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HEXTRAN–SMABRE calculation of the VVER-1000 coolant transient-1 benchmark

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Abstract

The VVER-1000 coolant transient benchmark is intended for validation of couplings of the thermal hydraulic codes and three-dimensional neutron kinetic core models. It concerns switching on a main coolant pump when the other three main coolant pumps are in operation. The problem is based on an experiment performed in Kozloduy NPP in Bulgaria. In addition to the real plant transient, an extreme scenario concerning a control rod ejection after switching on a main coolant pump was calculated. At VTT the three-dimensional advanced nodal code HEXTRAN is used for the core dynamics, and the system code SMABRE as a thermal hydraulic model for the primary and secondary loop. The parallelly coupled HEXTRAN–SMABRE code has been in production use since early 1990s, and it has been extensively used for analyses of VVER NPPs. The SMABRE input model is based on the standard VVER-1000 input used at VTT. The whole core calculation is performed with HEXTRAN. Also the core model is based on earlier VVER-1000 models. Nuclear data for the calculation were specified in the benchmark. The paper outlines the input models used for both codes. Calculated results are introduced both for the coupled core system with inlet and outlet boundary conditions and for the whole plant model. Parametric studies have been performed for selected parameters.

Keywords: Reactor dynamics; Coupled codes; VVER-1000

1. Introduction

The VVER-1000 coolant transient benchmark (Ivanov et al., 2004) is intended for validation of couplings of the thermal hydraulic codes and three-dimensional neutron kinetic core models. It concerns switching on a main coolant pump when the other three main coolant pumps are in operation. The problem is based on an experiment performed in Kozloduy NPP in Bulgaria in 1992.

VTT participated in exercises 2 and 3 of the first phase of the V1000CT-1 benchmark. Exercise 2 was a coupled 3D neutronics/core thermal hydraulics response evaluation, with given boundary conditions. Exercise 3 was a best-estimate coupled code plant transient calculation. In addition to the real plant transient, an extreme scenario concerning a control rod ejection after switching on a main coolant pump was calculated. Since both the basic case and extreme scenario were quite mild transients, some additional variation calculations were performed.

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At VTT the three-dimensional advanced nodal code HEXTRAN is used for the core dynamics, and thermo-hydraulic system code SMABRE as a thermal hydraulic model for the primary and secondary loop. The parallelly coupled HEXTRAN–SMABRE code has been in production use since early 1990s, and it has been extensively used for analyses of VVER NPPs. The SMABRE input model is based on the standard VVER-1000 input used at VTT. Last plant specific modifications to the input model have been made in EU projects. The whole core calculation is performed with HEXTRAN. Also the core model is based on earlier VVER-1000 models. Nuclear data for the calculation were specified in the benchmark.

2. Three-dimensional reactor dynamic codes

2.1. Reactor dynamic codes at VTT

The coupled codes combine thermal hydraulic system codes and 3D neutron kinetics codes to perform best-estimate calculations of plant transients of nuclear power plants. These codes improve the capability to analyze plant conditions and are particularly needed in transient conditions due to strong coupling between the coolant flow in the primary circuit and the nuclear power generation in the reactor core affected by the reactivity feedback.

VTT has a comprehensive calculation system for reactor physics and dynamics for different types of reactors, Fig. 1. The three-dimensional core neutronics, heat transfer and thermal hydraulics solution method of the HEXTRAN (Kyrki-Rajamäki, 1995) code are based on coupling and extension of the steady state hexagonal core simulator HEXBU-3D and the one-dimensional thermal hydraulics code TRAB.

The two-group neutron diffusion equations are solved in HEXTRAN by a nodal expansion method in x–y–z geometry within the reactor core. A basic feature of the method is decoupling of the two-group equations into separate equations for two spatial modes, and reconstruction of group fluxes from characteristic solutions to these equations. The fundamental mode has a fairly smooth behaviour within a homogenized node, and the transient mode deviates significantly from zero only near material discontinuities. The nodal equations are solved with a two-level iteration scheme where only one unknown per node, the average of fundamental mode, is determined in inner iterations. The nodal flux shapes are improved in outer iterations by recalculation of the coupling coefficients.

The thermal hydraulic calculation of the reactor core is performed in parallel one-dimensional hydraulic channels, which can be further divided into axial sub-regions. Usually each channel is coupled with one fuel assembly. Parallel

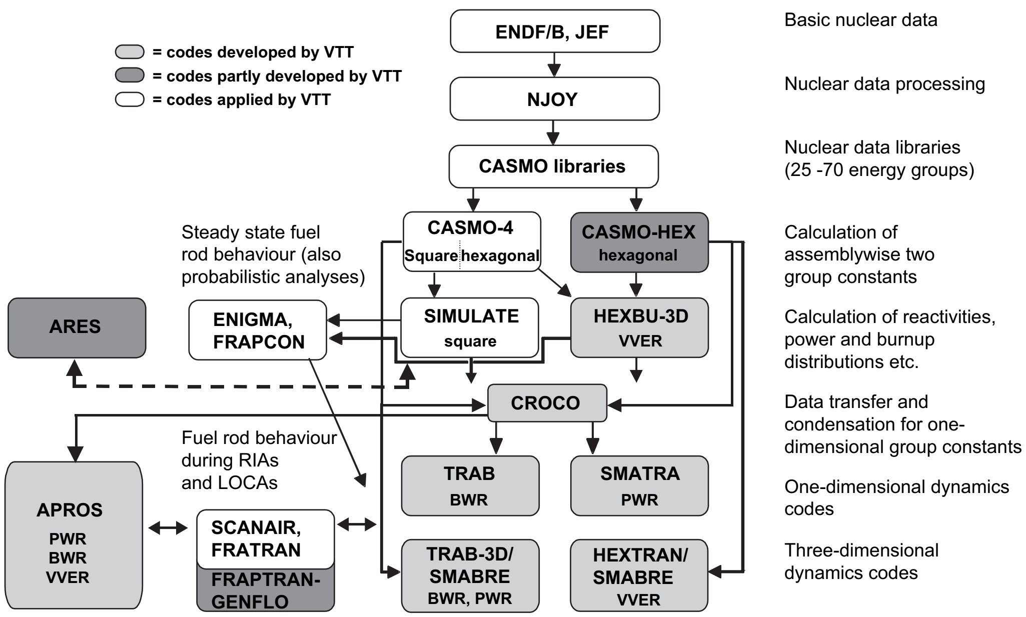

Fig. 1. VTT's calculation system.