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VALIDATION OF A VVER 1000 TRACE MODEL

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Abstract

The presented article discusses developing and validation of a VVER 1000 TRACE V5.0p5 model. Developing such model for a TRACE computer code is important as it increases the capability of the analyses in simulating nuclear power plant behaviour for different accidents including capabilities for simulating 3D thermal hydraulic phenomena in the reactor core simultaneously with 3D reactor physics calculations. In this way, it is possible to conduct transients involving spatial thermal-hydraulic processes in the reactor core.

In developing a VVER 1000 model for TRACE code an existing RELAP5 1D model has been used developed in the INRNE-BAS and widely validated and used for safety assessment of a VVER 1000 reactor system.

The validation process includes comparisons of steady-state and transient calculations. The behaviour of selected important parameters calculated from the existing RELAP5 Mod3.3 VVER 1000 model has been compared with results received by using a TRACE V5.0p5 VVER 1000 new computer model.

The integral response of the reactor system during a total station blackout (SBO) event has been investigated. The transient includes shutdown of the reactor system, isolating of the turbine, a transition from forced to natural circulation of the coolant in the primary circuit, switching off the feed water system from all steam generators (SGs), activation of safety valves in the secondary and primary circuits, dryout of SGs, loss of natural circulation, reactor core heat up, etc.

The performed comparison shows a very good agreement between results received by TRACE model compared to the results received by RELAP5 model.

Key words: validation, VVER 1000, RELAP5, TRACE, SBO event

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I. Introduction. Most of the design and the safety analyses of VVER 1000 reactors in Kozloduy NPP are carried out using the best estimate code RELAP5. Some of these analyses are used for supporting the Emergency Operating Procedures (EOP) [1,2] and the Probabilistic Safety Analyses (PSA) as well as for analyses of specific issues important for the safety operation of the existing units [3]. The increasing of safety requirements requests involving different computer codes for safety assessment allowing extending the capabilities of assessment. For example, the integral computer code TRACE allows simulation of 3D thermal hydraulic phenomena in the reactor vessel as well as 3D neutronics calculation for investigation of specific spatial effects such as switching one main coolant pump (MCP) to 3 working, steam line break, SG tubing break, etc. Before developing and executing fully 3D reactor core capabilities there is a need to investigate the capability of new TRACE model with 1D reactor core to simulate integral phenomena and processes. In this way, the TRACE VVER 1000 model will be compared against RELAP5 VVER 1000 to predict properly integral transients using both a 1D reactor core. This is important in simulating nuclear power plant response in case of many accidents, where the spatial reactor core effects are not observed. On the other hand, it will allow to compare different parameters of nuclear power plant for integral response to the initiating events keeping in mind that both reactor cores are 1D. For this purpose, a specific scenario has been selected that includes important processes and phenomena that are often encountered in conducting safety analyses of nuclear accidents at nuclear power plants.

This paper discusses developing and validating by comparing code to code behaviour of some important parameters during accident progression [4,5]. For this purpose SBO initiating event is selected, which involves loss of all MCPs, transition from forced to natural circulation, dryout of all SGs and loss of natural circulation as well as heat up of reactor core due to loss of cooling [6–8]. During the accident progression Steam dump to the atmosphere (BRU-As) is activated and after losing the effectiveness of a SGs in heat removing from primary circuit by secondary side of SGs, the activation of Pressurizer Safety Valve (SV) will occur. After opening the Pressurizer SV, it will continue to cycle between its set points of opening and closing.

Some of the key phenomena which are difficult to model but are significantly important for the assessment of the natural circulation system performance are as follows: low flow natural circulation mainly because the flow is not fully developed and can be multi-dimensional in nature; flow instabilities; critical heat flux under oscillatory condition; flow stratification particularly in large diameter vessel; thermal stratification in large pools; effect of non-condensable gases on condensation, etc.

A TRACE model for a VVER 1000, V320 reactor type has been developed. It is part of the VVER-1000 CAMIVVER project in the frame of Horizon 2020 research and innovation programme [9].

II. Main steps in developing of VVER 1000 TRACE input model. The VVER 1000 TRACE V5.0p5 input model has been built using the SNAP software, with converting of existing VVER 1000 RELAP5/MOD3.3 model to the TRACE model. The RELAP5/MOD3.3 VVER 1000 model has been developed and validated by the Institute for Nuclear Research and Nuclear Energy [10,11].

For developing of a base TRACE input model a RELAP5 model has been used. The RELAP5 model was opened in SNAP platform and converted automatically to the TRACE format using software SNAP (Symbolic Nuclear Analysis Package) [12] provided to the INRNE-BAS by NRC Agreement and a following refinement that is made by hand. In fact, the automatic conversion can lead to a variety of errors due to the differences in the TRACE and RELAP5 specifications and capabilities, and to the particular modelling choices made in the model to be processed. For avoiding some problems, most of the logic used in RELAP5 model was reduced and simplified to the requested for performing selected initiating transient.

The discretization (or nodalization) of the system is defined by the code user and is provided in an input file whose format depends on the code specifications. Several components are defined in order to simplify the hydrodynamic modelling.

The hydrodynamic systems of RELAP5 1D and TRACE 1D models are discretized in a network of fluid control volumes connected by junctions and are consistent with a real flow path in VVER 1000 nuclear power plant. In the codes field equations are solved for each of the control volumes according to the specified flow path. The TRACE system nodalization is defined by the RELAP5 nodalization. The specific components "FILL" in TRACE model are used to provide time-dependent temperature, pressure and velocity of the flow at the inlet of the system. In RELAP5, the boundary conditions are given with the time-dependent volume and time-dependent junction components, while in TRACE the "FILL" component for feeding and "BREAK" components for outflows are used. It should be mentioned that all feed components have been rebuilt from TDV and TDJ to the FEED components for better reflecting the boundary parameters.

Some components are useful to reduce the amount of information needed in the input file. For instance, the PIPE component that is available in both RELAP5 and TRACE consists of a series of control volumes and junctions. By using the PIPE component, the user only has to insert the data for one control volume and the code will automatically set that value to all the control volumes making up the PIPE component. Other components allow including data for features and additional processing. One example is the VALVE component where type characteristics and data for determining the flow area of the valve can be entered. The trips used in RELAP5 have been reviewed one by one and reorganized to correspond to their purposes.

For comparison of plant response on initiation of Station blackout accident in NPP with a VVER 1000 reactor type, a VVER 1000 model developed for RE-

LAP5/MOD3.3 computer code [13] and VVER 1000 model developed for TRACE V5.0p5 computer code [14] have been used.

In both input models all important equipment is included for primary and secondary side and a detailed nodalization of primary, secondary side and safety systems were kept the same for easier comparison [15].

III. Description of the scenario for SBO event. The initial power is 100%. The status of reactor core is at the end of life. All active safety systems are not used due to the specifics of the IE.

1. Switching off of all MCPs at 0 s due to the IE "Total station blackout" – SBO.
2. Reactor SCRAM with delay of 1.6 s due to loss of 3 from 4 MCP at power $N > 75\%$.
3. Switching off of Make up – Let down systems – 2 s.
4. Switching of main FW system with delay 5 s.
5. Switching off of all Pressurizer heaters (they are not available to the end of transient).
6. Isolating Turbine generator with delay 10 s.
7. Increasing of secondary pressure and opening of BRU-As.
8. Dryout of SGs.
9. Loss of natural circulation (NC) in primary circuit.
10. Opening of Pressurizer SV after reaching its safety set point due to increasing of primary pressure. Beginning of cycling of SV between 176 kg/cm^2 and 182 kg/cm^2.
11. Loss of subcooling ($\Delta t_I < 10\,^{\circ}$C).
12. End of calculation at around 15 000 s (after beginning core exit heat up).

IV. Discussion of the results. Before running the calculations, both models have been stabilized at their nominal parameters at 100% reactor power. All initial parameters in both models as primary and secondary pressure, core inlet and exit temperatures, SGs and Pressurizer water levels, characteristic of SGs feed waters as well Make up/Let down systems have very similar (almost the same) initial values. As it can be seen, they are very close or almost identical to the plant design data. Some of the calculated important parameters of primary and secondary circuits for initial conditions are presented below:

Reactor thermal power, MW	a) 3000	b) 3000	c) 3000
Primary pressure, MPa	a) 15.7	b) 15.7	c) 15.7
Average coolant temperature at reactor outlet, K	a) 593.1	b) 593.6	c) 593.7
Average coolant temperature at reactor inlet, K	a) 563.0	b) 562.0	c) 562.0
Mass flow rate through one loop, kg/s	a) 4400	b) 4470	c) 4390
Pressure in SG, MPa	a) 6.27	b) 6.26	c) 6.38
Pressure in main steam header (MSH), MPa	a) 6.08	b) 6.10	c) 6.15
Steam mass flow rate through SG steam line, kg/s	a) 408.0	b) 409.0	c) 409.6

*Note: a) Plant Design b) RELAP5 c) TRACE

Once the initiating event (IE) is activated, loss of all four MCPs and a transition from forced to natural circulation is observed. Due to loss of 3 MCPs from 4 and reactor power being more than 75% there is a formation of signal for activation of reactor SCRAM and in 1.6 s after the IE all control rods drop to the lower reactor core point in 2 to 4 s. After the IE occurred disconnecting of a Make up/Let down system is assumed and all heaters are switched off. Additionally, loss of main and all auxiliary and emergency feed water systems to the SGs is observed in 5 s. The turbine is isolated in 10 s and it causes increasing of the secondary pressure. After increasing of pressure in the secondary side to the set point of activation of BRU-As, they are opened and start to support pressure at around 64 atm. After losing the capability of SGs for removing reactor residual heat from the reactor core a beginning of increase of primary side pressure to the set point of opening of Pressurizer safety valve is observed at 182 atm. The valve starts to cycle between 182 and 176 atm until the end of calculation. After losing of the natural circulation in primary circuit and loss of significant part of primary coolant through the Pressurizer SV the beginning of primary heat up at approximately 12 700 s is observed, as can be seen in Fig. 2.

Four important parameters calculated by RELAP5 and TRACE codes are presented in Fig. 1–4 which are compared in analyzing nuclear power plant response to the initiating event. The behaviour of the selected nuclear power plant parameters for primary and secondary sides describe the integral response of a VVER 1000 reactor on the selected initiating event.

The behaviour of primary pressure in both calculations is compared in Fig. 1. The trends of calculated curve are very similar. The observed deviations in primary pressure for the first 6000 s could be explained by the work of Pressurizers. The trends of calculated curves of primary pressure for the first 6000 s are similar. The most important fact when comparing the primary side pressures is almost the same time of their increasing or decreasing in this period. After reducing capabilities of SGs to remove reactor core heat at approximately 4500 s, in both calculations simultaneous increasing of pressure to the set point of Pressurizer safety valves operation is observed. After opening the Pressurizer safety valves, they begin to cycle between its set points of opening and closing in both cal-

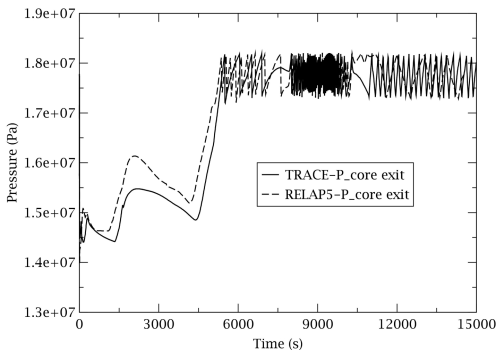

Fig. 1. Primary circuit pressure

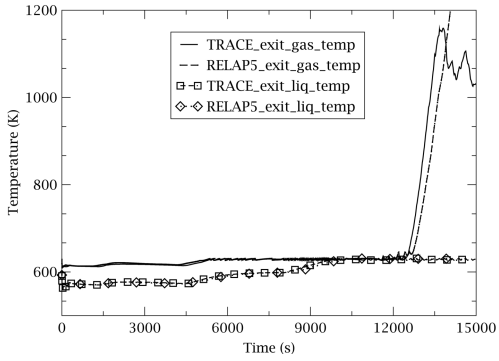

Fig. 2. Core exit temperature

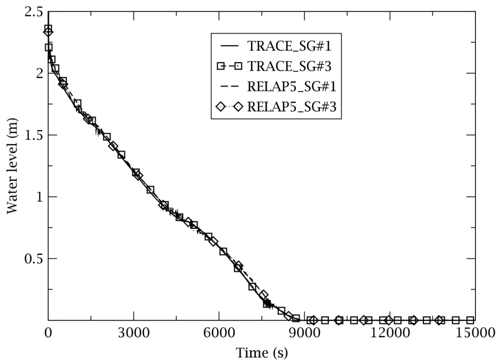

Fig. 3. Comparison of dryout of SGs

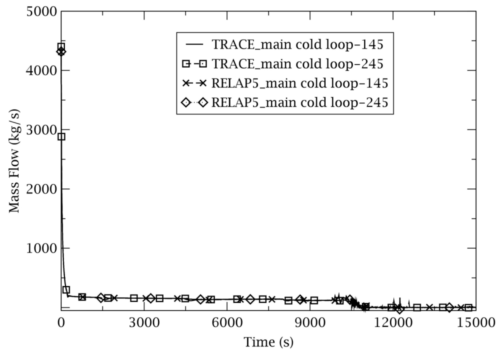

Fig. 4. Comparison of main coolant loop flows

culations to the end of scenario. The comparison of core exit temperatures for both calculations in Fig. 2 demonstrates the same behaviour. As can be seen the beginning of heat up starts at the same time 12 700 s in both calculations. The increasing the temperature above 1200 K is not investigated and deviations are not analyzed because the code is used for design based accidents, where the temperature is below 1200 K. The behaviour of SGs water level is compared in Fig. 3. A very close water levels decrease is observed in both calculations. After reducing water level at around 1 m there is a loss of effectiveness of SGs heat removing in both models which causes beginning of increase of primary pressure reaching the set points of opening of Pressurizer safety valves.

Primary main coolant loop flow rates are compared in Fig. 4. As can be seen there is almost identical transition from forced to natural circulation at approximately 240 s from the beginning and loss of natural circulation almost at the same time at approximately 10 000 s in both calculations.

V. Outlines and conclusions. The TRACE model has been developed and validated against RELAP5 model by code to code comparison in an analysis of SBO accident. The transition from forced to natural circulation was simulated and compared. The comparison of SGs dryout was performed. The works of BRU-As in secondary side have been compared. The activation and work of Pressurizer Safety Valves in primary circuit have been investigated and compared.

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