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PWR core safety analysis with 3-dimensional methods

A. Gensler *, K. Kühnel, S. Kuch

AREVA GmbH, Postbox 1109, 91001 Erlangen, Germany

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ABSTRACT

The main focus of safety analysis is to demonstrate the required safety level of the reactor core. Because of the demanding requirements, the quality of the safety analysis strongly affects the confidence in the operational safety of a reactor. To ensure the highest quality, it is essential that the methodology consists of appropriate analysis tools, an extensive validation base, and last but not least highly educated engineers applying the methodology.

The sophisticated 3-dimensional core models applied by AREVA ensure that all physical effects relevant for safety are treated and the results are reliable and conservative. Presently AREVA employs SCIENCE, CASMO/NEMO and CASCADE-3D for pressurized water reactors. These codes are currently being consolidated into the next generation 3D code system ARCADIA®. AREVA continuously extends the validation base, including measurement campaigns in test facilities and comparisons of the predictions of steady state and transient measured data gathered from plants during many years of operation. Thus, the core models provide reliable and comprehensive results for a wide range of applications. For the application of these powerful tools, AREVA is taking benefit of its interdisciplinary know-how and international teamwork. Experienced engineers of different technical backgrounds are working together to ensure an appropriate interpretation of the calculation results, uncertainty analysis, along with continuously maintaining and enhancing the quality of the analysis methodologies.

In this paper, an overview of AREVA's broad application experience as well as the broad validation base of its code systems is given. The importance and necessity of the comprehensive 3-dimensional methodology is illustrated by example analyses of a rod ejection accident and an 'inadvertent opening of the pressurizer safety valve' transient. The examples refer to the safety criteria pellet averaged fuel enthalpy, fast fuel enthalpy rise, number of fuel assemblies in film boiling and the maximum fuel temperature and illustrate, how 3D-methods provide evidence of bigger safety margins or archive more reliable results.

1. Introduction

The methodologies of the safety analysis developed with the advancement of computing power. In the past, the different analysis areas were considered uncoupled with one way and manual data transfer between the codes: plant dynamics with point kinetics, 2D core kinetics, thermal–hydraulics with a single hot sub-channel and fuel rod model with a single hot fuel rod. Transient processes were approximated with steady state calculations. The potential errors of these simple models restrict an economical fuel management analysis as well as the quality of safety analysis. Upgraded computers and advanced numerical algorithms allow consideration of transients, 3D full core calculations and coupling of the plant dynamics with core kinetics, thermal–hydraulics as well as a fuel rod model. Even full core, 3D, pin-by-pin representations are available. The coupling enables an automatic data transfer between the codes in both ways, i.e. the results of one code are input of the other and also vice versa. Simultaneously with these sophisticated codes, advanced safety analysis methodologies are developed and penalties can be reduced to reveal higher safety margins. However, the usage of such complex codes and methodologies requires users skilled and experienced in all subjects or – at least – the users must be supported by experts on all relevant fields. In close cooperation with these experts, the correct and conservative input can be composed and the results can be interpreted correctly.

2. Codes

Several pressurized water reactor (PWR) codes are available at AREVA for PWR safety analysis. Presently AREVA employs SCIENCE (Girieud, 1994) CASMO/NEMO (Hobson et al., 1993) and CASCADE-3D (Boer et al., 1999) for PWR. These codes are currently being

* Corresponding author. Tel.: +49 (0) 9131 90095735; fax: +49 (0) 9131 90094082.

E-mail address: Albin.Gensler@areva.com (A. Gensler).

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consolidated into the next generation 3D code system ARCADIA®. The examples presented in this paper to illustrate the behavior of coupled 3-D analyses use the following codes of the CASCADE-3D code system:

– Thermal–hydraulics sub-channel model COBRA-FLX™ (Wende et al., 2005): COBRA-FLX allows 3-dimensional steady-state and transient full-core as well as sub-channel analyses to be performed. Geometry specification for the reactor core is flexible, relying on information about flow areas and lateral interconnections, i.e. no particular coordinate system is predefined. The capability to calculate cross flow effects is essential for application to mixed core situations with different fuel assemblies co-resident in a core. Conditions in and around a hot pin can be assessed, in combination with the general core performance, by varying the lateral calculation mesh refinement from sub-channel over single-assembly to lumped-assembly configurations. The basic conservation equations of the two-phase flow are written for the mixture quantities in time dependent, one-dimensional form. Cross flow effects are taken into account by additional terms included in these basic equations. The mathematical model neglects sonic velocity propagation. The code solves the flow field equations as boundary value problems. For the energy and continuity equations, the enthalpy and mass flow at the core inlet serve as boundary conditions. Also the axial momentum equation is treated as a boundary value problem, using the core exit pressure. Both temporal and spatial acceleration are accounted for in the cross flow momentum equation. Separated slip flow is assumed in each sub-channel, and the void fraction distribution is evaluated as a function of enthalpy, flow rate, heat flux and pressure. Several correlations for the sub-cooled void fraction are present in the code. This permits the effect of voidage on flow resistance to be accounted for and void fraction limits to be assessed.
– COBRA-FLX™ with integrated fuel rod model: fuel pin temperatures are calculated from the radial heat conduction equation. The gap between pellet and cladding is characterized by a gap heat transfer coefficient. This coefficient can be determined based on different models, taking into account conductive, radiation, fuel-clad contact, and contact pressure contributions. The coupling to the coolant determining heat transfer dynamics is realized by appropriate correlations, where optionally for penalization a user-defined multiplier to the clad-to-coolant heat transfer coefficient can be applied. Optionally, fuel enthalpies and fuel shell temperatures of a hollow pellet can be calculated.
– PANBOX (Boer et al., 1992) which is a kinetics core simulator with integrated COBRA-FLX™: PANBOX is designed to calculate steady-state and transient reactor core conditions in 3-dimensional geometry. Interface files from in-core fuel management calculations serve as basic neutron kinetics input data. These files contain node dependent neutron kinetics data such as cross sections, derivatives, geometric data of nodes, burnups and burnup dependent heterogeneous power form function tables. PANBOX solves the steady-state, time- and space-dependent neutron diffusion equations for an arbitrary number of neutron energy groups. A flux reconstruction method is used to determine local flux and power values inside the fuel assembly. Burnup-dependent heterogeneous power form functions can be applied to take the heterogeneous structure within a fuel assembly into account. As the finest solution, the pin-wise power distribution in several axial layers is calculated. The module COBRA-FLX is used twofold, for coupling of kinetics and thermal–hydraulics by means of a 1-D fuel assembly modelling and for a detailed hot sub-channel analysis in a subsequent step without feedback to the kinetics. Kinetic and thermal–hydraulics modules are coupled in an iterative manner to consider the respective feedback mechanisms: kinetic power distribution data are passed to the thermal–hydraulic module which performs at each time step a thermal–hydraulic analysis. The changes of calculated thermal–hydraulic quantities (moderator temperature and density, fuel temperature) are evaluated and fed back into kinetics via cross section updating. Accordingly PANBOX calculates the radial and axial redistribution of the power density during the transient.
– Core kinetics and plant dynamics coupled code R/P/C (RELAP/PANBOX/COBRA) (Jackson et al., 1999a,b): within R/P/C the point kinetic solver in RELAP is replaced by the 3D core simulator PANBOX. An example for the data exchange within R/P/C is shown in Fig. 1.

Depending on the transient and it's complexness an adequate code is used for the analysis.

The core simulator PANBOX will be replaced by the newly developed code ARTEMIS™ (Porsch et al., 2009) with extended features like a more sophisticated fuel rod model and a pin-by-pin and sub-channel-by-sub-channel representation of the thermal–hydraulics. ARTEMIS™ can also be coupled with the plant dynamic code RELAP.

3. Validation base

Since integral tests of the coupled code system cannot cover the total application range of the individual models, detailed validation of the individual models and empirical correlations are required. The integral tests demonstrate the appropriate coupling functionality and overall validity, which shows the essential model setup properties (e.g. nodalization). The validation cases were performed either for testing of new codes and methods or for an enlargement of the validation basis. Table 1 provides an overview of the broad validation base of the coupled PWR 3D code system R/P/C and its integrated codes PANBOX and COBRA-FLX™. If no reference is given in the table, at least an AREVA internal validation is available.

4. Application experience for licensing with coupled 3D-codes

AREVA uses coupled 3D-methods for a wide range of licensing purposes like new-builds, power upgrades, enrichment increase, introduction of MOX fuel and decennial safety revisions. They are applied for different PWR types (Framatome, Siemens, Westinghouse, and AREVA) with various types of fuel assemblies (14×14, 15×15, 16×16, 17×17, and 18×18) and a large range of rated power from 360 to 1600 MW_e.

Coupled 3D-methods are the most suitable tools for transients with asymmetric power distributions or strong axial and radial power redistribution. Table 2 shows AREVA's wide experience of licensing analyses performed with 3D-methods. Overall, the coupled 3D-methods are accepted for licensing in several countries: Belgium, Brazil, Finland, Germany, Netherlands, Spain, and Switzerland.

5. Examples of coupled 3D effects

5.1. Description of a selected case of a rod ejection analysis

The following example refers to a generic rod ejection accident (REA) analysis of a Siemens built PWR with 3950 MW thermal power and 193 fuel assemblies of the type 18×18–24. A rod ejection accident is simulated by means of PANBOX at the end of a representative cycle with an initial power of 30% P/P_N (P/P_N = -rated nominal power). At this power level, 16 of the 61 control rods are fully inserted according to the control bank insertion limit. An