Application of Coupled Codes for Safety Analysis and Licensing Issues

Siegfried Langenbuch*, Kiril Velkov
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH
85748 Garching, Germany

Abstract

An overview is given on the development and the advantages of coupled codes which integrate 3D neutron kinetics into thermal-hydraulic system codes. The work performed within GRS by coupling the thermal-hydraulic system code ATHLET and the 3D neutronics code QUABOX/CUBBOX is described as an example. The application of the coupled codes as best-estimate simulation tools for safety analysis is discussed. Some examples from German licensing practices are given which demonstrate how the improved analytical methods of coupled codes have contributed to solve licensing issues related to optimized and more economical use of fuel.

KEYWORDS: Coupled codes, ATHLET, QUABOX/CUBBOX, licensing issues

1. Introduction

A definite trend is observed in nuclear safety to perform accident analyses of nuclear power plants (NPPs) by best-estimate codes. Coupled codes represent best-estimate codes for the simulation of plant transients and accident conditions by integrating 3D neutronics models. The advantage of coupled codes is most obvious for plant conditions, which are determined by a strong coupling between neutron kinetics of the reactor core and the coolant flow in the primary circuit, as well as for asymmetrical disturbances in the core, which lead to a strongly space dependent power generation. The work performed in GRS by coupling the thermal-hydraulic system code ATHLET [1] and the 3D neutronics code QUABOX/CUBBOX [2] is presented in detail as an example of similar work for other thermal-hydraulic system codes. The status of coupled codes and their field of applications are described. The validation of these coupled codes has been performed within a strong international effort in the frame of the OECD/NEA LWR plant transient benchmark activities. Some examples from German licensing practices are reported which demonstrate that these improved calculation methods have already contributed to solve licensing issues related to optimized and more economical use of fuel.

2. Disadvantages of Separate Analyses of Reactor Core and Plant Behaviour

In the past the analyses of reactor core behaviour and the analyses of plant transients have been performed separately in each field.

* Corresponding author, Tel. +498932004424, Fax.+498932004599, E-mail: Siegfried.Langenbuch@grs.de

The detailed analysis of reactor core behaviour is based on 3D neutronics models which solve the neutron diffusion equations with two-energy groups including reactivity feedback effects caused by changes of coolant flow conditions or changes of fuel rod temperatures. Nowadays, the efficient solution of spatial and time-dependent neutronics equations is based on coarse mesh methods with local polynomial flux expansion or nodal functional expansion methods. These methods achieve high accuracy even for nodalizations using a single node per fuel assembly in the radial plane. The pin-wise power can be determined by reconstruction methods. The main results are the k_{eff} value describing the criticality condition and the spatial neutron flux distributions, which determine the spatial power density per fuel assembly or per fuel rod.

For reactor core calculations of specific accident conditions the time-dependent boundary conditions have to be defined, e.g. the time-functions of mass flow and coolant temperature distributions at the core inlet together with the pressure. In reality, these time-dependent boundary conditions will be affected by the power generation in the reactor core. This dependence limits the application to fast transients like a control rod ejection or to transients with a weak coupling between fluid dynamics in the primary circuit and the power generation in the reactor core.

The analysis of plant transients and accident conditions is based on system codes which solve the single-phase and two-phase fluid dynamic equations for a network of pipes including models describing components like main coolant pumps, pressurizer and steam generators and models describing the control systems and the protection system. GRS has developed its own thermal-hydraulic system code ATHLET which is comparable to other international codes like CATHARE, RELAP and TRACE. Usually these system codes describe the nuclear power generation by a point kinetics model or by a 1D neutronics model. Therefore, the application of the system codes needs attention to determine adequately the reactivity feedback functions for the point kinetics model.

At this stage of modelling separately the reactor core and the whole plant behaviour, the simulation tools are limited by defining properly the interface conditions. Another limitation is given by the need to describe flow mixing before the core inlet. An approach that takes into account all uncertainties for the reactor core behaviour by defining conservative boundary conditions leads to very unrealistic accident conditions. This contradicts the actual trend to perform realistic best estimate analyses of the reactor core and plant behaviour. The problems of separated analyses can only be avoided by directly coupling 3D neutronics models with the thermal-hydraulic system code.

3. Coupled Code Systems and Their Field of Application

In the international framework great efforts have been performed to develop coupled codes in which thermal-hydraulic system codes for plant simulation have fully integrated 3D neutron kinetics for modelling the reactor core behaviour. As an example, GRS has coupled its thermal-hydraulic system code ATHLET and its 3D reactor core model QUABOX/CUBBOX through a universal interface into a coupled code system. The coupling approach implemented in ATHLET [3] allows various options, which are also applied in other coupled code systems.

- Internal coupling

Coupling of the 3D neutronics model to the system code ATHLET, which models completely the thermal-fluid dynamics of the primary circuit in the core region.

- External coupling

Coupling of the 3D neutronics model including the fuel rod model and the fluid-dynamic model of the core region to the system code, which models only the thermal-fluid dynamics in the primary circuit excluding the core region.

- Parallel coupling

The 3D neutronics model including the fuel rod model and the fluid-dynamic model represents the reactor core. The system code models the thermal-fluid-dynamics in the primary circuit and the core region in a simplified manner. The calculated boundary conditions of the system code are transferred as time-dependent boundary conditions of the more detailed core calculation performed in parallel.

These coupling approaches maintain the capabilities of the separated codes and allow their further independent development. The interface provides the necessary exchange of main physical parameters:

- The power density distribution which is the result of the neutronics calculation and which must be transferred to the fluid-dynamics.

- The distribution of fuel temperature, coolant density and coolant temperature as well as the boron concentration, which are the result of the fluid-dynamic model including the boron transport model and which must be transferred to the neutronics as feedback parameters.

The coupled codes have been extensively validated within international co-operations by solving the series of OECD/NEA benchmark problems for LWR plant transients. The series consists of following benchmark problems: the PWR main steam line break (MSLB) [4], the BWR turbine trip (TT) [5] and the VVER-1000 coolant transient [6]. A summary of results and experiences from these investigations are given in [7].

As realistic best-estimate calculations should be supplemented by an uncertainty and sensitivity analysis, a study for plant transients was performed applying the GRS methodology based on the SUSA package [8].

Generally, the field of applications of coupled codes in safety analysis is the following:

- The cool-down transients with strongly negative moderator temperature reactivity coefficient (MTC) in PWR. The occurrence of a recriticality during cool-down and its consequences have to be analyzed. Such high values of MTC are obtained for increased high burnup fuel or for extended use of MOX fuel.

- The local boron dilution accident in PWR, which was identified as a potential reactivity initiated accident even in shutdown conditions when all control rods are inserted.

- The results of ATWS analyses are strongly affected by feedback reactivity coefficients. The uncertainties of inherent feedback determining power production and consequently pressure increase can be strongly reduced by applying 3D neutronics models. The spatial effects are emphasized if partial failure of control rod insertion is postulated.

- Power upgrading programs generate the demand for reducing uncertainties.

- The BWR instability in plant conditions beyond the stability threshold.

These cases are strongly determined by reactivity feedback effects or by asymmetric core conditions which need an accurate modelling of spatial reactivity effects.

4. Present Issues of Licensing and the Role of Coupled Codes

4.1 Licensing issues for optimized fuel designs

The operation of NPPs is accompanied by a continuous fuel optimization with the aim to reach higher burn-ups using the fuel more economically. This also leads to higher enrichment of Uranium fuel. In addition, MOX fuel is loaded into LWRs which leads to mixed core loadings with a greater heterogeneity of the reactor core configurations.

A general licensing approach has been established to take into account the corresponding changes of nuclear design characteristics. A framework of safety relevant design parameters is defined, which consists of parameters like the power peaking factors describing the power density distributions, the reactivity coefficients and the kinetic parameters. The acceptable range of values for each parameter for a specific NPP design is determined on the basis of extensive calculations for plant transients and design basis accident conditions for representative core loadings. For each core reload it is approved that the nuclear design characteristic of the core is consistent with the acceptable range of parameter values. Generally, no additional plant transient calculations are required for the safety review of a regular reload. If the parameters for new fuel assembly designs or optimized core loadings like low-leakage concepts exceed the justified ranges, the range of parameter values may be adjusted by additional safety evaluations. This is a practical and efficient approach performing the safety review of core reloads.

Nevertheless, the application of cycle independent transient and accident analyses may be too restrictive. One of the design basis accidents for PWR cores which may define restrictions for the core, is the steam line break accident. It has to be checked whether the cool down leads to a recriticality after the reactor trip, and if yes, it should be confirmed that the safety criteria are kept, i.e. that no fuel melting and no critical heat flux occurs. For this accident condition it was decided for some German plants to perform cycle specific calculations for safety justification. The extension of the safety review in this way is only possible because calculations by coupled codes taking into account the actual core loading can be performed very efficiently.

4.2 Experimental verification of design methods

The application of 3D reactor core calculations is accompanied by an extensive surveillance programme for operational power density distributions in German NPPs. Measured power density distributions from operational conditions are compared regularly with pre-calculations performed for the nuclear core design [9].

4.3 Safety evaluation for ATWS cases

Another example of cycle specific accident calculations is related to the analysis of anticipated transients without scram (ATWS) conditions. For PWRs the most important reactivity feedback mechanism for transients without reactor scram is the coolant density feedback. Consequently, the void reactivity curve describing the coolant density feedback is part of the framework of safety relevant parameters. An acceptable void reactivity curve is defined for each plant on the basis of plant transient calculations considering the plant specific system design and control and protection features. Usually, for each reload it is approved that the acceptable void reactivity curve is bounding the cycle specific one. In Germany a discussion started on the question whether the trip of main coolant pumps initiated by operational signals not classified as protection signals should be acceptable as countermeasure in ATWS conditions. The respective void reactivity curves for conditions with and without trip of main coolant pumps have been determined and have been compared with the cycle specific void reactivity curve. The plant transient calculations for ATWS cases were mainly performed applying point kinetics models in the thermal-hydraulic system code, because the reactivity feedback coefficients can be easily varied in such calculations. To determine the realistic plant behaviour for a specific core loading also coupled code calculations using ATHLET-QUABOX/CUBBOX have been performed. These results were taken as confirmation of the more extensive point kinetics calculations. Practically, it would be possible to perform such coupled code calculations as part of the safety review of each core reload.

The experience confirms that the capability of coupled codes already contributes to the safety evaluation. Presently, in cases where conservative evaluation methods define too strong restrictions the application of coupled codes could maintain the necessary safety margins. The application of coupled codes for these cases were supported by the experiences gained in the benchmark problems. It is expected that coupled codes will be more widely used in licensing reviews.

5. Summary

The paper describes the status of coupled codes and their application for safety justifications. The development of these coupled codes was an important step for improving simulation methods. Their validation was performed in close international co-operations. On hand of representative examples it is shown how the improved analytical tools contribute to solve licensing issues for advanced fuel and core loading concepts and actual safety questions.

References

1. G. Lerchl, H. Austregesilo, ATHLET Mod2 Cycle A, User’s Manual, GRS, 2003.
2. S. Langenbuch, W. Maurer, W. Werner, Coarse Mesh Flux Expansion Method for the Analysis of Space-Time Effects in Large LWR Cores, Nuclear Science and Engineering, 63, 437-456, 1977.
3. S. Langenbuch, H. Austregesilo, P. Fomitchenko, U. Rohde, K. Velkov, Interface Requirements to Couple Thermal-Hydraulic Codes to 3D Neutronic Codes. OECD/CSNI Workshop on Transient Thermal-Hydraulic and Neutronic Codes Requirements, Annapolis, Md., U.S.A., November 5-8, 1996.
4. K. Ivanov, T. Beam, A. Baratta, A. Irani, N. Trikouros, PWR MSLB Benchmark, Vol.1: Final Specifications, NEA/NSC/DOC(99)8
5. J. Solis, K. Ivanov, B. Sarikaya, A. Olsen, K. Hunt, BWR TT Benchmark, Vol. 1: Final Specifications, NEA/NSC/DOC(2001)1.
6. B. Ivanov, K. Ivanov, P. Groudev, M. Pavlova, V. Hadjiev, VVER-1000 Coolant Transient Benchmark (V1000-CT), Phase-1 Final Specification, NEA/NSC/DOC(2002)6.
7. K. Ivanov, E. Sartori, E. Royer, S. Langenbuch, K. Velkov, Validation of Coupled Thermal-Hydraulic and Neutronics Codes in International Co-operation,

EUROSAFE 2005, Brüssel, November 7-8, 2005.
8. S. Langenbuch et al., Comprehensive uncertainty and sensitivity analysis for coupled code calculations of VVER plant transients, Nuclear Engineering and Design 235 (2005) 521-540.
9. H.-D. Berger, I. Endrizzi, Verification of PWR Design Methods with Reactor Physics Measurements, KTG-Technical Meeting: Status of Development of LWR Fuel and Design Methods, Rossendorf, March 2-3, 2006