Investigation of Modelling Aspects for Coupled Code Applications in Safety Analysis

S. Langenbuch, K.-D.- Schmidt, K. Velkov
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS)mbH
85748 *Garching, Germany*
lab@grs.de ; smk@grs.de ; vek@grs.de

Abstract

Simulation of NPP accident conditions requires 3D modelling of the reactor core to ensure a realistic description of the physical phenomena. Accident conditions that are strongly determined by non-uniform disturbances at reactor core inlet or in the core itself, require at any case the extension of thermal-hydraulic plant system codes by coupling 3D neutronics models. Many activities are performed to develop further and validate such coupled code systems. This paper describes the code capabilities and some specific modelling considerations of the coupled system code ATHLET with the 3D neutronics code QUABOX/CUBBOX, both developed by GRS. Emphasis is given to a generic problem of coupled codes, namely the rules and effect of grouping of fuel assemblies into thermal-hydraulic channels (THC). The influence of different mapping schemes and specific coupled code considerations on the accuracy of the results is discussed on the basis of a set of calculations done for the OECD/CSNI steam line break accident and for reactivity initiating accidents (RIA) due to boron dilution slug propagation in a PWR core.

1 Introduction

Historically, the development of thermal-hydraulic system codes and 3D neutronics codes has gone parallel for a long time. Mainly that was a consequence of limited computer capabilities. Extensive projects have been launched to validate these codes and nowadays when powerful computers are reality the trend is to couple already well verified and validated thermal-hydraulic system codes and 3D core neutronic models. Efforts are made to accelerate the existing code's performance either by application of better and of higher level program languages and compilers that enables performance of parallel process computation or/and optimising the coupling methods and coupling schemes. Nowadays to meet the requirements of best estimate simulations of NPP transients and the requirement of performing realistic accident analysis, the application of coupled codes system is obligatory. Only simulating the transients in a coupled code manner, the analysis of the whole plant behaviour can be improved and can allow to study accurately the reactivity feedback on power generation and the effect on local power density distributions. Transients that at any case require coupled code system calculations are the main steam line break, boron dilution accidents and ATWS cases.

Since several years, GRS is applying its coupled code system ATHLET-QUABOX/CUBBOX for performing accident analysis and has gained experience in optimising the coupling schemes. In the frame of the international MSLB benchmark problem (Ivanov,1997) and GRS studies of the propagation of a boron slug of different magnitude and speed through a PWR core, a lot of sensitivity studies have been perform to optimise the coupling methodology These accident sequences are determined by a very strong coupling between neutronics and thermal fluiddynamics in the reactor core typical for RIA conditions.

2 Coupled code system QUABOX/CUBBOX-ATHLET

The ATHLET-QUABOX/CUBBOX coupling approach has been described by (Langenbuch, 1996a, 1996b). It is based on a general interface, which separates data structures from neutronics and thermo-fluiddynamic code and performs the data exchange between system code ATHLET and neutronic code QUABOX/CUBBOX. In addition, the approach has been successfully applied to couple other 3D neutronics codes as DYN3D, KIKO3D, BIPR and STEPAN to the system code ATHLET. The coupling method is classified as internal and has the following main features: The fluiddynamic equations for the primary circuit and the flow channels in the reactor core region are completely modelled and numerically solved by ATHLET methods. The time-integration in the neutronics code QUABOX/CUBBOX is performed separately. Therefore, both codes keep their capabilities. The time-step size is adaptively synchronised during the transient. The coupling allows a flexible mapping defined by input between fuel assemblies of the core loading and the thermo-fluiddynamic channels. Also the axial meshes for neutronics and fluiddynamics can be defined independently for each one of the codes.

The 3D kinetic core model QUABOX/CUBBOX (Langenbuch, 1977) has the following features: The neutron diffusion equations are solved with two prompt neutron groups and six groups of delayed neutron precursors. The coarse mesh method is based on a polynomial expansion of neutron flux in each energy group. The time-integration is performed by a matrix-splitting method which decomposes the solution into implicit one dimensional steps for each spatial direction. The reactivity feedback is taken into account by dependence of homogenised cross-section on feedback parameters, the functional dependence can be defined in a very general and flexible manner.

The system code ATHLET (Lerchl, 1998) is applied for analysis of the whole plant behaviour under accident conditions. It is a thermal fluiddynamic system code based on 1D pipe components with a wide range of applications comprising anticipated and abnormal plant transients, small and intermediate leaks as well as large breaks in PWRs and BWRs. The code offers the possibility of choosing between different models of fluiddynamics. The two-phase flow can be described with up to a full 6-equation model for mass, energy and momentum of both phases including models for non-condensables. A boron tracking model for single and two phase flow has been implemented. The code structure is highly modular, and allows an easy implementation of different physical models. The basic modules are: Thermo-fluiddynamics, Neutron Kinetics, General Control Simulation Module and Numerical Integration Method FEBE. ATHLET provides a modular network approach for the representation of a thermal-hydraulic system. The possibility to implement connections among group of parallel channels in reactor core allows to model cross-flow in the reactor core.

3 Thermal-hydraulic and kinetic modelling considerations

The safety analysis of accident conditions (especially RIAs) with strong coupling between neutronics and thermofluiddynamics combined with non-uniform disturbances at reactor core inlet or core inside require application of a great number of THC or a correct core mapping schemes. That means that the nodalization schemes of the thermofluid code and the neutronics code must be in such a way overlapped that no important physical or thermalhydraulic information should be lost or distorted in the model. The mapping scheme must be at first adapted to the type of the reactor (PWR or BWR) and second also to the specific features of the studied transient. That is of particular importance when the thermalhydraulic code has a 1D parallel channel model approximations but not 3D thermal hydraulics, which is the case with the GRS coupled code system ATHLET-QUABOX/CUBBOX. The OECD main steam line break benchmark (Ivanov, 1997) has been used as a bases to perform optimisation studies for mapping of core thermal hydraulic channels to different numbers and configurations of fuel assemblies. The results of these sensitivity studies which influence the feedback model are reported in (Langenbuch,1999) for phase 2 of the benchmark and in (Langenbuch, 2000) for phase 3.

Three different mapping schemes have been analysed and compared in the frame of this benchmark. The first nodalization scheme is 1:1 modelling of the core (Fig.1). We called it reference scheme. That means that each fuel assembly corresponds to a single THC. The total number of THC in this case is 178 (177 fuel assemblies channels and one channel for the reflector zone). In each channel the thermo-hydraulic feedback is described by a pipe and a fuel rod module of the ATHLET code. No cross flows are taken into account among the THC. The second mapping scheme is the proposed one in the specification of the benchmark. It corresponds to 19 THC (one of them is a reflector channel) shown in Fig.2. The THC are located in three radial rings of the core. Each ring has 6 THC symmetrically located along the ring. The flow area of each THC in a ring is the same. This mapping scheme is proposed on the basis of the analysis done with the TRAC code. To each THC is attached a heat slab describing the fuel rod with 24 equidistant nodes in axial direction. On the basis of our experience with coupled code calculations we tried to create another optimised mapping scheme. That is the scheme with 15 THC (1 for the reflector zone) presented in Fig.3. The number that is put on each assembly (square) of the mapping schemes shows to which THC is related the fuel assembly. The stuck rod assembly (number one in Fig.3) is assigned to a separate THC. These comparisons prove that deviation from 177 THC scheme in case of the optimised mapping of 14 THC are smaller than in case of 18 THC. As an example of local parameter differences for the three cases is shown in Fig. 4 the time history of the maximum fuel temperature. The effect of correct grouping of THC plays a considerable role in a realistic prediction of the neutronic and thermal hydraulic process in the reactor core described by coupled code systems. The one-to-one scheme has the advantage of precise description of the core behaviour but in many cases it is time expensive to

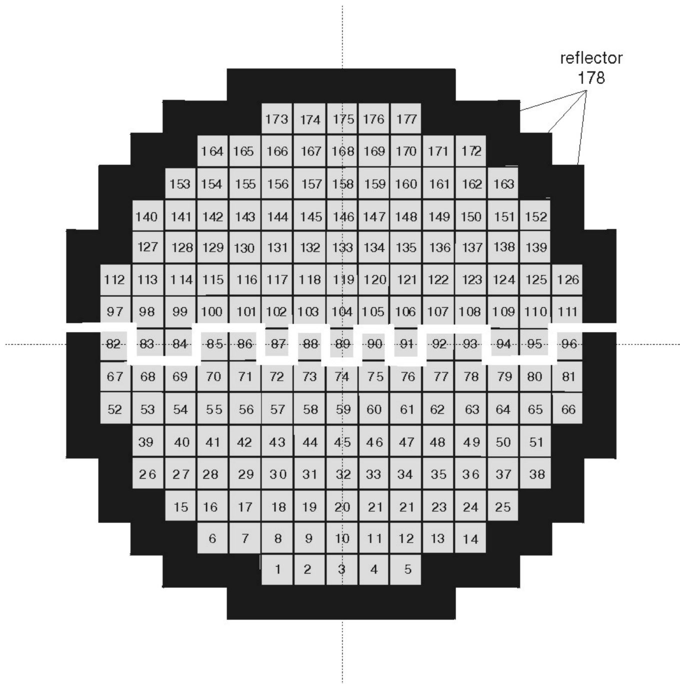

Fig.1 A mapping scheme for 178 THC

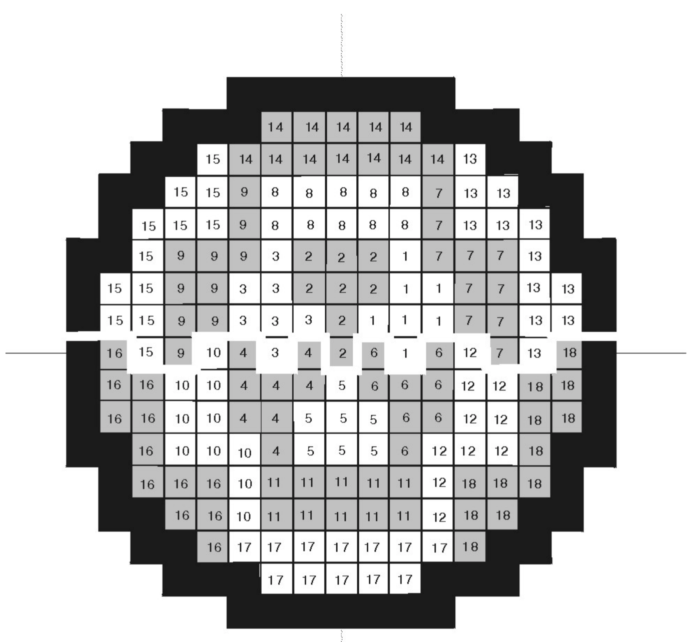

Fig.2 Specified mapping scheme with 18 THC

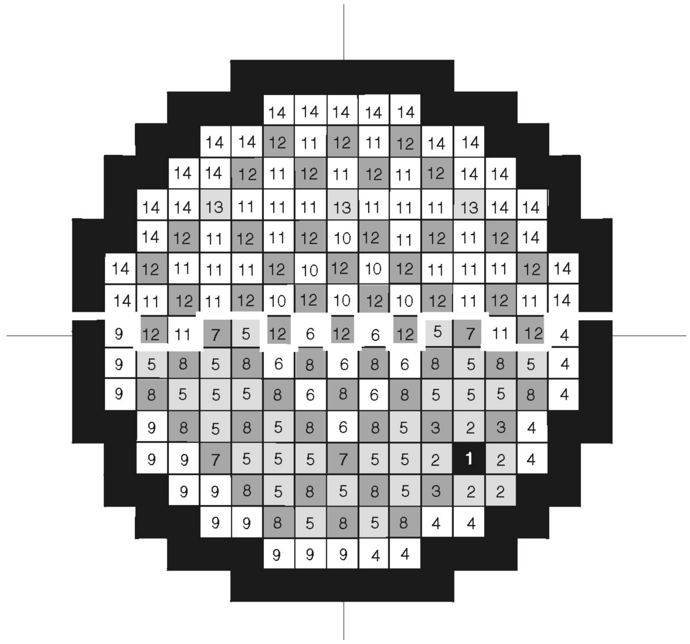

Fig.3 Optimised mapping scheme with 14 THC

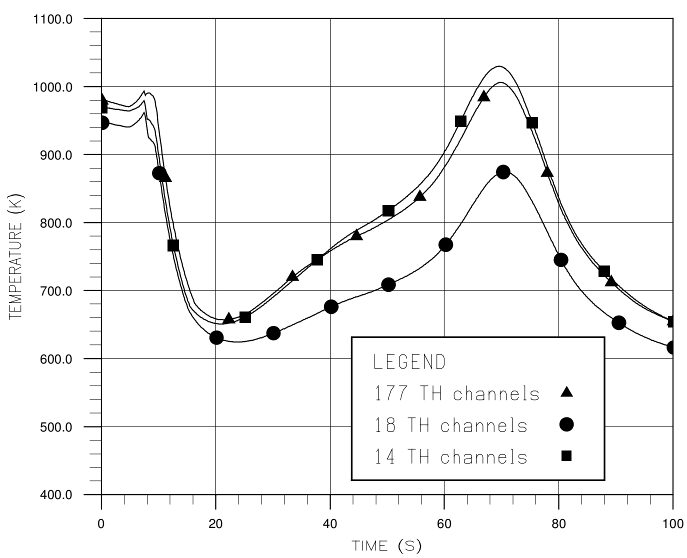

Fig.4 Maximum (local) fuel temperature histories

be realised. On the basis of these studies we applied and checked the following general rules for creating optimised mapping schemes that can achieve high accuracy in predicting local parameters with a minimum number of THC:

- assemblies with control rods should be united in a separate THC,

- each control rod cluster should have its own THC if it has different initial insertion depth in comparison with the other clusters or if at different time moment is expected to start rods movement,

- assemblies are grouped in separate THC if they are located around strong neutron flux disturbances (stuck rod, dropped rod, assembly with very different burnup or very different enrichment),

- non-controllable assemblies are grouped in one THC if they are located at one radial core ring,

- for unsymmetrical disturbances at core inlet around the location of the disturbance each assembly should be assigned a separate THC.

The above listed rules have been checked and applied in GRS studies on boron dilution RIA for PWR. Different mapping schemes have been used by different form of the boron slug at core inlet. The most probable case was considered an unsymmetrical boron diluted slug propagation in 1/4 core at hot zero power conditions. The most effective scheme chosen to perform the analyses is the mapping scheme presented at Fig.5, where 3/4 core is presented as one THC and the rest 1/4 core with 49 THC and 49 fuel rods. In order to depict correctly the neutronic response of the core, of importance is to make correct assumption how the diluted coolant or boron-free water is mixing with the undiluted coolant on the way throughout the cold leg, downcomer and lower plenum of the reactor vessel. The spatial and temporal boron concentration distribution at core inlet can be correctly obtained only with the help of experimental results on test facilities with similar geometry to the studied reactor condition. A number of sensitivity studies have been performed to study the influence of the different assumptions on the mixing phenomena at lower plenum. Different types of front form have been studied - a step change and ramps with different gradients Fig.6. The speed and the volume of the diluted coolant are another important parameters that directly affect core reactivity response. These parameters are a complex function of the loop responses. Reactor states at near to natural circulation conditions are of great interest but at the same time are rather complicated to be estimated because the core comes quickly to voiding after the first power peak and soon will be established a considerable radial pressure profile around the hot spot assembly that can contradict to the limitations of the applied parallel channel model of the core. To avoid that, it is reasonable in the vicinity of the voiding region to assign only one THC that will group several by the pressure affected assemblies, each modelled with a separate fuel rod, or to introduce cross flow junctions among the strongly pressure affected assemblies. Using parallel THC models, it is recommended to study the sensitivity of results on the specific mapping of THCs to determine the effect on safety relevant phase parameters.

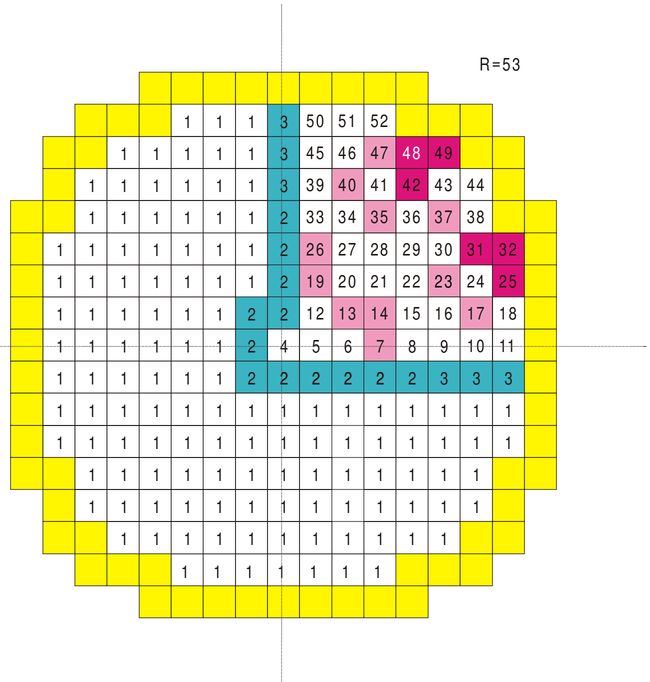

Fig.5 A mapping scheme with 53 THC

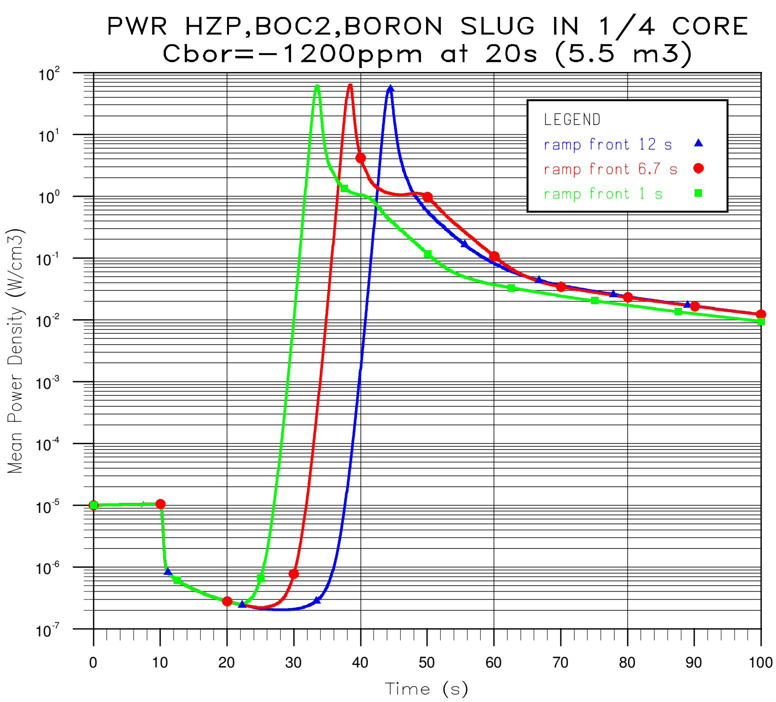

Fig.6 Reactor power for different ramp gradients

4 Conclusions

The safety analysis of accident conditions with strong coupling between neutronics and thermo-fluiddynamics combined with non-uniform disturbances at the reactor core inlet require the application of system codes with coupled 3D neutronics. Only with coupled code calculations the margins of the reactor safety limits can be correctly estimated, because most of them are based on local core parameters. The 3D neutronics model is necessary to describe correctly the reactivity feedback effects and the changes of local power density. Nevertheless, the accuracy in prediction of the local reactor parameters are very strongly affected from reactivity feedback which depends on the modelling of the THC.

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6. References

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