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Uncertainty and sensitivity analyses of the Kozloduy pump trip test using coupled thermal–hydraulic 3D kinetics code

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Abstract

The modeling of complex transients in nuclear power plants (NPP) remains a challenging topic for best estimate three-dimensional coupled code computational tools. This technique is, nowadays, extensively used since it allows decreasing conservatism in the calculation models and performs more realistic simulation and more precise consideration of multidimensional effects under complex transients in NPPs. Therefore, large international activities are in progress aiming to assess the capabilities of coupled codes and the new frontiers for the nuclear technology that could be opened by this technique. In the current paper, a contribution to the assessment and validation of coupled code technique through the Kozloduy VVER100 pump trip test is performed. For this purpose, the coupled RELAP5/3.3-PARCS/2.6 code is used. The code results were assessed against experimental data. Deviations between code predictions and measurements are mainly due to the used models for evaluating and modeling of the Doppler feedback effect. Further investigations through the use of two "antagonist" uncertainty GRS and the CIAU methods, were considered in order to evaluate and quantify the origin of the observed discrepancies. It was revealed on one hand that relative error quantification discrepancies exist between the two approaches, and further enhancements for both methods are needed.

1. Introduction

The evaluation of complex phenomena in NPPs is closely related to the ability of determining the time-space core flux distribution as well as the flow field conditions and the associated effects from heat sources and heat sinks throughout the reactor coolant system. The recent availability of powerful computer and computational techniques has enlarged the capabilities of getting better realistic simulations of complex phenomena including multidimensional effects in NPPs. The application of coupled computer codes was recently identified as an area of high collective interest, especially for nuclear plant design and safety (D'Auria et al., 2004). Therefore, large international activities are in progress aiming to assess the capabilities of coupled codes and the new frontiers for the nuclear technology that could be opened by this technique. This later could be applied for different purposes. A typical example is the coupling of primary system thermal–hydraulic codes with 3D neutron kinetics codes. Other cases include coupling of primary system thermal–hydraulics with structural mechanics, computational fluid dynamics (CFD), nuclear fuel and containment behaviors. The capabilities of the coupled code calculations in simulating, in a best estimate (BE) way, nuclear plant behavior under a wide variety of transient and accident conditions have been largely investigated through several international programs. To pursue this goal, a series of code qualification processes is carried out. This could be performed, for instance, through the consideration of experimental data issued from operational NPP data. In the current framework, a validation of the coupled code technique against a well-documented Kozloduy VVER1000 pump trip is investigated (Vanttola et al., 2005). The transient

Abbreviations: ADF, assembly discontinuity factor; BE, best-estimate; CRISSUES, critical issues in nuclear reactor technology: a state of the art report; EC, European Commission; LBLOCA, large break loss of coolant accident; LOCA, loss of coolant accident; LOFW, loss of feed water; MSH, main steam header; NEA, nuclear energy agency; NPP, nuclear power plant; OECD, organization for economic co-operation and development; PVM, parallel virtual machine; SBLOCA, small break loss of coolant accident; VALCO, validation of coupled neutronics/thermal–hydraulics codes for VVERs; VVER, water-cooled water-moderated energy reactor; 3D, three-dimensional

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Table 1
Sequence of main events

Time (s)	Event
0.0	MCP #3 off, while MCP #1, MCP #2, and MCP #4 are ON The reactor power is 65% of its nominal value
4.0	MCP #1 of the 1st primary loop is switched off (one of three operating MCPs)
4.3	SCRAM signal is activated by the low flow rate logic trip
4.5	10th group of control rods starts its downward movement from 71.83%
20	Set point "low level in SG1" is reached and the second auxiliary feed water pump (AFWP-2) is put into operation
31	10th control group reaches the lower position of 55%
36	10th control group is withdrawn from its 55% position to its final course position around 73% in 300 s
52	Switch off of the first pressurizer heater group
58	First opening of the pressurizer spray valve (maximum 21% at 75 s, closure at 83 s)
88	Second opening of the pressurizer spray valve (maximum 7% at 98 s, closure at 103 s)
380	End of test

under consideration concern a relatively common operational transient and could be well-simulated using a best estimate thermal–hydraulic system code. However, the huge amount of experimental data gathered during the test including 2D radial power distribution makes it valuable for coupled code validation purposes. Results of calculation were assessed against experimental data and also through the code-to-code comparisons. However, computer codes, as any simulation tool, are affected by errors arising from the unavoidable approximations connected with the modeling requirements and limitations (Bousbia Salah, 2004). Thus, sensitivity and uncertainty analyses must be carried out to supplement the code results. In order to identify and assess the observed deviations between the measurement and the coupled code calculations a series of uncertainty and sensitivity calculations using statistical GRS (Langenbuch et al., 2005) and the deterministic CIAU (D'Auria and Giannotti, 2000) methods are considered. The use of these "apparently" antagonist methods allows getting a global vision about the applicability of these methods for best estimate tools (Wickett et al., 1998).

2. Transient description

The reference transient is the commissioning experiment at the sixth unit of Kozloduy NPP (VVER-1000) documented in the framework of the EU FP5 (VALCO) project (Weiss et al., 2003). The main transient sequences are summarized below in Table 1.

3. Calculation models and hypothesis

To perform a numerical simulation of the considered experimental test, the coupled RELAP5 Mod3.3/PARCSV2.6 code was used. The codes are run separately through a parallel processing way using the parallel virtual machine (PVM) tool. In this scheme, practically, no modifications of the codes programs are carried out. PARCS (Joo et al., 1998) utilizes the thermal–hydraulics solution for the moderator temperatures/densities and fuel temperatures calculated by RELAP5 (Fletcher and Schultz, 1995) to incorporate appropriate feedback effects. Likewise, RELAP5 takes the space-dependent power calculated in PARCS and solves the thermal–hydrodynamic conservation equations. The temporal coupling is explicit in nature, and the two codes are locked into the same time step.

3.1. Neutron kinetic modeling

For hexagonal geometry, the PARCS solution is obtained using the triangular polynomial expansion nodal (TPEN) method (Downar et al., 2004). In the TPEN method, the three-dimensional problem is first decoupled into a radial and an axial problem.

In the current kinetic modelling, each of the core fuel assembly (FA) is considered to be homogeneous. Therefore, the solution is obtained by discretizing the reactor core into several homogeneous nodes. Each radial node is characterized by its nuclear properties that depend on the fuel material, and the burnup (exposure, spectral history, and burnable poison history). The reactor core is loaded in 60° rotational symmetry. Such a core symmetry is typical for VVER-1000 reactors (see also Fig. 1). According to the available measurements and calculations, there is no big difference in the burnup between FA having the same symmetry position. Therefore, 29 fuel assembly types including one reflector element are identified to represent the kinetics of the whole core.

Axially, the core is subdivided into 20 nodes, where the first and the last nodes correspond to the lower and upper reflector, respectively. In total 507 kinetic nodes or compositions (28 × 18 nodes in the fuel and 3 reflector nodes representing the lateral, the lower and upper plenum reflector) are considered to represent the kinetic behavior of the core.

3.1.1. Cross section modeling

The cross section formalism of the PARCS code is based upon a set of base macroscopic cross sections generated at a reference kinetic (burnup) and thermal–hydraulic conditions (T_f, ρ_m). For each composition are assigned the "reference" or the base cross sections data (including scattering, absorption, fission cross sections, for each rodded (presence of control rod in the homogenized lattice) and unrodded (homogenized cell without the control rod)), and four sets of partial (derivative) cross sections to describe the boron and the thermal–hydraulic feedback effects. Therefore, during the steady or transient state calculation, the macroscopic cross sections are derived through the following formula:

$$\begin{aligned}\Sigma(B, T_\mathrm{f}, T_\mathrm{m}, \rho_\mathrm{m}) = \Sigma_\mathrm{Ref} + a_1(B - B_\mathrm{Ref}) \\ + a_2\left(\sqrt{T_\mathrm{f}} - \sqrt{T_\mathrm{f}^\mathrm{Ref}}\right) + a_3(T_\mathrm{m} - T_\mathrm{m}^\mathrm{Ref}) \\ + a_3(\rho_\mathrm{m} - \rho_\mathrm{m}^\mathrm{Ref}) \qquad (1)\end{aligned}$$