5.11 Peculiarity by Modeling of the Control Rod Movement by the OECD Kalinin-3 Benchmark

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Abstract

The paper presents an important part of the results of the OECD/NEA benchmark transient 'Switching off one main circulation pump at nominal power' analyzed as a boundary condition problem by the coupled system code ATHLET-BIPR-VVER. Some observations and comparisons with measured data for integral reactor parameters are discussed. Special attention is paid on the modeling and comparisons performed for the control rod movement and the reactor power history.

Introduction

The aim of this work is to compare integral reactor parameters with the available transient sets with measured data. These comparisons are of very big importance by the validation of the coupled system code ATHLET-BIPR-VVER as they show the correctness of the adopted methodology by simulation of a pseudo 3D core thermal-hydraulics in a parallel thermal-hydraulic channel (PTHC) approximation and prove the influence and the importance of the interaction of local neutron-physical and thermal-hydraulic feedback parameters, which is in fact the main advantage of the 'best estimate' 3D coupled code prediction capabilities compared to the point kinetics models.

Under the guidance of the NEA/OECD a benchmark on the basis of measurements at Kalinin-3 NPP is being specified [1]. Three exercises are defined in it. The performed work is near to the specification of Exercise #2 but is more directed to the future OECD/NEA UAM Benchmark analysis [2]. Some differences in the comparisons of the global parameters are tried to be explained on the basis of performed calculations and previous experiences reported in [3-7].

Description of the Benchmark scenario

After switching off of MCP #1 the reactor power is automatically decreases in 71 s to 67.8% P_{nom} according to the measured data histories, by insertion at the beginning of control rod group (CRG) #10 and later on when it reaches 50% insertion depth - CRG #9. At time t=

29 s a reverse flow in the affected loop #1 takes place. That leads to a rapid redistribution of the coolant flow through the reactor pressure vessel resulting into a spatially dependent coolant temperature change. This leads to an asymmetric power distribution that is being recorded by the local in-core temperature and neutron flux measuring devices. Thereby, the simulation of the transient requires evaluation of the core response from a multi-dimensional perspective (coupled 3D neutronics/core thermal-hydraulics). After 300 s the reactor parameters stabilize.

The applied reactor pressure vessel (RPV) nodalization scheme for the coupled code system ATHLET-BIPR-VVER is discussed in detail in [8]. The code system has very realistic models of all important for this transient controllers.

Comparisons of some global parameters

Figure 1 shows the predicted integral power evolution during the transient compared to the measured one, once on the basis of the ex-core fission ionization chambers (FIC) and another one on the basis of the in-core self powered neutron detectors (SPND). At the beginning the power is reduced by the insertion of the CRG #10 and CRG #9 (Fig. 2) under the algorithm for fulfilling the safety requirements of the NPP operation with one switched off main circulation pump (MCP). After reaching the minimum power level at about 70 s the power starts to increase within about 2-3% after which it stabilizes at a level of 68.7%. This effect of power increase is due to the colder coolant flow that enters the core and in that way introduces a positive density reactivity effect. The analysis of the curves in Fig. 1 and Fig. 2 shows that:

- After the 70-th sec of the transient there is a difference of the measured power evolution by the two systems - ex-core and in-core. The SPND based power (restored from neutron flux sensors located in 64 assemblies in 7 layers) ‘catches’ the power increase but based on FIC power does not ‘notice’ it.
- The predicted power by ATHLET-BIPR-VVER matches the power behavior very good depicting correctly the return to power after the 70-th sec measured with the in-core system.
- The comparisons of the movement histories of the CRG #10 and CRG #9 show some differences based on the fact that the control rod group insertion/withdrawal logic of the VVER-1000 is based on the FIC measurements, and by the ATHLET-BIPR-VVER

simulations - on the predicted total core power generation. Due to this reason it is not observed by the simulated results the small withdrawal of the CRG #10 and CRG #9 as by

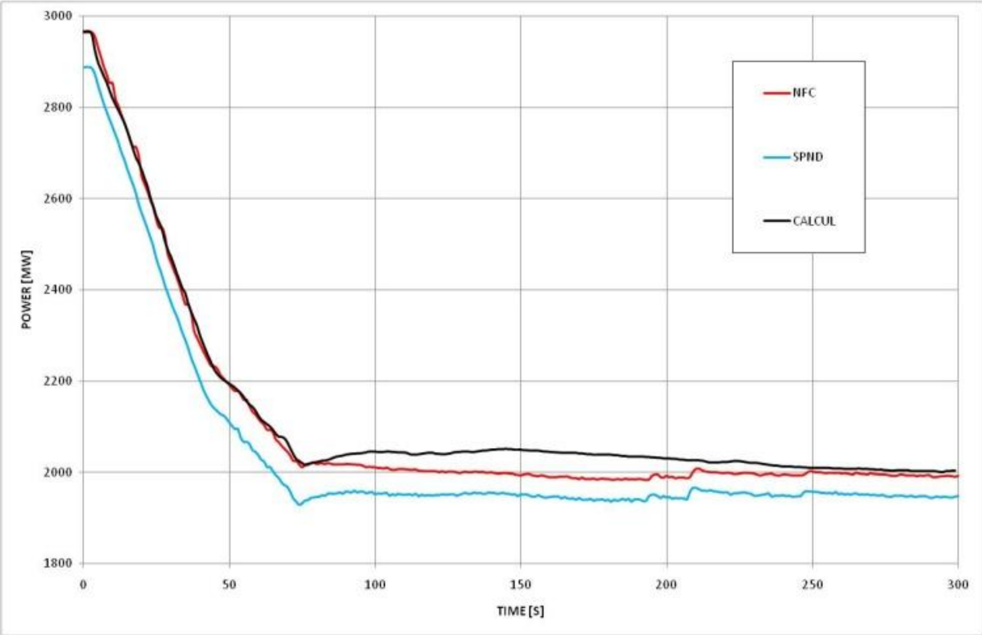

Fig. 1 Comparison of calculated total reactor power evolution with measured by the FIC and SPND systems

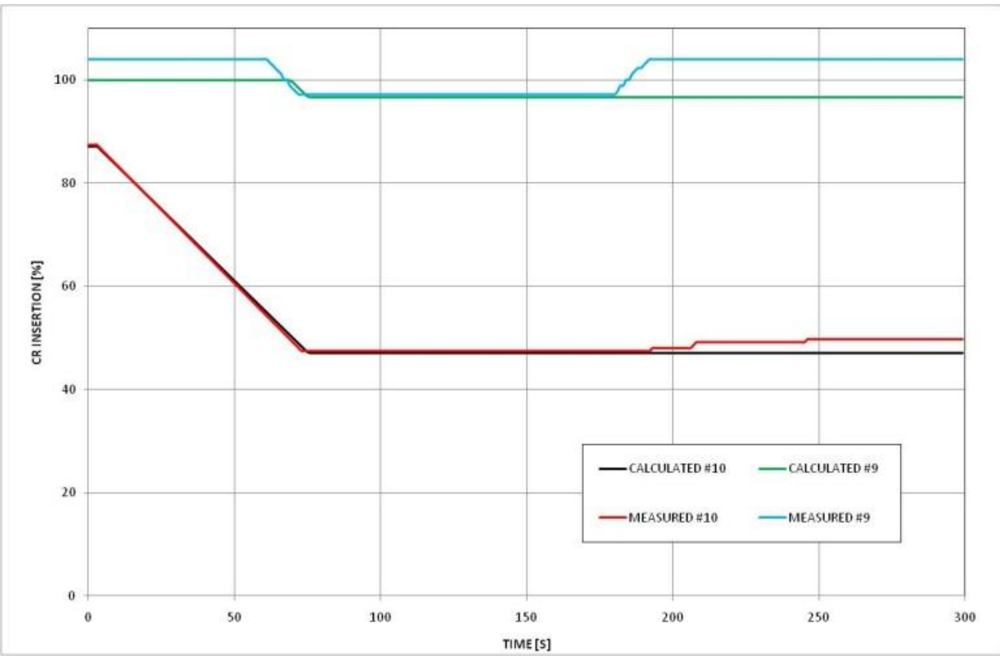

Fig. 2 Comparison of predicted CRG #9 and CRG #10 insertion histories with the measured

the measurements after the 180-th sec. The small difference for CRG #9 starting location (104 % by the measurements) at the beginning of the transient is connected with the fact that the upper control rods' end-switches are located at 14 cm above the active core (as described in the specification) which is modeled in ATHLET as a delay of 7.29 s for the CRG #9 beginning of insertion.

- A very good overlapping of the measured and predicted values are observed for the insertion in the active core time points of the CRG #10 and CRG #9, for their gradient of movement and also for the reached absolute insertion depth during the main phase of the transient. This fact is a proof of the very good simulation capability of modeling the NPP controllers and RPV thermal-hydraulics by ATHLET system code and also the 3D power prediction of BIPR-VVER code.
- It must be taken into account that the analysis are performed with boundary conditions taken from the temperature measurements at reactor inlet (outlet), pressure at reactor outlet and the volume of the coolant in the primary loop which are described in detail in [8].

As another example of comparison of global parameters is presented in Fig. 3 the hot legs' coolant temperatures which independently from the applied boundary condition problem are simulated values with the exception of that of the hot leg #1 for the first 29s (till the moment of flow reverse in this loop). The time histories of the predicted coolant temperatures fit the measurements within accuracy of ±0.5°C.

The same accuracy is detected by the comparisons of the cold leg #1 coolant temperature (Fig. 4) after the first 29s (after the moment of flow reverse in this loop) of the transient. By all comparisons is being applied the inertia model [5] for the coolant temperatures at different locations to enable the correctly comparison of the predicted data with the measurements. Other global parameters (power, mass flows …) are predicted with the accuracy of ± (1-2) %.

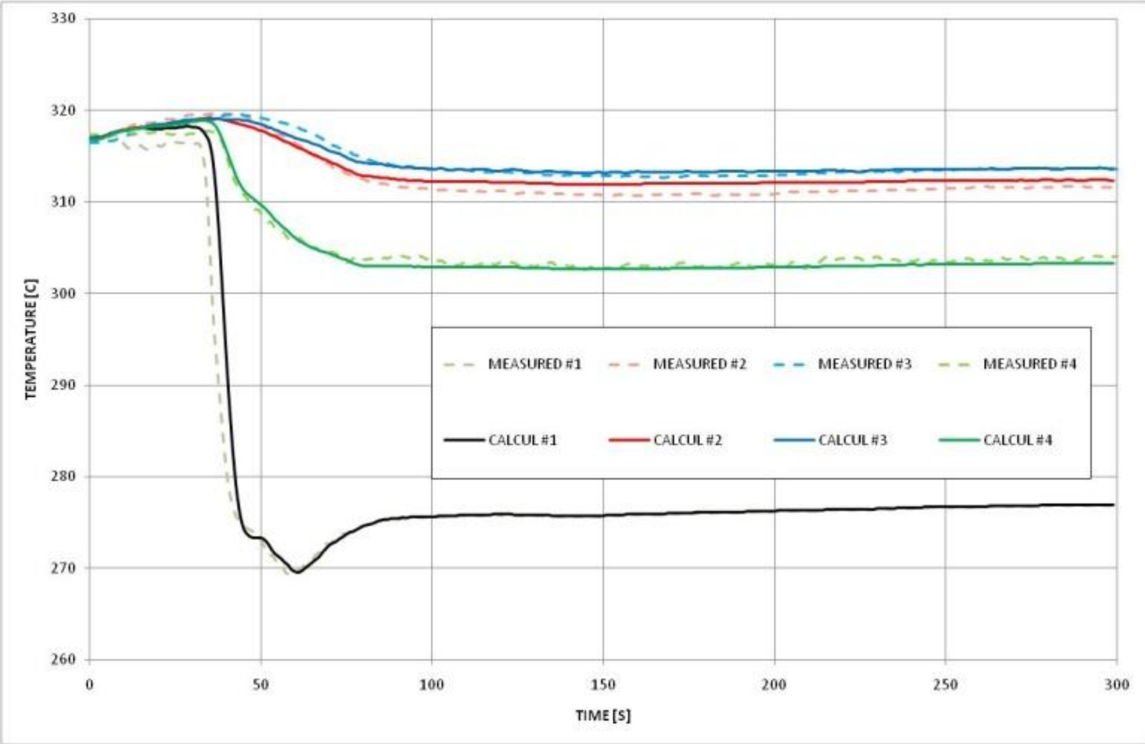

Fig. 3 Comparison of predicted and measured hot legs' coolant temperatures

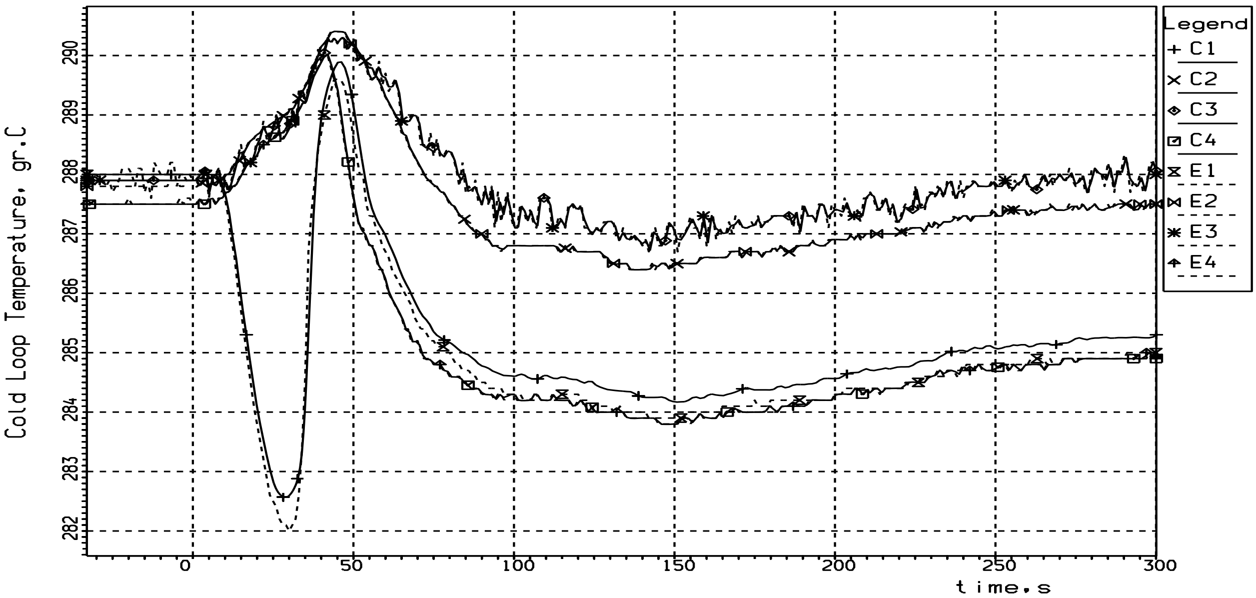

Fig. 4 Comparison of predicted and measured cold legs' coolant temperatures

Summary

The presented comparisons of the important global parameters show a very good fit to the available measured data. The cold and hot leg temperatures during the whole transient history do not exceed 0.6°C from the measurements which are reported to have 2°C measurement error. Detailed comparisons for the local (in core) measurements are discussed in [8].

The differences of the reactor power measurements on the base of SPND (fine) and NFC (global) data put a question on the necessity of refining of the control rod algorithm of VVER-1000 applying additional information from the SPND from the in-core monitoring system. It turns out that the small (2-3%) power increase around 80 s due to the decrease of the coolant temperature remains undetected by the NFC and so no control signal is generated for control rod movement initiation.

Acknowledgments

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