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Application of coulped code method for the analysis of VVER-1000 coolant transient benchmark

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Abstract

In the light of the sustained development in computer technology, the possibilities of code capabilities have been enlarged substantially. Consequently, advanced safety evaluations and design optimizations, which were not possible few years ago, can now be performed. Nowadays, it becomes possible to switch to new generation of computational tools, namely, coupled code (CC) technique. The application of such method is mandatory for the analysis of transient events where strong coupling between the core neutronics and the primary circuit thermal-hydraulics exits, and more especially when asymmetrical processes take place in the core leading to local space-dependent power generation. Through the current study, a demonstration of the maturity level achieved in the calculation of 3-D core performance during complex accident scenarios is emphasized. The study is followed by a typical application through which the main features and limitations of this technique are discussed.

Keywords: Coupled code technique; Best-estimate tools; Nuclear safety analysis

1. Introduction

Computer codes are widely used for safety analysis within the framework of licensing and safety improvement programs of existing NPPs, for better utilization of nuclear fuel and higher operational flexibility, for justification of lifetime extensions, development of new emergency operating procedures, analysis of operational events, and development of accident management programs. The recent availability of powerful computer and computational techniques has enlarged the capabilities of making more realistic simulations of complex phenomena in NPPs and more precise consideration of multidimensional effects.

The capabilities of the coupled code calculations have been largely investigated through several international programs under a wide variety of transient and accident conditions. These activities include the OECD/NEA benchmarks (Ivanov et al., 2004). However, notwithstanding the complexity of these codes and the level of the present scientific knowledge, a computer code cannot be expected to accurately model phenomena that are not yet fully understood by the scientific community. In general, the results of code predictions, specifically when compared with experimental data reveal deviations. These discrepancies could be attributed to several reasons as model deficiencies, approximations in the numerical solutions, nodalization effects, imperfect knowledge of boundary and initial conditions. Reliability prediction of best-estimate (BE) coupled code tools includes the need of a general code qualification process. This could be performed through the consideration of experimental data issued from operational NPP data, integral test facilities (ITF) or separate effects test facilities (SETF) validation matrices (Aksan et al., 1993). In addition, nodalization qualifications, as well as qualitative and quantitative accuracies of the code results, are also needed for the code qualification process (D'Auria and Galassi, 1998). Therefore, it is necessary to investigate the uncertainty of the code results and the sensitivity effect of the most effective parameters.

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Nomenclature

ATWS	Anticipated transient without scram
BDBA	Beyond design basis accident
BE	Best-estimate
BEPU	Best-estimate plus uncertainty
BWR	Boiling water reactor
CFD	Computational fluid-dynamics
CIAU	Code with capability of internal assessment of uncertainty
CPU	Central process unit
CSC	Cross section code
CSNI	Committee on the safety of nuclear installations
DBA	Design basis accident
EC	European commission
GRS	Gesellschaft fuer Anlagen- und Reaktorsicherheit
IAEA	International atomic energy agency
ITF	Integral test facility
LOCA	Loss of coolant accident
LBLOCA	Large break loss of coolant accident
LWR	Light water reactor
MOX	Mixed U-Pu oxide nuclear fuel
NEA	Nuclear energy agency
NKC	Neutron kinetic code
NPP	Nuclear power plant
OECD	Organization for economic co-operation and development
PWR	Pressurized water reactor
RIA	Reactivity initiated (or induced) accident
TH	Thermal-hydraulic
THSC	Thermal-hydraulic system code
VVER	Water-cooled water-moderated energy reactor
3D or 3-D	Three-dimensional

The lack of immediate industrial interest on the coupled code technique, owing to the natural caution and conservatism from the regulatory bodies in accepting innovations, prevented so far the full exploitation of the considered technique (D'Auria et al., 2004, Bousbia Salah and D'Auria, 2007; Ivanov and Avramova, 2007). An attempt is made herein to emphasize the state-of-the-art and the main features of such computational tools through a typical application in a VVER1000 nuclear reactor.

2. Coupled computational tools

The need of coupled code for safety analyses calculations is nowadays enough cute. The approach could be applied for different purposes and using appropriate computational tools. 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 behaviour and containment behaviour (Aumiller et al., 2001).

For thermal-hydraulic kinetics dynamic simulation of a given nuclear reactor a general procedure is generally performed as illustrated in Fig. 1 (Stamm'ler and Abbate, 1983). The basic structure of the model consists of a module to generate macroscopic group constants, followed by a module to calculate the core flux, and power distribution. The power distribution can then be used as input to a thermal-hydraulics module which solves the heat transfer and fluid flow equations to determine the temperature and coolant density distribution in the core. This latter information is required for the generation of macroscopic group constants (feedback to the earlier module and possible iteration will be necessary).

When a critical core configuration is achieved ($k_{\mathrm{eff}} \approx 1$), depletion calculation, describing fuel isotope evolution and fission product, is buildup using as input the flux distribution for a given period of reactor operation. After this depletion step, the new isotopic composition of the fuel is returned to the group-constant module, and the entire calculation is repeated until criticality is reached. After a series of depletion steps, the fissile inventory drops sufficiently low that the control system can no longer return the core to criticality (end of cycle).

Basically, to carryout thermal-hydraulic and kinetic dynamic simulations the following codes, as shown in Fig. 2, are applied (D'Auria et al., 2004).

- Thermal-hydraulic system code (THSC).
- Neutron kinetics codes (NKC).
- Code for deriving neutron kinetics cross sections (CSC or cross section code). The CSC provides the macroscopic cross section for the NKC codes; it is generally used out of line while the THSC and the NCK are run simultaneously.

3. Code qualification and uncertainty evaluation

In order to perform adequate coupled 3D NKC–THSC calculations capable to gain reliable results, a consolidated procedure should be used. The steps reported below outline the proposed procedure. However, four fundamental preconditions shall be fulfilled for the correct application of coupled codes simulation of transient scenarios expected in NPP.

- The code should be frozen to ensure that no unjustified modifications of the constitutive models would alter the results. The code must be able to correctly simulate almost all the transient dominant phenomena using the models of the adopted frozen version.