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High-fidelity multi-physics system TORT-TD/CTF/FRAPTRAN for light water reactor analysis

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ABSTRACT

A need exists in the nuclear industry for higher-fidelity tools for light water reactor (LWR) analysis, due to increasing core heterogeneity and higher burnup of fuels. In order to address this need, a high-fidelity multi-physics (HFMP) system has been developed at the Pennsylvania State University (PSU). It consists of three codes – CTF for thermal hydraulics, TORT-TD for neutron kinetics, and FRAPTRAN for fuel performance. FRAPCON, which is applied to long-term steady-state fuel performance, is left separate and not modified, but is relevant to the system because it generates the initial conditions used in FRAPTRAN. These codes have been combined into a system in which they are coupled by means of serial integration. FRAPTRAN is the latest addition to the system while the initial coupling of TORT-TD and CTF was verified in different applications. Recent efforts have been directed at the design of an object-oriented system of interfaces for the coupled codes, by which the main program may control them in terms of high-level functionality. Further modifications to the system include the ability to use coolant-centered rather than fuel-centered channels, and the ability for TORT-TD to use a time step size that differs from that of CTF. The obtained results verify this new coupling, as well as demonstrate the advantages of using a fuel-performance code for modeling fuel rod feedback.

1. Introduction

The trend in nuclear core design has been toward longer refueling cycles with higher burnups, as well as greater heterogeneity due to MOX fuel and burnable absorbers. Such heterogeneities result from the use of mixed-oxide (MOX) fuel, as well as burnable absorbers. Furthermore, longer fuel cycles are resulting in higher burnup, increasing the importance of accurately modeling the material properties and behavior of the fuel. Finally, given that the overall behavior of a reactor system is determined by the synergistic interaction of many physical phenomena, all relevant aspects of the physics must not only be simulated in sufficient detail, but must incorporate feedback from other aspects of the physics. This has resulted in several modeling challenges in core design and accident simulations. The presence of adjacent regions with significantly different cross sections causes a large angular dependence of the neutron flux, for which the diffusion approximation is not adequate. Additionally, the neutron energy spectrum becomes more complex due to the overlapping influence of multiple material zones, with this spectrum affecting the isotopic composition of burned fuel. Higher burnups furthermore increase the importance and difficulty of correctly predicting fuel behavior, as it is necessary to account for the burnup dependence of properties while ensuring the integrity of the fuel rods.

Such developments must be supported by the creation of new software tools which will be applied to the licensing and deployment of new fuel designs, as well as core design with improved safety margins. Such tools must have a sufficiently high level of fidelity to predict power distributions and fuel behavior under normal and accident conditions, in light of the previously-mentioned complexity. In particular, fuel failures must be accurately predicted and avoided.

Research and follow-up developments have been performed elsewhere on improved fuel performance codes for whole core calculations (Rossiter, 2011) as well as on high-fidelity multi-physics code systems (Valtonen et al., 2002). The later reference reports on previous work at VTT Energy in Finland, which created a coupling between FRAPTRAN and the high-efficiency non-iterative thermal hydraulics code GENFLO. The purpose was to improve the treatment of flow regimes and heat transfer coefficients used in FRAPTRAN, as the built-in models are much simpler than those in GENFLO. Thus, FRAPTRAN is the master code, and calls GENFLO. GENFLO – FRAPTRAN has a number of limitations:

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- As in FRAPTRAN, only a single rod may be modeled at a time. Cross-flow cannot be taken into account.
- There is no feedback to the neutron kinetics calculation. This is determined by separate codes, and the power distribution is set as a boundary condition.
- There is no feedback to the overall reactor coolant system, and thus no feedback to the pressure, flow, and enthalpy boundary conditions.

The purpose of the high-fidelity system developed in this work differs from GENFLO – FRAPTRAN in that it aims to use the fuel performance code to increase the accuracy and capabilities of the thermal hydraulics code, rather than vice versa. With coupling to CTF and TORT-TD, the first two limitations will not apply.

The modeling of fuel behavior is a multi-physics, multi-scale problem, containing the coupled interaction of many physical phenomena on a variety of temporal and spatial scales. The power distribution is dependent in part of feedback from fuel temperature – a parameter for which a great deal of uncertainty exists. This fuel temperature is in turn dependent on the current power, as well as the power history, which influences the properties of the fuel through burnup. The thermal resistance between the cladding and fuel surface is a particularly important determinant of the fuel temperature. This depends on dimensional changes in the fuel and cladding due to densification, swelling, thermal expansion, and mechanical deformation. It also depends on the gap gas pressure and composition, and the texture of the fuel and cladding interior surfaces. The cladding can deform and fail due to a buildup of pressure from the release of fission gas by excessive fuel temperatures, and this is highly influenced by the clad temperature, which may become high due to insufficient cooling under accident conditions. This may cause the cladding to balloon, reducing the thermal–hydraulic flow area.

Current codes are limited in their ability to take into account such phenomena. Although neutron-kinetic core calculations are typically coupled to thermal hydraulic models, fuel performance analysis is done separately, with boundary conditions passed unidirectionally from the core calculation to a fuel performance code which concentrates on a single rod. The fuel performance code's more detailed model may result in expansion of the cladding, a different time-dependent rate of clad-to-coolant heat transfer, and most importantly, a different fuel temperature. None of these is given as feedback to the thermal hydraulics and neutron kinetics calculations.

Furthermore, the current methodology is based on nodal diffusion calculations with homogenized fuel assemblies. This is inadequate to account for the large angular variation of the flux near boundaries between very different material regions, such as burnable absorbers, MOX fuel, or fuel with different burnup. The infinite lattice assumption made when calculating homogenized assembly cross sections is currently considered the largest source of error in reactor physics calculations. This lack of modeling fidelity increases the uncertainty of the pin power distribution, as well as the spectrum, affecting the burnup rate, the isotopics of the depleted fuel, and the behavior of the fuel over the length of the cycle as well as during transient and accident conditions.

In order to develop simulation tools that are adequate to model the multi-physics/multi-scale behavior of a reactor while reducing safety margins, it is necessary to answer several questions:

(1) What level of fidelity with respect to angular, energy, and spatial discretization of the neutron flux is adequate to predict the pin power distribution with a sufficient degree of precision?
(2) What level of fidelity in fuel rod modeling is adequate (i.e. which fuel phenomena must be taken into account) in feedback to thermal hydraulics and reactor physics?
(3) What is the impact of uncertainties (in power, flow, material behavior, etc.) on the coupled interaction of reactor phenomena?

In order to address these questions, the high-fidelity multi-physics ((HFMP) system was developed at the Pennsylvania State University (PSU), consisting of CTF (PSU's version of subchannel thermal hydraulics code COBRA-TF), TORT-TD (a discrete-ordinates neutron kinetics code), and transient fuel performance code FRAPTRAN. The system has an increased level of fidelity over current methods by (1) modeling on a pin-wise, rather than assembly-wise, scale, (2) using the discrete ordinates method rather than diffusion approximation to model the angular dependence of the neutron flux, and (3) using fuel rod feedback from a fuel performance code rather than the simplified model in a thermal hydraulics code. Multi-directional feedback is completely taken into account:

- The neutron kinetics code TORT-TD receives coolant and fuel conditions for cross section feedback.
- The thermal hydraulics code CTF receives clad temperature, heat flux, and changes in clad dimensions from the fuel performance code FRAPTRAN.
- The fuel performance code FRAPTRAN receives power from TORT-TD and thermal–hydraulic rod surface conditions from CTF.

First, the developed multi-physics code system will be described, followed by presenting the obtained results on multi-pin arrays supplemented by a comparative analysis demonstrating the differences in results caused by using FRAPTRAN over CTF's less detailed built-in fuel rod models.

2. Description of code system

The multi-physics code system, developed in the course of this work, directly incorporates three codes into a single executable: TORT-TD for neutron kinetics, CTF for thermal hydraulics, and FRAPTRAN for fuel performance. The multi-group Discrete Ordinates code TORT-TD (TORT-time-dependent) (Seubert et al., 2008) was developed at Gesellschaft für Anlagen und Reaktorsicherheit (GRS) based on TORT, but has been modified to handle time-dependence and reactor kinetics phenomena such as delayed neutrons. Additionally, it has a different input format which is designed for reactor calculations. It is applied to the pin-by-pin modeling of reactors, using homogenized-pin-cell cross sections that are produced by a Lattice Physics code. Before being coupled with CTF, TORT-TD was coupled with the thermal–hydraulic system code ATHLET. COBRA-TF (Coolant Boiling in Rod Arrays – Two Fluid), a subchannel-based thermal hydraulics code, was developed at Pacific Northwest National Laboratory (PNNL) for the U.S. Nuclear Regulatory Commission (NRC) for best-estimate analysis of design basis accidents and anticipated transients in LWRs. CTF is PSU's improved version, which has in recent years undergone significant improvement in both efficiency and modeling of two-phase flow (Salko and Avramova, 2013). CTF's modeling advantages include a three-field representation of vapor, continuous liquid, and liquid droplets, the ability to model fully 3 dimensional heat conduction, and dynamic gap conduction between the fuel pellet and cladding. FRAPCON-3 and its companion code FRAPTRAN were developed by the NRC to simulate fuel performance under steady-state and transient conditions, respectively. While FRAPCON-3 is primarily used for auditing vendors' fuel performance codes, FRAPTRAN has been used to develop

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