A PROCEDURE FOR COUPLED NEUTRON PHYSICS / THERMAL HYDRAULIC CALCULATION OF THE PINWISE POWER DISTRIBUTION WITHIN A FUEL ASSEMBLY

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1. Introduction

Most of the 3D neutron kinetic models of coupled 3D-kinetics/thermal hydraulic system codes employ in radial direction planar meshes of the size of one fuel assembly (FA) or part of them. Since the safety-relevant parameters that determine the accident consequences, such as fuel rod enthalpy, departure from nucleate boiling ratio (DNBR), maximum fuel rod cladding temperature, etc. have to be evaluated in terms of a single rod rather than assembly-wise response, there is a need to extend the simulation capability of such codes from the current FA- to a fuel pin-based approach. At the Research Center Karlsruhe, Institute of Reactor Safety, the coupled codes RELAP5/PARCS [1] and TRACE/PARCS [2] are being used and qualified. The nodal transient core reactor simulator PARCS contains a transient pin power reconstruction model [3]. To take profit of this model in the frame of safety evaluations, additional developmental effort is needed e.g. coupling with a sub-channel code for both steady state and time-dependent calculations. In addition, the PARCS pin power reconstruction model expects as input “form functions” which describe the specific heterogeneity within a FA that can be predicted by transport codes. In such approaches, the radial discretization is very detailed while in axial direction typically one representative value for parameters of interest (fuel, clad and coolant temperature) is assumed. This is also a common approach in pin-wise exposure calculations [4].

With the aim of improving the prediction accuracy of form functions for the pin power reconstruction method of PARCS, a coupling procedure is being developed. It includes the implementation of a subchannel code COBRA-TF [5] into the modular system KAPROS-E [6], where the 2D/3D transport program DANTSYS [7] is already a module. In this paper, the first exploring results obtained with the coupled system KAPROS-E/DANTSYS/COBRA-TF for a PWR fuel assembly of type FA 18x18-24 will be presented and discussed. Further developmental and validation needs are outlined.

2. Developed coupling procedure

The KArlsruhe PROgram System KAPROS [6, 8] is an open modular system on UNIX workstations for a wide range of nuclear reactor calculations. It offers a robust and flexible framework to incorporate new modules, possibly in combination with existing ones, for specific purposes. To implement a new module in KAPROS-E, a specific procedure has to be written in FORTRAN programming language with support for shell-scripts and stand-alone executables. Powerful standardized data exchange between modules is available on several levels. Some of the unique KAPROS-E options are e.g. generation of multi-group cross sections based on self-shielding resonance treatment, up-scattering, group condensation, etc. KAPROS-E is already coupled with stand alone thermal hydraulic codes like SAS4 [9, 10] and RELAP5 [11].

Three specific FORTRAN procedures were developed for the coupling of DANTSYS and COBRA-TF within the KAPROS-E system [12]; the steering procedure COBRAP for organizing the sequence of calculations of 1) macroscopic group constants with the KAPROS-E module KARBUS, of 2) power distributions with the stand-alone code DANTSYS and 3) thermo hydraulic parameters like material temperatures and densities with the stand-alone code COBRA-TF. Additionally two auxiliary KAPROS modules were implemented for the data exchange between the neutron physics and thermal hydraulic part during the coupled calculation:1) the module KCNTTI for creation of input data for COBRA-TF based on KAPROS output files and 2) the module KCTTNI for creation of input data for KAPROS-E based on COBRA-TF output files. The coupled calculations are performed in a loop until a converged solution is achieved tanking into account the neutronic/thermal hydraulic feedbacks like fuel temperature, the moderator temperature, etc.

3. Testing of the developed coupled system

3.1. Problem definition

To assess the prediction capability of the coupled system KAPROS-E/DANTSYS/COBRA-TF, a well defined PWR benchmark problem was selected [13]. In this benchmark, a pin-based burn-up calculation of a fuel assembly with 18X18-24 pins is specified. The PWR fuel assembly consists of 300 fuel pins and 24 water rods (Figure 1). The active fuel rod length amounts 390 cm and the fuel rod pitch is 1.27 cm long. The Zircaloy-4 cladding thickness is 0.064 mm and the outer fuel rod diameter is 9.5 mm. The fuel is UO_2 with a U-235 content of 4 wt %. In the present investigation, a more realistic axial power profiles and hence moderator and fuel temperature distributions are taken into account by the coupled approach. Due to symmetry considerations only one quarter (neutronics) and one eighth (thermal hydraulics) of the fuel assembly is selected as computational domain (figure 1).

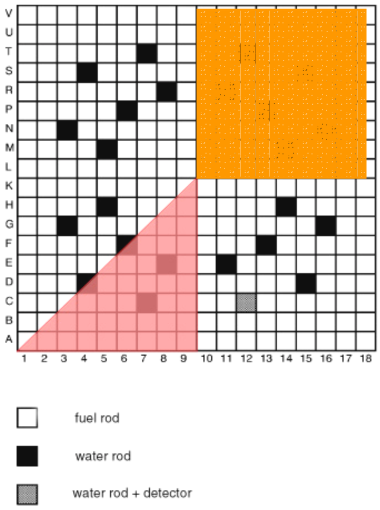


Figure 1 The PWR fuel assembly FA 18x18-24

3.2. Neutron physics and thermal hydraulic model issues

For the 3D multi-group transport solution, the transport code DANTSYS uses the discrete ordinates approximations (S_n) with order n=8 in x-y-z geometry. A neutron flux equals to zero at the extrapolated length is used as z boundary condition while in x-y direction a zero neutron current as boundary condition (reflective b.c.) is assumed. In this way, the model consists of 81 radial zones and 10 axial layers, resulting in a total of 810 material zones, each one characterized by a specific material composition and hence by particular cross section sets based on particular state parameters (fuel temperature and moderator temperature and density) in these zones. These computational domains are mapped to the thermal hydraulic nodalization of the sub-channel model so that the feedbacks can be directly updated for each of the 810 material compositions.

For each material composition one-dimensional standard Wigner-Seitz cell calculations (with reflecting boundary conditions) are performed, using a cross section library with 28 groups. The 28 group library was created from a 69 group master library based on ENDF/B-6.5 with the KAPROS-E procedure COLLIB [9], using the 69 group weighting spectrum of the fuel cell in the FA. The group boundaries and the number of coarse groups for the collapsing were carefully assessed. The selected 28 group boundaries show good agreement with corresponding 69 group master library calculations. In figure 2 the FA-quarter is shown, where each fuel (number 1) and water (number 2) rod is individually represented in the radial plane, resulting in a total of 81 radial computational domains per axial elevation. The 390 cm long FA is axially divided in 10 axial nodes of equal length.

The thermal hydraulic model of COBRA-TF encompasses one eighth of a FA as shown in figure 3. There one can find the numbering of each sub-channel as well as the fuel pin numbering. The cross-flow through the sub-channel boundaries is described by appropriate mixing models. In this problem the inlet mass flow rate for the FA, the coolant temperature at the FA-inlet as well as the pressure at the FA-outlet are important boundary conditions.

1	1	1	1	1	1	1	1	1
1	1	1	1	2	1	1	1	1
1	1	2	1	1	1	1	1	1
1	1	1	1	1	2	1	1	1
1	1	1	1	1	1	1	2	1
1	1	1	2	1	1	1	1	1
1	1	1	1	1	1	2	1	1
1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1

Figure 2 Radial arrangement of the fuel pins and water rods within a quarter of the fuel assembly (1: Pin, 2: water rods)

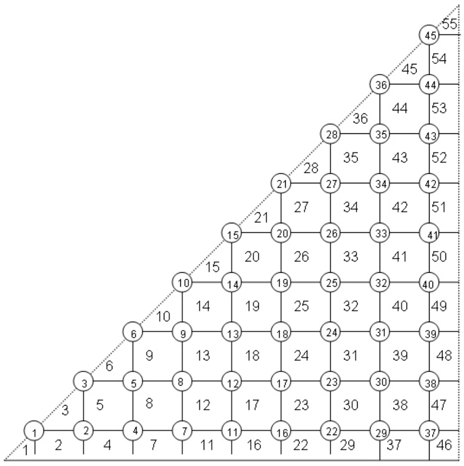


Figure 3 Arrangement of the subchannels and fuel pins in the COBRA-TF model (1/8 of a the FA)

3.3. Selected, preliminary results

The coupled system KAPROS-E/DANTSYS/COBRA-TF was run on a LINUX-Cluster using the models explained above. The coupled simulation starts with an assumed cosine-shaped axial power profile. The simulation was stopped after a number of predefined iteration steps. The feedback from the thermo hydraulic code was directly used in the neutron physics calculation, without any relaxation technique. With this procedure, the iteration behaviour of figure 4 was obtained; during 8 steps the power profile shape showed stabilizing oscillatory behaviour. The final shape of the axial power profile differs significantly from the initially assumed cosine shape. The other parameters like water density or fuel temperature shows similar convergence behaviour as the linear power.

The relative value of the pin power of the FA-quarter is given in figure 5 as predicted by the coupled system. The yellow positions in figure 5 are the water rods. Hence no power is there. It can be seen that around the water filled control rod positions, the power of the fuel rods is higher than elsewhere due to the enhanced neutron moderation. In addition, the axial power profile, fuel, and cladding temperature of each fuel pin as well as the coolant temperature and density are calculated taking into account the feedback effects. By this new approach the prediction capability is significantly extended to a pin-based approach. Consequently the pin behaviour of a FA can be described in a more realistic manner than by conventional methods (no coupling) by the developed coupled system.

4. Summary and outlook

The sub-channel code COBRA-TF was successfully implemented as a module in the neutron physics system KAPROS-E. The coupled system KAPPROS-E/DANTSYS/COBRA-TF was used to predict the pin power of a PWR-fuel assembly by directly taking into account the feedbacks between the neutron physics and thermal hydraulic part. It could be shown that the coupled solution converged after 8 iterations. The detailed radial pin power distribution illustrates the extended capability of coupled code systems to predict the behaviour of FAs in a very detailed spatial resolution. This is an important intermediate step for the overall goal to develop and use pin-power reconstruction methods in advanced transient calculations. Additional work is underway to improve the convergence of the coupled solution and to further validate and qualify the coupled procedure e.g. by Monte Carlo methods in connection with subchannel codes.

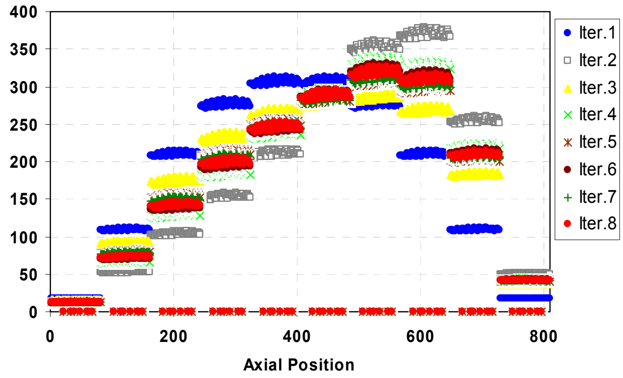
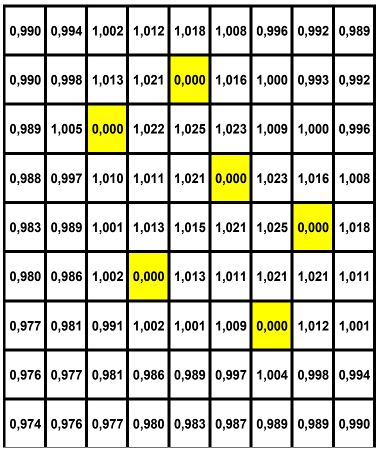


Figure 4 Development of the linear power of all 81 fuel rods along the axial nodes (10 nodes)



0,990	0,994	1,002	1,012	1,018	1,008	0,996	0,992	0,989
0,990	0,998	1,013	1,021	0,000	1,016	1,000	0,993	0,992
0,989	1,005	0,000	1,022	1,025	1,023	1,009	1,000	0,996
0,988	0,997	1,010	1,011	1,021	0,000	1,023	1,016	1,008
0,983	0,989	1,001	1,013	1,015	1,021	1,025	0,000	1,018
0,980	0,986	1,002	0,000	1,013	1,011	1,021	1,021	1,011
0,977	0,981	0,991	1,002	1,001	1,009	0,000	1,012	1,001
0,976	0,977	0,981	0,986	0,989	0,997	1,004	0,998	0,994
0,974	0,976	0,977	0,980	0,983	0,987	0,989	0,989	0,990

Figure 5 Relative radial power within the fuel assembly as predicted by the coupled code

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