the accident. Lack of heat removal from the secondary side results in an increase of secondary circuit pressure that leads to the opening of control SRVs at all SGs at ~400.0 s (Figure 3-26 – Figure 3-31).

The maximum secondary pressure reached in the calculation is 57.0 kgf/cm², and the maximal cladding temperature after scram is 309.0/301.0 °C.

The plots of the main parameters of calculation are presented below on Figure 3-1 – Figure 3-51.
Figure 3-1 MSH Break. Core Thermal Power

Figure 3-2 MSH Break. RCS Pressure
Figure 3-3 MSH Break. Pressurizer Level

Figure 3-4 MSH Break. Coolant Temperature in Hot Leg, Loop 1
Figure 3-5 MSH Break. Coolant Temperature in Hot Leg, Loop 2

Figure 3-6 MSH Break. Coolant Temperature in Hot Leg, Loop 3
Figure 3-7 MSH Break. Coolant Temperature in Hot Leg, Loop 4

Figure 3-8 MSH Break. Coolant Temperature in Hot Leg, Loop 5
Figure 3-9  MSH Break. Coolant Temperature in Hot Leg, Loop 6

Figure 3-10  MSH Break. Coolant Temperature in Cold Leg, Loop 1
Figure 3-11  MSH Break. Coolant Temperature in Cold Leg, Loop 2

Figure 3-12  MSH Break. Coolant Temperature in Cold Leg, Loop 3
Figure 3-13  MSH Break. Coolant Temperature in Cold Leg, Loop 4

Figure 3-14  MSH Break. Coolant Temperature in Cold Leg, Loop 5
Figure 3-15  MSH Break. Coolant Temperature in Cold Leg, Loop 6

Figure 3-16  MSH Break. RCS Loop 1 Mass Flow Rate
Figure 3-17  MSH Break. RCS Loop 2 Mass Flow Rate

Figure 3-18  MSH Break. RCS Loop 3 Mass Flow Rate
Figure 3-19  MSH Break. RCS Loop 4 Mass Flow Rate

Figure 3-20  MSH Break. RCS Loop 5 Mass Flow Rate
Figure 3-21  MSH Break. RCS Loop 6 Mass Flow Rate

Figure 3-22  MSH Break. Make-Up and Let-Down Mass Flow Rate
Figure 3-23  MSH Break. PRZ Heaters Power

Figure 3-24  MSH Break. PRZ Spray Mass Flow Rate
Figure 3-25  MSH Break. Core Reactivity

Figure 3-26  MSH Break. SG-1 Pressure
Figure 3-27  MSH Break. SG-2 Pressure

Figure 3-28  MSH Break. SG-3 Pressure
Figure 3-29  MSH Break. SG-4 Pressure

Figure 3-30  MSH Break. SG-5 Pressure
Figure 3-31  MSH Break. SG-6 Pressure

Figure 3-32  MSH Break. MSH-1 Pressure
Figure 3-33  MSH Break. MSH-2 Pressure

Figure 3-34  MSH Break. SG-1 Level (Wide Range)
Figure 3-35  MSH Break. SG-2 Level (Wide Range)

Figure 3-36  MSH Break. SG-3 Level (Wide Range)
Figure 3-37  MSH Break. SG-4 Level (Wide Range)

Figure 3-38  MSH Break. SG-5 Level (Wide Range)
Figure 3-39  MSH Break. SG-6 Level (Wide Range)

Figure 3-40  MSH Break. Turbines Mass Flow Rate
Figure 3-41  MSH Break. MFW Pump No.1 Mass Flow Rate

Figure 3-42  MSH Break. MFW Pump No.2 Mass Flow Rate
Figure 3-43  MSH Break. MFW Pump No.3 Mass Flow Rate

Figure 3-44  MSH Break. MFW Pump No.4 Mass Flow Rate
Figure 3-45  MSH Break. HPIS-1 Mass Flow Rate

Figure 3-46  MSH Break. HPIS-2 Mass Flow Rate
Figure 3-47  MSH Break. HPIS-3 Mass Flow Rate

Figure 3-48  MSH Break. Maximal Cladding Temperature
Figure 3-49  MSH Break. Break Mass Flow Rate

Figure 3-50  MSH Break. Break Mass Flow Rate (Fragment)
3.2 Loss of Turbine Condenser Vacuum

3.2.1 Brief Description of Initiating Event

This initiating event postulates loss of vacuum in the condenser of one of the turbines that leads to a trip of this turbine with closure of its stop valves. According to the expected frequency of occurrence the initiating events is categorized as a postulated transient and leads to a decrease of heat removal by the secondary circuit.

3.2.2 Initial Conditions

The initial conditions selected for transient calculation correspond to those specified in Table 3-1.

3.2.3 Boundary Conditions

Assumptions on the systems availability that are considered in calculation of the transient are specified below.

No operator actions are simulated in the scenario.

Control of reactor power, primary pressure, PRZ and SG levels, and MSH pressure is performed by automatic controllers. Operation of RPL and automatic switching of ARM to "T" mode (MSH pressure maintenance mode) due to increase of MSH pressure is not considered.

Operation of PRZ spray, as well as operation of the make-up and let-down system is also not taken into account.

Figure 3-51 MSH Break. Boric Acid Concentration in the Core
RCP coast-down is performed according to pump characteristics.

Conservatively, operation of Level 4, Level 3 and Level 2 emergency reactor protection is not accounted.

The 1.0 s delay is postulated for start of control assemblies drop to the reactor core after reactor scram initiation.

Steam pressure in MSH is maintained by operation of steam dump to atmosphere (BRU-A) no.1 controller. BRU-A-2 operation is not taken into account.

3.2.4 Calculation Results

Sequence of events for this transient is presented in Table 3-3.
<table>
<thead>
<tr>
<th>TRACE Time, s</th>
<th>RELAP5 Time, s</th>
<th>Event</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0</td>
<td>0.0</td>
<td>Closure of turbine no.1 stop valves</td>
<td>Due to loss of vacuum in a condenser of turbine no.1</td>
</tr>
<tr>
<td>1.1</td>
<td>1.0</td>
<td>Maximal cladding temperature is reached: 333.0 °C (TRACE) and 339 °C (RELAP5)</td>
<td></td>
</tr>
<tr>
<td>9.0</td>
<td>10.0</td>
<td>Switching off of PRZ electric heaters' group no.1</td>
<td>PRZ pressure increase</td>
</tr>
<tr>
<td>10.5 – 99.5</td>
<td>10.5 – 99.5</td>
<td>Periodic signals to withdraw control assemblies from ARM</td>
<td>ARM operation in &quot;N&quot; (neutron power maintenance) mode</td>
</tr>
<tr>
<td>30.0</td>
<td>30.0</td>
<td>Actuation of BRU-A automatic control mode</td>
<td>MSH semi-header pressure increase up to 52.0 kgf/cm²</td>
</tr>
<tr>
<td>44.0</td>
<td>43.0</td>
<td>Complete opening of BRU-A valve</td>
<td>Steam dump to atmosphere</td>
</tr>
<tr>
<td>100.0-1200.0</td>
<td>100.0-1200.0</td>
<td>ARM operation in &quot;N&quot; mode</td>
<td></td>
</tr>
<tr>
<td>1230.0</td>
<td>1240.0</td>
<td>MFW pumps trip</td>
<td>Due to decrease of secondary circuit deaerator down to 0.5 m</td>
</tr>
<tr>
<td>1220.0</td>
<td>1240.0</td>
<td>Start of EFW pump and start of water supply to SG</td>
<td>Cutoff valve at charging line is opened at level decrease in 2 out of 6 SGs for 0.4 m from the nominal value</td>
</tr>
<tr>
<td>1216.0</td>
<td>1232.0</td>
<td>Reactor scram signal</td>
<td>Level decrease in 2 SGs by 0.45 m from the nominal value and correspondent RCPs are in operation</td>
</tr>
<tr>
<td>1216.0</td>
<td>1232.0</td>
<td>RCPs trip</td>
<td>SG level decrease for 400 mm from the nominal value (holding for 20 s)</td>
</tr>
<tr>
<td>1217.0</td>
<td>1233.0</td>
<td>Start of control assemblies drop by scram signal</td>
<td></td>
</tr>
<tr>
<td>1416.0</td>
<td>1433.0</td>
<td>End of RCPs coast-down</td>
<td>Establishment of natural circulation in RCS loops</td>
</tr>
<tr>
<td>1500.0</td>
<td>1440.0</td>
<td>Maximal RCS pressure (136 kgf/cm²)</td>
<td></td>
</tr>
<tr>
<td>3600.0</td>
<td>3600.0</td>
<td>End of calculation</td>
<td>Stabilization of main parameters</td>
</tr>
</tbody>
</table>
In the initial period of the transient after closure of stop valves of turbine no.1 (Figure 3-91), the steam flow to the turbine that remains in operation is not sufficient to remove heat transferred from primary circuit to SGs, and the secondary pressure reaches the setpoints of BRU-A actuation at 30 s of the calculation time (Figure 3-92). Increase in secondary circuit pressure causes increase in the primary circuit parameters: pressure (Figure 3-53) and temperature (Figure 3-55 – Figure 3-66).

The calculation scenario does not take into account automatic ARM switching to "T" (secondary circuit pressure maintenance) mode at the increase MSH pressure, so ARM controller continues to maintain a specified power setpoint till the moment of scram actuation.

Uncompensated loss of the secondary circuit coolant through BRU-A results in a decrease of secondary circuit deaerators level that causes trip of MFW pumps (Figure 3-93 – Figure 3-96). Consequential level decrease for 0.45 m from the nominal in 2 SGs with correspondent RCPs in operation leads to actuation of reactor scram at 1216.0/1232.0 s. Due to a drop of all control assemblies into the reactor core the reactor power decreases to the decay value. At the same time excessive (compared to decay heat) steam flow to the operating turbine and steam loss via BRU-As causes decrease of secondary circuit pressure, that in turn results in a primary pressure decrease. BRU-A controllers remain in MSH pressure maintenance mode throughout the calculation (Figure 3-92).

Upon SG level decrease for 400 mm from the nominal value (holding for 20 s), all operating RCPs are tripped (Figure 3-67 – Figure 3-72).

Decrease of SG level (Figure 3-85 – Figure 3-90) causes actuation of EFW pumps (Figure 3-97) and their operation restores SG levels.

The maximal primary circuit pressure reached in calculation is 136.0 kgf/cm² (Figure 3-53).

The plots of the main parameters of calculation are presented below on Figure 3-52 – Figure 3-98.
Figure 3-52  Loss of Turbine Condenser Vacuum. Core Thermal Power

Figure 3-53  Loss of Turbine Condenser Vacuum. RCS Pressure
Figure 3-54  Loss of Turbine Condenser Vacuum. Pressurizer Level

Figure 3-55  Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 1
Figure 3-56  Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 2

Figure 3-57  Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 3
Figure 3-58  Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 4

Figure 3-59  Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 5
Figure 3-60  Loss of Turbine Condenser Vacuum. Coolant Temperature in Hot Leg, Loop 6

Figure 3-61  Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 1
Figure 3-62  Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 2

Figure 3-63  Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 3
Figure 3-64  Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 4

Figure 3-65  Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 5
Figure 3-66  Loss of Turbine Condenser Vacuum. Coolant Temperature in Cold Leg, Loop 6

Figure 3-67  Loss of Turbine Condenser Vacuum. RCS Loop 1 Mass Flow Rate
Figure 3-68  Loss of Turbine Condenser Vacuum. RCS Loop 2 Mass Flow Rate

Figure 3-69  Loss of Turbine Condenser Vacuum. RCS Loop 3 Mass Flow Rate
Figure 3-70  Loss of Turbine Condenser Vacuum. RCS Loop 4 Mass Flow Rate

Figure 3-71  Loss of Turbine Condenser Vacuum. RCS Loop 5 Mass Flow Rate
**Figure 3-72** Loss of Turbine Condenser Vacuum. RCS Loop 6 Mass Flow Rate

**Figure 3-73** Loss of Turbine Condenser Vacuum. Make-Up and Let-Down Mass Flow
Figure 3-74  Loss of Turbine Condenser Vacuum. PRZ Heaters Power

Figure 3-75  Loss of Turbine Condenser Vacuum. PRZ Spray Mass Flow Rate
Figure 3-76  Loss of Turbine Condenser Vacuum. Core Reactivity

Figure 3-77  Loss of Turbine Condenser Vacuum. SG-1 Pressure
Figure 3-78  Loss of Turbine Condenser Vacuum. SG-2 Pressure

Figure 3-79  Loss of Turbine Condenser Vacuum. SG-3 Pressure
Figure 3-80  Loss of Turbine Condenser Vacuum. SG-4 Pressure

Figure 3-81  Loss of Turbine Condenser Vacuum. SG-5 Pressure
Figure 3-82  Loss of Turbine Condenser Vacuum. SG-6 Pressure

Figure 3-83  Loss of Turbine Condenser Vacuum. MSH-1 Pressure
Figure 3-84  Loss of Turbine Condenser Vacuum. MSH-2 Pressure

Figure 3-85  Loss of Turbine Condenser Vacuum. SG-1 Level (Wide Range)
Figure 3-86  Loss of Turbine Condenser Vacuum. SG-2 Level (Wide Range)

Figure 3-87  Loss of Turbine Condenser Vacuum. SG-3 Level (Wide Range)
Figure 3-88  Loss of Turbine Condenser Vacuum. SG-4 Level (Wide Range)

Figure 3-89  Loss of Turbine Condenser Vacuum. SG-5 Level (Wide Range)
Figure 3-90  Loss of Turbine Condenser Vacuum. SG-6 Level (Wide Range)

Figure 3-91  Loss of Turbine Condenser Vacuum. Turbines Mass Flow Rate
Figure 3-92  Loss of Turbine Condenser Vacuum. BRU-A Steam Mass Flow Rate

Figure 3-93  Loss of Turbine Condenser Vacuum. MFW Pump No.1 Mass Flow Rate
Figure 3-94  Loss of Turbine Condenser Vacuum. MFW Pump No.2 Mass Flow Rate

Figure 3-95  Loss of Turbine Condenser Vacuum. MFW Pump No.3 Mass Flow Rate
Figure 3-96  Loss of Turbine Condenser Vacuum. MFW Pump No.4 Mass Flow Rate

Figure 3-97  Loss of Turbine Condenser Vacuum. EFW Pumps Mass Flow Rate
3.3 Trip of 4 Out of 6 RCPs

3.3.1 Brief Description of Initiating Event

This initiating event assumes simultaneous trip of 4 out of 6 RCPs. According to the expected frequency of occurrence the initiating events is categorized as transient and leads to a decrease of reactor coolant flow.

3.3.2 Initial Conditions

The initial conditions selected for transient calculation correspond to those specified in Table 3-1.

3.3.3 Boundary Conditions

Assumptions on the systems availability that are considered in calculation of the transient are specified below.

No operator actions are simulated in the scenario.

Control of reactor power, primary pressure, PRZ and SG levels, and MSH pressure is performed by automatic controllers taking into account their characteristics.

Conservatively the operation of RPL, Level 4, Level 3, Level 2 emergency protection is not considered. This allows ARM to operate in neutron power maintenance mode without a prohibition to withdraw the control group of control assemblies and prolong reactor operation at...
the increased power level (comparing to the one that corresponds to number of operating RCPs). Make-up system is assumed to fail at the moment of IE occurrence.

The 1.0 s delay is postulated for start of control assemblies drop to the reactor core after reactor scram initiation.

Steam pressure in MSH is maintained by operation of BRU-A. Operation of turbine bypass to condenser (BRU-K) is not taken into account.

RCP coast-down is performed according to pump characteristics.

### 3.3.4 Calculation Results

Sequence of events for this transient is presented in Table 3-4.

<table>
<thead>
<tr>
<th>TRACE Time, s</th>
<th>RELAP5 Time, s</th>
<th>Event Description</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0</td>
<td>0.0</td>
<td>Trip of 4/6 RCPs (RCP-2, 4, 5, 6)</td>
<td>IE occurrence</td>
</tr>
<tr>
<td>0.0</td>
<td>0.0</td>
<td>Failure of make-up system Failure of BRU-K</td>
<td>Boundary conditions</td>
</tr>
<tr>
<td>3.0</td>
<td>3.0</td>
<td>Reactor scram signal</td>
<td>Scram actuation due to a trip of 4 (or more) RCPs with 3 s delay</td>
</tr>
<tr>
<td>4.0</td>
<td>4.0</td>
<td>Start of control assemblies drop due to actuation of scram</td>
<td>1 s delay after scram signal initiation is postulated</td>
</tr>
<tr>
<td>7.5</td>
<td>7.5</td>
<td>Maximal primary circuit pressure of 124.1/124.8 kgf/cm² is reached</td>
<td></td>
</tr>
<tr>
<td>9.0</td>
<td>9.0</td>
<td>Closure of the stop valves of both turbines</td>
<td>Due to scram actuation (with time delay)</td>
</tr>
<tr>
<td>17.0</td>
<td>16.6</td>
<td>Closure of let-down valves</td>
<td>PRZ level decrease for 0.300 m from the nominal level</td>
</tr>
<tr>
<td>24.0</td>
<td>26.7</td>
<td>Opening of BRU-A1, BRU-A2</td>
<td>Automatic operation of BRU-A controllers in the pressure maintenance mode</td>
</tr>
<tr>
<td>45.0</td>
<td>25.2</td>
<td>Maximal steam lines pressure of 52.3 kgf/cm² (TRACE) and 52.7 kgf/cm² (RELAP) is reached</td>
<td></td>
</tr>
<tr>
<td>200.0</td>
<td>199.4</td>
<td>End of RCP-2,4,5,6 coast-down</td>
<td></td>
</tr>
<tr>
<td>1000.0</td>
<td>1000.0</td>
<td>End of calculation</td>
<td>Stabilization of parameters</td>
</tr>
</tbody>
</table>
After a trip of RCP-2, 4, 5, 6 the coolant flow rate in the loops with tripped RCPs rapidly decreases, while flow rate in the loops with operating RCPs increases due to a decrease of total pressure losses (Figure 3-114 – Figure 3-119). Reactor scram signal is generated with 3.0 s time delay by the "Trip of 4/6 RCPs" signal. In 1 s after the scram signal actuation (additional delay for signal transmission and disconnection of control assemblies' drives) the control assemblies start to drop into the reactor core. By this time, neutron reactor power is already decreased below the nominal due to the temperature reactivity feedback (Figure 3-99).

After scram actuation core power decreases down to decay heat, and hot legs temperature (Figure 3-102 – Figure 3-107) and RCS pressure (Figure 3-100) rapidly decrease. Coolant temperature decrease and coolant shrinkage cause rapid drop of PRZ level (Figure 3-101).

Closure of the stop valves of both turbines that occurs at 9 s of calculation (with a delay after reactor scram) causes sharp increase of the secondary circuit pressure (Figure 3-130, Figure 3-131). BRU-As open automatically and start to dump steam in order to maintain MSH pressure according to their operation logic (Figure 3-139). The maximum pressure reached in MSH semi-headers is 52.3/52.7 kgf/cm².

Decay heat power decrease and stable coolant circulation (from operating RCPs) causes decrease and subsequent stabilization of hot legs temperature (Figure 3-102 – Figure 3-107). Initial (within first 50 s of transient) decrease of SG level (Figure 3-132 – Figure 3-137) is compensated by operation of MFW controllers and to the 150-200 s of transient the SG levels are restored.

Reaching the balance between core decay heat and heat removed by dumping steam via BRU-A (Figure 3-139) the main parameters of the primary and secondary circuit are stabilized.

The plots of the main parameters of calculation are presented below on Figure 3-99 – Figure 3-144.
Figure 3-99  Trip of 4/6 RCPs. Core Thermal Power

Figure 3-100  Trip of 4/6 RCPs. RCS Pressure
Figure 3-101  Trip of 4/6 RCPs. Pressurizer Level

Figure 3-102  Trip of 4/6 RCPs. Coolant Temperature in Hot Leg, Loop 1
Figure 3-103 Trip of 4/6 RCPs. Coolant Temperature in Hot Leg, Loop 2

Figure 3-104 Trip of 4/6 RCPs. Coolant Temperature in Hot Leg, Loop 3
Figure 3-105  Trip of 4/6 RCPs. Coolant Temperature in Hot Leg, Loop 4

Figure 3-106  Trip of 4/6 RCPs. Coolant Temperature in Hot Leg, Loop 5
Figure 3-107 Trip of 4/6 RCPs. Coolant Temperature in Hot Leg, Loop 6

Figure 3-108 Trip of 4/6 RCPs. Coolant Temperature in Cold Leg, Loop 1
Figure 3-109  Trip of 4/6 RCPs. Coolant Temperature in Cold Leg, Loop 2

Figure 3-110  Trip of 4/6 RCPs. Coolant Temperature in Cold Leg, Loop 3
Figure 3-111 Trip of 4/6 RCPs. Coolant Temperature in Cold Leg, Loop 4

Figure 3-112 Trip of 4/6 RCPs. Coolant Temperature in Cold Leg, Loop 5
Figure 3-113 Trip of 4/6 RCPs. Coolant Temperature in Cold Leg, Loop 6

Figure 3-114 Trip of 4/6 RCPs. RCS Loop 1 Mass Flow Rate
Figure 3-115  Trip of 4/6 RCPs. RCS Loop 2 Mass Flow Rate

Figure 3-116  Trip of 4/6 RCPs. RCS Loop 3 Mass Flow Rate
Figure 3-117  Trip of 4/6 RCPs. RCS Loop 4 Mass Flow Rate

Figure 3-118  Trip of 4/6 RCPs. RCS Loop 5 Mass Flow Rate
Figure 3-119  Trip of 4/6 RCPs. RCS Loop 6 Mass Flow Rate

Figure 3-120  Trip of 4/6 RCPs. Make-Up and Let-Down Mass Flow
Figure 3-121  Trip of 4/6 RCPs. PRZ Heaters Power

Figure 3-122  Trip of 4/6 RCPs. PRZ Spray Mass Flow Rate
Figure 3-123 Trip of 4/6 RCPs. Core Reactivity

Figure 3-124 Trip of 4/6 RCPs. SG-1 Pressure
Figure 3-125 Trip of 4/6 RCPs. SG-2 Pressure

Figure 3-126 Trip of 4/6 RCPs. SG-3 Pressure
Figure 3-127 Trip of 4/6 RCPs. SG-4 Pressure

Figure 3-128 Trip of 4/6 RCPs. SG-5 Pressure
Figure 3-129  Trip of 4/6 RCPs. SG-6 Pressure

Figure 3-130  Trip of 4/6 RCPs. MSH-1 Pressure
Figure 3-131  Trip of 4/6 RCPs. MSH-2 Pressure

Figure 3-132  Trip of 4/6 RCPs. SG-1 Level (Wide Range)
Figure 3-133  Trip of 4/6 RCPs. SG-2 Level (Wide Range)

Figure 3-134  Trip of 4/6 RCPs. SG-3 Level (Wide Range)
Figure 3-135  Trip of 4/6 RCPs. SG-4 Level (Wide Range)

Figure 3-136  Trip of 4/6 RCPs. SG-5 Level (Wide Range)
Figure 3-137 Trip of 4/6 RCPs. SG-6 Level (Wide Range)

Figure 3-138 Trip of 4/6 RCPs. Turbines Mass Flow Rate
Figure 3-139  Trip of 4/6 RCPs. BRU-A Steam Mass Flow Rate

Figure 3-140  Trip of 4/6 RCPs. MFW Pump No.1 Mass Flow Rate
Figure 3-141 Trip of 4/6 RCPs. MFW Pump No.2 Mass Flow Rate

Figure 3-142 Trip of 4/6 RCPs. MFW Pump No.3 Mass Flow Rate
Figure 3-143 Trip of 4/6 RCPs. MFW Pump No.4 Mass Flow Rate

Figure 3-144 Trip of 4/6 RCPs. Maximal Cladding Temperature
3.4 Uncontrolled Withdrawal of Control Assemblies Group

3.4.1 Brief Description of Initiating Event

This initiating event assumes uncontrolled withdrawal of control group of control assemblies from the reactor core with a normal operating speed of 20 mm/s that can be caused by a malfunction of the reactor power control system. According to the expected frequency of occurrence the initiating events is categorized as a postulated transient that leads to an unintended increase of reactor power at the beginning of transient. The IE pertains to the IE group of anomalies in reactivity and power distribution in the reactor core.

3.4.2 Initial Conditions

The initial conditions selected for transient calculation correspond to those specified in Table 3-1.

3.4.3 Boundary Conditions

Assumptions on the systems availability that are considered in calculation of the transient are specified below.

No operator actions are simulated in the scenario.

ARM, RPL, Level 2, Level 3, and Level 4 emergency protection are assumed to be inoperable to allow maximal withdrawal of control group of control assemblies.

3.4.4 Calculation Results

Sequence of events for this transient is presented in Table 3-5.
Table 3-5  Sequence of Events for Uncontrolled Withdrawal of Control Assemblies

<table>
<thead>
<tr>
<th>TRACE Time, s</th>
<th>RELAP5 Time, s</th>
<th>Event Description</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0</td>
<td>0.0</td>
<td>Start of uncontrolled withdrawal of control group of control assemblies</td>
<td>Reactor scram due to high reactor power (&gt;110%) signal actuation</td>
</tr>
<tr>
<td>4.0</td>
<td>2.5</td>
<td>Reactor scram</td>
<td>Turbine trip due to scram with 5.0 s delay</td>
</tr>
<tr>
<td>9.0</td>
<td>7.5</td>
<td>Closure of stop valves of both turbines</td>
<td>Maintenance of RCS pressure according to design setpoints</td>
</tr>
<tr>
<td>10.0-240.0</td>
<td>10.0-240.0</td>
<td>Operation of PRZ heaters groups</td>
<td>Automatic operation of BRU-K controllers in the pressure maintenance mode</td>
</tr>
<tr>
<td>14.0</td>
<td>15.0</td>
<td>Maximal SG pressure is 51.9-52.0 kgf/cm²</td>
<td></td>
</tr>
<tr>
<td>15.0</td>
<td>15.0</td>
<td>Opening of BRU-A1 due to pressure increase in correspondent MSH semi-header</td>
<td>Automatic operation of BRU-A controllers in the pressure maintenance mode</td>
</tr>
<tr>
<td></td>
<td>15.0-27.0</td>
<td>Opening of BRU-A1, BRU-A2 due to pressure increase in correspondent MSH semi-header over 52.0 kgf/cm²</td>
<td></td>
</tr>
<tr>
<td>20.0-440.0</td>
<td>20.0-390.0</td>
<td>Full opening of make-up and closure of let-down control valves</td>
<td>PRZ level maintenance</td>
</tr>
<tr>
<td>50.0</td>
<td>60.0</td>
<td>Minimal PRZ level is 4.0 m (TRACE) and 4.4 m (RELAP5)</td>
<td></td>
</tr>
<tr>
<td>250.0</td>
<td>220.5</td>
<td>Start of PRZ spray valves operation</td>
<td>Maintenance of RCS pressure according to design setpoints</td>
</tr>
<tr>
<td>250.0</td>
<td>230.0</td>
<td>Maximal RCS pressure is 130.5 kgf/cm² (TRACE) and 128.0 kgf/cm² (RELAP5)</td>
<td></td>
</tr>
<tr>
<td>440.0</td>
<td>420.0</td>
<td>Start of primary pressure decrease</td>
<td>Restoration of nominal RCS pressure by PRZ spray operation</td>
</tr>
<tr>
<td>650.0</td>
<td>840.0</td>
<td>Operation of PRZ heaters groups</td>
<td>Maintenance of RCS pressure according to design setpoints</td>
</tr>
<tr>
<td>1800.0</td>
<td>1800.0</td>
<td>End of calculation</td>
<td></td>
</tr>
</tbody>
</table>

Uncontrolled withdrawal of the control group of control assemblies with a normal operating speed of 20 mm/s causes insertion of positive reactivity and thus, the reactor power increase (Figure 3-145).
At 4.0/2.5 s, the reactor neutron power reaches 110.0% of the nominal value (Figure 3-145) that causes scram signal actuation. Increase of coolant temperature (Figure 3-148 – Figure 3-159) at the beginning of transient due to initial reactor power increase is quickly terminated after reactor scram, and the temperature decreases rapidly.

Closure of stop valves of both turbines after the reactor scram causes sharp increase of the secondary circuit pressure (Figure 3-176, Figure 3-177). This causes actuation of BRU-K (Figure 3-185) and BRU-A in pressure maintenance mode that automatically decrease the secondary circuit pressure according to a design algorithm.

Decrease of primary coolant temperature (Figure 3-148 – Figure 3-159) within first 100 s of transient and correspondent coolant shrinkage result in a decrease of PRZ level (Figure 3-147), which is gradually restored at 450 s by operation of PRZ level controllers that adjust make-up and let-down flow (Figure 3-166).

After reaching the minimal value at ~60.0 s RCS pressure (Figure 3-146) starts to recover by PRZ heaters operation (Figure 3-167). At 250.0/220.5 s the primary pressure reaches PRZ spray valves opening setpoints (Figure 3-168), and after slow decrease is maintained close to a nominal value by PRZ heaters operation (Figure 3-167).

Within 1800 s of transient all primary and secondary circuit parameters are stabilized. The decay heat is removed in forced circulation mode by the secondary circuit via BRU-K and MFW pumps operation.

The plots of the main parameters of calculation are presented below on Figure 3-145 – Figure 3-191.
Figure 3-145  Uncontrolled Withdrawal of Control Assemblies Group. Core Thermal Power

Figure 3-146  Uncontrolled Withdrawal of Control Assemblies Group. RCS Pressure
Figure 3-147 Uncontrolled Withdrawal of Control Assemblies Group. Pressurizer Level

Figure 3-148 Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Hot Leg, Loop 1
Figure 3-149  Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Hot Leg, Loop 2

Figure 3-150  Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Hot Leg, Loop 3
Figure 3-151  Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Hot Leg, Loop 4

Figure 3-152  Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Hot Leg, Loop 5
Figure 3-153  Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Hot Leg, Loop 6

Figure 3-154  Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Cold Leg, Loop No.1
Figure 3-155  Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Cold Leg, Loop No. 2

Figure 3-156  Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Cold Leg, Loop No. 3
Figure 3-157  Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Cold Leg, Loop No.4

Figure 3-158  Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Cold Leg, Loop No.5
Figure 3-159 Uncontrolled Withdrawal of Control Assemblies Group. Coolant Temperature in Cold Leg, Loop No. 6

Figure 3-160 Uncontrolled Withdrawal of Control Assemblies Group. RCS Loop 1 Mass Flow Rate
Figure 3-161 Uncontrolled Withdrawal of Control Assemblies Group. RCS Loop 2 Mass Flow Rate

Figure 3-162 Uncontrolled Withdrawal of Control Assemblies Group. RCS Loop 3 Mass Flow Rate
Figure 3-163  Uncontrolled Withdrawal of Control Assemblies Group. RCS Loop 4 Mass Flow Rate

Figure 3-164  Uncontrolled Withdrawal of Control Assemblies Group. RCS Loop 5 Mass Flow Rate
Figure 3-165  Uncontrolled Withdrawal of Control Assemblies Group. RCS Loop 6 Mass Flow Rate

Figure 3-166  Uncontrolled Withdrawal of Control Assemblies Group. Make-Up and Let-Down Mass Flow
Figure 3-167  Uncontrolled Withdrawal of Control Assemblies Group. PRZ Heaters Power

Figure 3-168  Uncontrolled Withdrawal of Control Assemblies Group. PRZ Spray Mass Flow Rate
Figure 3-169  Uncontrolled Withdrawal of Control Assemblies Group. Core Reactivity

Figure 3-170  Uncontrolled Withdrawal of Control Assemblies Group. SG-1 Pressure
Figure 3-171  Uncontrolled Withdrawal of Control Assemblies Group. SG-2 Pressure

Figure 3-172  Uncontrolled Withdrawal of Control Assemblies Group. SG-3 Pressure
Figure 3-173  Uncontrolled Withdrawal of Control Assemblies Group. SG-4 Pressure

Figure 3-174  Uncontrolled Withdrawal of Control Assemblies Group. SG-5 Pressure
Figure 3-175  Uncontrolled Withdrawal of Control Assemblies Group. SG-6 Pressure

Figure 3-176  Uncontrolled Withdrawal of Control Assemblies Group. MSH-1 Pressure
Figure 3-177  Uncontrolled Withdrawal of Control Assemblies Group. MSH-2 Pressure

Figure 3-178  Uncontrolled Withdrawal of Control Assemblies Group. SG-1 Level (Wide Range)
Figure 3-179  Uncontrolled Withdrawal of Control Assemblies Group. SG-2 Level (Wide Range)

Figure 3-180  Uncontrolled Withdrawal of Control Assemblies Group. SG-3 Level (Wide Range)
Figure 3-181  Uncontrolled Withdrawal of Control Assemblies Group. SG-4 Level (Wide Range)

Figure 3-182  Uncontrolled Withdrawal of Control Assemblies Group. SG-5 Level (Wide Range)
Figure 3-183  Uncontrolled Withdrawal of Control Assemblies Group. SG-6 Level (Wide Range)

Figure 3-184  Uncontrolled Withdrawal of Control Assemblies Group. Turbines Mass Flow Rate
Figure 3-185 Uncontrolled Withdrawal of Control Assemblies Group. BRU-K Steam Mass Flow Rate

Figure 3-186 Uncontrolled Withdrawal of Control Assemblies Group. BRU-A Steam Mass Flow Rate
Figure 3-187  Uncontrolled Withdrawal of Control Assemblies Group. MFW Pump No.1
Mass Flow Rate

Figure 3-188  Uncontrolled Withdrawal of Control Assemblies Group. MFW Pump No.2
Mass Flow Rate
Figure 3-189  Uncontrolled Withdrawal of Control Assemblies Group. MFW Pump No.3
Mass Flow Rate

Figure 3-190  Uncontrolled Withdrawal of Control Assemblies Group. MFW Pump No.4
Mass Flow Rate
3.5 PRZ Surge Line Break

3.5.1 Brief Description of Initiating Event

This initiating event assumes break of surge line with equivalent diameter of 277 mm connecting hot leg of RCS loop no.1 with PRZ. According to the expected frequency of occurrence the initiating events is categorized as a design basis accident and pertains to loss of coolant accidents (LOCA) IE group.

3.5.2 Initial Conditions

The initial conditions selected for calculation of this accident correspond to those specified in Table 3-1.

3.5.3 Boundary Conditions

Assumptions on the systems availability that are considered in calculation of the accident are specified below.

Operator actions are not modeled.

Primary makeup is terminated at the moment of accident occurrence.

To prolong reactor operation at the nominal power, a failure of Level 2, Level 3 and Level 4 emergency protection systems is postulated. ARM operates in the reactor power maintenance mode until scram signal is generated.
Simultaneously with reactor scram actuation the loss of normal power supply, as well as failure of BRU-A1 is postulated to reach higher cladding temperature.

### 3.5.4 Calculation Results

Sequence of events for this accident are presented in Table 3-6.

#### Table 3-6 Sequence of Events for PRZ Surge Line Break Accident

<table>
<thead>
<tr>
<th>TRACE Time, s</th>
<th>RELAP5 Time, s</th>
<th>Event</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0</td>
<td>0.0</td>
<td>PRZ surge line break (LOCA with a diameter of 277 mm)</td>
<td>IE occurrence. Make-up failure (postulated)</td>
</tr>
<tr>
<td>2.0</td>
<td>3.0</td>
<td>Reactor scram actuation Loss of normal power supply RCS pressure decrease below 95.0 kgf/cm²</td>
<td></td>
</tr>
<tr>
<td>7.0</td>
<td>8.0</td>
<td>Closure of turbines stop valves Loss of vacuum in turbine condensers is assumed 5 s after loss of power supply</td>
<td></td>
</tr>
<tr>
<td>5.0</td>
<td>6.0</td>
<td>ECCS safeguard actuation, signal to start emergency DG Postulated loss of normal power supply</td>
<td></td>
</tr>
<tr>
<td>20.0</td>
<td>20.0</td>
<td>Start of ECCS HA injection to the reactor Decrease of RCS pressure below 60.0 kgf/cm²</td>
<td></td>
</tr>
<tr>
<td>20.0</td>
<td>21.0</td>
<td>Start of emergency DG sequential loading program, HPIS and LPIS pumps start 15 s delay for DG start-up after ECCS safeguard actuation</td>
<td></td>
</tr>
<tr>
<td>25.0</td>
<td>20.5</td>
<td>Maximal secondary circuit pressure 52.0/51.5 kgf/cm² is reached</td>
<td></td>
</tr>
<tr>
<td>40.0</td>
<td>41.0</td>
<td>Start of HPIS-1,2,3 injection</td>
<td>Delay after sequential loading program actuation (for pumps start-up and transport delay)</td>
</tr>
<tr>
<td>73.0</td>
<td>73.0</td>
<td>Start of AFW pump no.1 injection to SG 35 s after sequential loading program + delay for pump start-up and transport delay. Only one AFW pump (no.1) is powered from emergency DG</td>
<td></td>
</tr>
<tr>
<td>160-170</td>
<td>130</td>
<td>End of ECCS HA injection</td>
<td>HA depletion</td>
</tr>
<tr>
<td>250.0</td>
<td>425.0</td>
<td>Start of stable LPIS injection to RCS RCS pressure decrease below LPIS pump shut-off head</td>
<td></td>
</tr>
<tr>
<td>900.0</td>
<td>900.0</td>
<td>End of calculation Stable core cooldown, stabilization of the main reactor parameters</td>
<td></td>
</tr>
</tbody>
</table>
Break of PRZ surge line with a diameter of 277 mm results in large break LOCA that causes fast decrease of RCS pressure (Figure 3-193) and PRZ level (Figure 3-194). After IE occurrence, due to RCS pressure decrease down to 95 kgf/cm², a scram signal is generated at 2.0/3.0 s.

Postulated loss of normal power supply results in a trip of the equipment powered from correspondent busbars, namely, RCP, MFW pumps, PRZ heaters, BRU-K, and causes actuation of ECCS safeguard with a start of emergency DG. Turbine stop valves are closed by the "Loss of vacuum in turbine condensers" signal after loss of normal power supply. This causes temporary increase of SG pressure (Figure 3-217 – Figure 3-222) with a maximal value reached at 25/20 s, however BRU-A actuation setpoints are not reached due to intensive energy loss via the break. Nearly at the same time RCS pressure decreases below 60 kgf/cm² and HA injection to the reactor begins (Figure 3-243 – Figure 3-246).

15 s after start of DG (delay for DG start-up) the sequential loading program is actuated with connection of equipment to emergency power supply busbars according to a design algorithm, and at 40.0/41.0 s after IE occurrence (with postulated delay for HPIS pump start-up and transport delay) HPIS starts to inject boric acid to the primary circuit (Figure 3-237 – Figure 3-239).

At 250.0/425.0 s of the accident RCS pressure decreases below LPIS pump shut-off head and LPIS injection to the primary circuit is started (Figure 3-240 – Figure 3-242). Combined operation of HPIS and LPIS pumps compensates coolant loss via the break and to 600 s of calculation the majority of the primary and secondary circuit parameters are stabilized. Decay heat removal is provided by heat-up of the cold coolant injected by LPIS and HPIS, and energy loss via a break. Secondary circuit pressure decreases due to SG cooling down by the primary circuit.

During the accident, a stable decrease of the maximal cladding temperature from the initial value at normal power operation to ~125 °C is observed (Figure 3-246).

The plots of the main parameters of calculation are presented below on Figure 3-192 – Figure 3-249.
Figure 3-192  PRZ Surge Line Break. Core Thermal Power

Figure 3-193  PRZ Surge Line Break. RCS Pressure
Figure 3-194  PRZ Surge Line Break. Pressurizer Level

Figure 3-195  PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 1
Figure 3-196  PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 2

Figure 3-197  PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 3
Figure 3-198  PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 4

Figure 3-199  PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 5
Figure 3-200  PRZ Surge Line Break. Coolant Temperature in Hot Leg, Loop 6

Figure 3-201  PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop No. 1
Figure 3-202  PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop No. 2

Figure 3-203  PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop No. 3
Figure 3-204 PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop No. 4

Figure 3-205 PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop No. 5
Figure 3-206  PRZ Surge Line Break. Coolant Temperature in Cold Leg, Loop No. 6

Figure 3-207  PRZ Surge Line Break. RCS Loop 1 Mass Flow Rate
Figure 3-208  PRZ Surge Line Break. RCS Loop 2 Mass Flow Rate

Figure 3-209  PRZ Surge Line Break. RCS Loop 3 Mass Flow Rate
Figure 3-210  PRZ Surge Line Break. RCS Loop 4 Mass Flow Rate

Figure 3-211  PRZ Surge Line Break. RCS Loop 5 Mass Flow Rate
Figure 3-212  PRZ Surge Line Break. RCS Loop 6 Mass Flow Rate

Figure 3-213  PRZ Surge Line Break. Make-Up and Let-Down Mass Flow
Figure 3-214  PRZ Surge Line Break. PRZ Spray Mass Flow Rate

Figure 3-215  PRZ Surge Line Break. Core Reactivity
Figure 3-216  PRZ Surge Line Break. SG-1 Pressure

Figure 3-217  PRZ Surge Line Break. SG-2 Pressure
Figure 3-218  PRZ Surge Line Break. SG-3 Pressure

Figure 3-219  PRZ Surge Line Break. SG-4 Pressure
Figure 3-220  PRZ Surge Line Break. SG-5 Pressure

Figure 3-221  PRZ Surge Line Break. SG-6 Pressure
Figure 3-222  PRZ Surge Line Break. MSH-1 Pressure

Figure 3-223  PRZ Surge Line Break. MSH 2 Pressure
Figure 3-224  PRZ Surge Line Break. SG-1 Level (Wide Range)

Figure 3-225  PRZ Surge Line Break. SG-2 Level (Wide Range)
Figure 3-226  PRZ Surge Line Break. SG-3 Level (Wide Range)

Figure 3-227  PRZ Surge Line Break. SG-4 Level (Wide Range)
Figure 3-228  PRZ Surge Line Break. SG-5 Level (Wide Range)

Figure 3-229  PRZ Surge Line Break. SG-6 Level (Wide Range)
Figure 3-230  PRZ Surge Line Break. Turbines Mass Flow Rate

Figure 3-231  PRZ Surge Line Break. MFW Pump No.1 Mass Flow Rate
Figure 3-232  PRZ Surge Line Break. MFW Pump No.2 Mass Flow Rate

Figure 3-233  PRZ Surge Line Break. MFW Pump No.3 Mass Flow Rate
Figure 3-234  PRZ Surge Line Break. MFW Pump No.4 Mass Flow Rate

Figure 3-235  PRZ Surge Line Break. AFW Mass Flow
Figure 3-236  PRZ Surge Line Break. HPIS-1 Mass Flow Rate

Figure 3-237  PRZ Surge Line Break. HPIS-2 Mass Flow Rate
Figure 3-238  PRZ Surge Line Break. HPIS-3 Mass Flow Rate

Figure 3-239  PRZ Surge Line Break. LPIS-1 Mass Flow Rate
Figure 3-240  PRZ Surge Line Break. LPIS-2 Mass Flow Rate

Figure 3-241  PRZ Surge Line Break. LPIS-3 Mass Flow Rate
Figure 3-242  PRZ Surge Line Break. HA-1 Mass Flow Rate

Figure 3-243  PRZ Surge Line Break. HA-2 Mass Flow Rate
Figure 3-244  PRZ Surge Line Break. HA-3 Mass Flow Rate

Figure 3-245  PRZ Surge Line Break. HA-4 Mass Flow Rate
Figure 3-246  PRZ Surge Line Break. Maximal Cladding Temperature

Figure 3-247  PRZ Surge Line Break. Break Mass Flow Rate
Figure 3-248  PRZ Surge Line Break. Break Mass Flow Rate (Fragment)

Figure 3-249  PRZ Surge Line Break. Boric Acid Concentration in the Core
4 CONCLUSIONS

After validation of VVER-440/V-213 thermal-hydraulic model for TRACE code by simulating several operational events that had occurred at Ukrainian NPPs, the validation effort was extended by calculations of 5 DBA scenarios and comparing the results obtained with this model and correspondent RELAP5/Mod3.2 model. Both TRACE and RELAP5 models applied for calculations are nearly identical with respect to model scope, nodalization, components geometry and equipment characteristics.

Comparison of the results obtained with TRACE and RELAP5 models indicates some differences in calculated parameters. In particular, the differences in the primary circuit pressure that were observed in some of the scenarios are caused by different mathematical models and correlations for steam condensation, which are used in the special PRESSURIZER model in TRACE, as compared to the pressurizer modelling in RELAP5. Cladding temperature differences are related to the specifics of the heat structure modeling approach and the absence of TRACE correlation options, which does not allow more precise model adjustment to ensure complete convergence with the relevant RELAP5 models.

Nevertheless, these differences do not affect significantly the overall behavior of the main parameters of the primary and secondary circuit and the sequence of events in the analyzed scenarios calculated with TRACE and RELAP5 models, and quantitatively the values are in a good agreement.

The results of cross-code validation calculations demonstrate that developed VVER-440/V213 thermal-hydraulic model for TRACE code is able to reproduce adequately the NPP response to transients and accidents without core melt that were calculated previously in safety analysis reports using the RELAP5 model. For the majority of plant parameters good correspondence between TRACE and RELAP5 results is obtained.

Based on the results of model validation it can be concluded that developed WWER-440/V-213 thermal hydraulic model for TRACE computer code can be used for calculations of transients and accidents in support of regulatory review of safety analyses documentation.
5 REFERENCES


This report is developed by the State Nuclear Regulatory Inspectorate of Ukraine (SNRIU) and its technical support organization, the State Scientific and Technical Center for Nuclear and Radiation Safety of Ukraine (SSTC NRS), under Implementing Agreement On Thermal-Hydraulic Code Applications And Maintenance Between The United States Nuclear Regulatory Commission and State Nuclear Regulatory Inspectorate of Ukraine (signed in 2014) in accordance with Article III, Section C, of the Agreement.

The report provides results of the comparison calculations conducted with application of SSTC NRS model of VVER-440/V-213 for TRACE and RELAP5 computer codes. The calculation scenarios analyzed include design basis accidents and transients from several initiating event groups usually evaluated in safety analysis reports.

Accident Analysis, Pressurized Water Reactor, WWER-440/V-213, RELAP5, TRACE, Design Basis Accident, Comparative Calculations, Validation