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A methodology for the coupling of RAMONA-3B neutron kinetics and TRAC-BF1 thermal-hydraulics

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Abstract

The initial objective of this project was to directly couple the RAMONA and TRAC codes running on different PCs. The idea was to use the best part of each one and eliminate some of their limitations and widen the applicability of these codes to simulate different BWR and system components. It was required to try to minimize the amount of changes to present code subroutines and calculation procedures. If possible, just substitute values obtained in the parallel code. Preliminary results indicated that using a CHAN component of TRAC to model thermal-hydraulic phenomena for each neutronic channel modeled in RAMONA is rather difficult. Large amounts of CPU time consumption are obtained and lots of PCs would make this solution difficult, besides considerable large transients are introduced by the differences in thermal-hydraulic results of these codes. The substitution of the thermal-hydraulics of RAMONA, by the TRAC channel calculations, is possible but simulation of a null transient on both codes must be planed and a gradual change must be controlled by an additional supervisory subroutine. An indirect coupling of these codes, it is therefore proposed, in order to eliminate most of these limitations. In this indirect coupling, a thermal-hydraulic model of the average tube in a bundle and the global channel cooling fluid dynamics is programmed for each neutronic

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channel while the global reactor vessel and core is modeled by TRAC with just four channels and four rings. Results are more reliable, coupling is simpler and faster simulations are possible.

1. Introduction

The operation and the development of new nuclear power plants require the use of advanced computational methods to increase plant competitiveness. Of major interest to nuclear engineers is the use of Best-Estimate methods that can predict important safety margins and their associated uncertainties.

A number of projects to upgrade the quality of safety analysis software have been undertaken by the nuclear industry. During the last years, the coupling of different 3D neutronics code like BIPR8 (KI), DYN3D (FRZ), KIKO-3D (KFKI), QUABOX/CUBBOX (GRS), ENTREE/TRAC-BF1 (USA) and TRAC-PF1/NEM (USA) was implemented in order to simulate LWR, VVER and RBMK core conditions (Akitoshi et al., 2000; Forschungszentrum and Cacuci, 2000; Grandrille et al., 2000; Mittag et al., 2000; Nadejda and Kostadin, 2001).

Langenbuch et al. (2000) presents an overview on the development of coupled system of 3D neutronics and fluid-dynamic system codes. The objectives and representative results of international activities to validate such coupled codes are described. In addition, the experiences from applications in accident analysis are summarized and further improvements are recommended.

The United States Nuclear Regulatory Commission (USNRC, Mahaffy et al., 2000) has developed the thermal-hydraulic analysis code TRAC-M to consolidate the capabilities of its suite of reactor safety analysis codes (TRAC-P, TRAC-B, RAMONA, and RELAP5). One of the requirements is that it supports parallel computations to extend code functionality and to improve execution speed. A flexible request driven Exterior Communication Interface was developed for use with the consolidated code and has enabled distributed parallel computing. Parallel applications of the code were reported for several multiprocessor platforms and operating systems.

In this work, two schemes are proposed and tested in order to couple the RAMONA neutronic module and the TRAC thermal-hydraulic module. The main goal is to obtain a computer tool to perform detailed analysis in traditional BWR and advanced BWR reactors.

TRAC and RAMONA have been used widely for BWR transient and accident analyses and very good results are reported in different publications. RAMONA is a 3D coarse mesh neutron flux calculator solving the time-dependant 11/2 neutron diffusion equation. Neutron cross-sections are updated at each mesh-cell by time updated coolant and fuel conditions obtained from a model of the main reactor vessel components and fuel channels in the core. The thermal-hydraulics model solves four conservation equations: the mixture energy and the masses of two phase flows in each cell, in addition to an integral momentum for each closed loop trajectory in the reactor vessel. All physical properties and closure relationships are computed