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## TOPIC ⑤

## MULTI-PHYSICS AND UNCERTAINTY ANALYSIS

### 5.1 From best estimate tools to direct coupling approach

### From best estimate coupling towards first principles high-fidelity multi-physics

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## Background

The main tools for performing design and safety analyses of nuclear power plants (NPPs) are computer codes for simulation of physical processes. The current trends in nuclear power generation and regulation are to perform design and safety studies by "best-estimate" codes that allow a realistic modeling of nuclear and thermal-hydraulic processes of the reactor core and the entire plant behavior including control and protection functions. These best-estimate results should be supplemented by uncertainty and sensitivity analysis. Significant progress was achieved by the development of coupled codes incorporating full three-dimensional (3D) reactor core models into thermal-hydraulic system codes. Such codes are needed to perform best-estimate calculations of interactions between the core behavior and the thermal-hydraulic behavior of the plant dynamics. The development and application of these advanced codes was supported by the continuous growth of computer capabilities. The code development was accompanied by international activities on the comprehensive validation and uncertainty quantification of such coupled codes.

The efficient and safe operation of current reactor fleet and the design of next generation reactor concepts along with the continuing computer technology progress stimulate the development, qualification and application of first principles high-fidelity multi-physics methodologies for reactor analysis. The current efforts have been focused on extending the analysis capabilities by coupling models, which simulate different phenomena or different reactor components, as well as on refining the scale and level of detail of coupling.

## Best-estimate approach

Realistic methods are referred to as "best-estimate" calculations, implying that they use a set of data, correlations, and methods designed to represent the phenomena, using the best available techniques. The regulations also require that the uncertainty in these calculations be evaluated.

Over the last two decades many organizations from different countries have accumulated experience in developing, validating and applying coupling methodologies for reactor design and safety analyses. Major requirements for coupling of thermal-hydraulics system codes and neutronics models are summarized in [1]. The methodologies needed to meet these requirements have reached maturity and have several common features [2]:

a) The 3D core neutronics model is based on two-group coarse-mesh diffusion approach using nodal nuclear data (library of parameterized nodal equivalent parameters);

b) The spatial neutronics/thermal-hydraulic coupling is performed on assembly/channel level;

c) The temporal neutronics/thermal-hydraulic coupling is performed in the operator-splitting approach using nested loop iteration (fixed point iterations) for steady state simulations and sequential parameter exchange (explicit coupling) at each time step for transient simulations;

d) The evaluation of safety related parameters on pin/sub-channel level for the hot assembly/channel is performed via pin-power reconstruction combined with sub-channel calculations using boundary conditions provided by the coupled core model;

e) Separate utilization of deterministic methods (for cross-section uncertainty propagation in $k_{eff}$ for criticality neutronics calculations) and statistical methods (for thermal-hydraulics and coupled thermal-hydraulics/neutronics calculations). In statistical methods in order to incorporate uncertainties into the process usually many runs of the computer code are required and the result is a range of results with associated probabilities.

Appropriate benchmarks have been developed in international co-operation led by the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD) that permits testing the neutronics/thermal-hydraulics coupling, and verifying the capability of the coupled codes to analyze complex transients with coupled core/plant interactions [3]. These benchmarks provide a validation basis for the current generation of coupled best-estimate codes.

The modeling of the 3D reactor core together with the thermal-hydraulics of the coolant system is mandatory for accident conditions determined by a tight coupling of neutron kinetics and thermal-hydraulics, because the core boundary conditions are consistently derived from the entire plant simulation. In case of asymmetric perturbations an adequate modeling of mixing is required. Another aspect is that only the coupled 3D core model provides a realistic description of the neutronics and thermal-hydraulic phenomena within the core and takes correctly into account the spatial feedback effects determining local and integral power generation. Therefore, the application of coupled codes effectively contributes to reduce the uncertainties in plant transient calculations.

These codes are able to predict the non-symmetrical core power perturbations in a more realistic manner and they can predict safety margins more accurately by means of 3D core model with a spatial resolution at fuel assembly level for a wide range of operational transients or postulated accidents. The approximation level is acceptable, however, the safety-relevant parameters, that determine the accident consequences, such as fuel rod enthalpy, departure from nucleate boiling ratio (DNBR), burn-out, maximum fuel rod cladding temperature, fuel rod centerline temperature, etc., have to be evaluated at local conditions i.e. in terms of a single rod (pin) response rather than based on an assembly-wise response.

**High-fidelity multi-physics**

There is a need to extend the model capability of current "best-estimate" coupled codes for design and safety assessment in such a way that the prediction of the fuel rod response is possible based on local parameters. Subsequently, the current developments are towards performing whole core high-fidelity multi-physics calculations on pin/sub-channel level and even on sub-pin level in order to understand and predict key aspects of fuel and cladding performance. Such capabilities allow determining radial, axial and azimuthal flux distribution, energy deposition, coolant and fuel temperature distributions for different pin cells/sub-channels (fuel rod with surrounding coolant) at different core locations. High-fidelity multi-physics tools have been developed as part of two large projects – NURESIM and NURISP in Europe (supported by European Commission) [4] and Consortium for Advanced Simulations of Light Water Reactors (CASL) in USA (supported by the Department of Energy) [5].

The recent advances and developments in reactor design and safety analysis towards more physics-based high-fidelity simulations are:

a) For deterministic neutronics methods replacing two group diffusion neutronics solvers with multi-group transport solutions [6]. Coupling continuous energy Monte Carlo codes with thermal-hydraulic solvers for reference coupled solutions [7];

b) Coupling spatially on pin/sub-channel level (scale) and even on within pin heterogeneity level, which requires the implementation of improved and flexible coupling methodologies Such refined coupled simulations can be performed in direct and embedded manner (utilizing multi-level coupling schemes) [2];

c) Efficient utilization of High Performance Computing (HPC) – utilization of modern and user friendly simulation platforms and parallel computing. This subject will be discussed in the lecture in section 5.2 "Advanced multi-physics using HPC capabilities";

d) Implementation of fully implicit time-integration method for simultaneous solution of the coupled non-linear system [8];

e) Coupling high-fidelity neutronics, hydraulics and fuel modeling in reactor core simulations as shown in Figure 1;

f) Further development of sensitivity and uncertainty analysis capabilities for comprehensive coupled code simulations with nonlinear feedback mechanisms. This subject will be covered in the lecture of session 5.3 Sensitivity and evaluation for Neutronics/Thermal-Hydraulic coupled solutions;

g) Establishing of comprehensive OECD Light Water Reactor Uncertainty Analysis in Modeling benchmark framework [9] for uncertainty propagation through multi-physics multi-scale calculations in order to compare different uncertainty analysis methods.

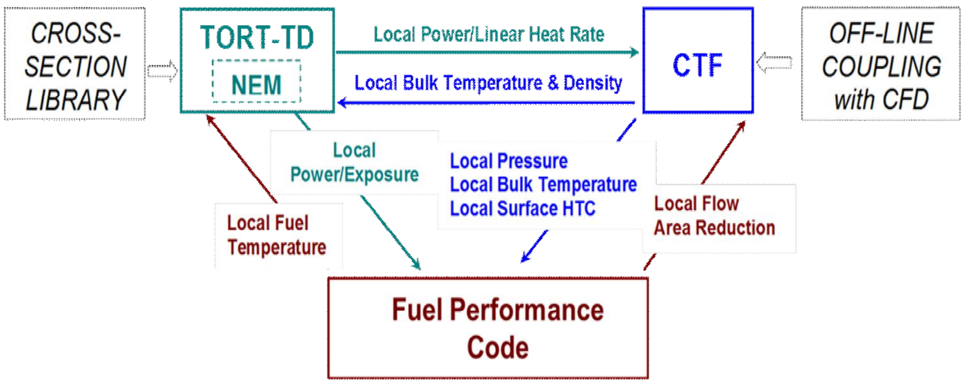


**Figure 1. Coupling Approach: Neutronics/Pin Mechanics/Thermal-Hydraulics**

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