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Comprehensive uncertainty and sensitivity analysis for coupled code calculations of VVER plant transients

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Abstract

The development of coupled codes, combining thermal-hydraulic system codes and 3D neutron kinetics codes, is an important step to perform best-estimate calculations for plant transients of nuclear power plants. For applications in safety analysis, these coupled codes should be validated by benchmark calculations and, preferably, by comparison with plant transient data from operating plants. In addition, the results should be supplemented by applying uncertainty and sensitivity analysis methods, which allow to identify relevant parameters of models and solution procedures affecting the results and to quantify their relative importance. Both objectives were part of the VALCO project. The aspect of validation is presented in [S. Mittag, et al., 2004. Neutron-Kinetic Code Validation against Measurements in the Moscow V-1000 Zero-Power Facility, in press; T. Vanttola et al., 2004. Validation of coupled codes using VVER plant measurements, in press], the aspect of a comprehensive uncertainty and sensitivity analysis for coupled code calculations is the topic of this contribution. The results and experiences obtained by the analysis for two plant transients in a VVER-440 and a VVER-1000, respectively, are presented and discussed.

1. Introduction

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 (NPPs). Such codes improve the capabil-

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Nomenclature

NPP	nuclear power plant
PIRT	phenomena identification and ranking tables
PWR	pressurized water reactor
VVER	pressurized water reactor designed in Russia (water/water energetic reactor)

ity to analyze plant conditions, which are determined by a strong coupling between the coolant flow in the primary circuit and the nuclear power generation in the reactor core affected by the reactivity feedback. Meanwhile, each thermal-hydraulic system code like ATHLET, CATHARE, RELAP or TRAC has been coupled with 3D neutron kinetics codes. The validation of these codes is part of international co-operations. Within the OECD framework, several benchmark problems have been defined and analyzed, e.g. the PWR main steam line break (MSLB) benchmark (Todorova et al., 2003) and the BWR turbine trip (TT) Benchmark (Solis et al., 2001). Corresponding work for VVER NPPs has been performed in projects with participants from MOE countries and Russia funded by the European Commission (EC). In project SRR1/95, data of plant transients from VVER-440 and VVER-1000 have been collected and evaluated for validation, and two plant transients have been chosen to perform detailed calculations by coupled codes. These transients are: a load drop of one turbo-generator in Loviisa-1, a VVER-440 (Hämäläinen et al., 2002) and a switch-off of one out of two working main feed water pumps in Balakovo-4, a VVER-1000 (Mittag et al., 2001). Within the VALCO project (Weiß et al., 2003), funded by the EC in the frame of the FP5 programme, the efforts of code validation have been continued (Mittag et al., 2004; Vanttola et al., 2004). In addition, a work package was dedicated to the comprehensive uncertainty and sensitivity analysis of coupled code calculations. The GRS uncertainty and sensitivity method based on the SUSA code package (Krzykacz et al., 1994; Hofer, 1999) was applied to both transients previously analyzed in the SRR1/95 project.

Chapter 2 describes the main steps of the uncertainty and sensitivity analysis method developed by GRS, which is based on the statistical code package SUSA. Chapter 3 includes a description of the plant experiment performed in Loviisa-1 NPP, the considerations to determine the list of uncertain parameters for this transient and the results of the uncertainty and sensitivity analysis. The results are completely presented to give an overview on the analysis. Chapter 4 includes corresponding information on a Balakovo-4 plant transient, but the presented results are limited to particular aspects of the results. General conclusions from the uncertainty and sensitivity analysis performed for these VVER plant transients are summarized in Chapter 5.

2. The GRS method for uncertainty and sensitivity analysis

The GRS method for uncertainty and sensitivity analysis consists, firstly, of a systematic identification of relevant physical processes and, secondly, of a probabilistic quantification of the uncertainty of corresponding parameters. The Monte Carlo simulation and the statistical evaluation of the results follow subsequently. The first step is very similar to the phenomena identification within a PIRT review (Boyack et al., 1990; Wickett et al., 1998), and needs a careful analysis of physical effects.

The practical application of the GRS uncertainty and sensitivity method (Krzykacz et al., 1994; Hofer, 1999) has the following steps:

1. Identification of the potentially relevant uncertainties.
   In this step, the physical models and their input parameters are identified which contribute to the uncertainty of the model. All parameters should be included, which could have the potential of relevant contributions, may be in combination with other parameters.
   Then follows the quantification of the state of knowledge uncertainty on the parameter level.
2. Definition of uncertainty ranges of parameters (minimum and maximum values).
3. Specification of subjective probability distributions over these ranges (uniform or any other appropriate probability distribution).
4. Identification and quantification of state of knowledge dependencies between parameters, if present.