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Testing the NURESIM platform on a PWR main steam line break benchmark

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ABSTRACT

Within the NURESAFE project, a main steam line break benchmark has been defined and solved by codes integrated into the European code platform NURESIM. The paper describes the results of the calculations for this benchmark. Six different solutions using different codes and code systems are provided for the comparison. The quantitative differences in the results are dominated by the differences in the secondary system parameters during the depressurization. The source of these differences comes mainly from the application of different models for the two-phase leak flow available in the system codes. The use of two different thermal hydraulic system codes influences the results more than expected when the benchmark was created. The codes integrated into the NURESIM platform showed their applicability to a challenging transient like a main steam line break.

1. Introduction

The NURESAFE Collaborative Project aimed at making a new and significant step towards the establishing of a European Reference Simulation Platform. This platform with the name NURESIM has been developed over the last years and provides the following main simulation and related possibilities (Chauliac et al., 2011):

- Accurate representation of physical phenomena in reactor and core physics, two-phase flow thermal hydraulics and fuel modeling
- Multi-scale and multi-physics capabilities of reactor safety computations
- Easy coupling of different codes and solvers by using common data structures and generic functions
- Generic pre-processing and post-processing and supervision functions
- Deterministic and statistical sensitivity and uncertainty analyses

The NURESIM-platform is based on the open software simulation platform SALOME (http://salome-platform.org). The actual list of implemented codes with details of implementation is provided in (Chanaron et al., 2015). Examples of application of the NURESIM platform to different kind of reactor safety problems can be found, e.g., in (Jimenez et al., 2015; Bestion, 2017; Grahn et al., 2017).

In continuation of the application of the NURESIM, platform situation targets for different reactor types have been defined. The situation target for the pressurized and the VVER reactors is based on main steam line break (MSLB) benchmarks while an ATWS transient has been selected for the boiling water reactor. The current paper deals with the results of the main steam line benchmark for a pressurized water reactor.

The coupling of 3D neutron kinetic core models to advanced thermal hydraulic system codes started more than twenty years ago. The very first options of coupling used data exchange between codes by means of files (e.g. Feltus, 1994). Later on, the direct integration of the 3D neutron kinetic core models into the system codes and the corresponding creation of one module for the execution of calculations became a typical type of coupling (e.g. Grundmann et al., 1995). In order to reduce the coupling efforts, system codes were provided with well-defined interfaces for the incorporation of 3D neutron kinetic core models. The system code

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ATHLET (Austregesilo, 2012) is one example, which has been coupled to the neutron kinetic core models KIKO3D (Kereszturi et al., 2003), BIPR8 (Lizorkin et al., 2015) and DYN3D (Rohde et al., 2016) using an identical interface. All these types of coupling (including the NURESIM approach) use the so-called operator splitting approach. Recently, Jacobian-Free Newton Krylov methods started to be developed in order to avoid restrictions imposed by the operator splitting (e.g. Gaston et al., 2009). Such developments are mainly driven by the analysis of Generation-IV reactors. A review of both approaches can be found in (Mylonakis et al., 2014).

The need of more detailed analysis of postulated main steam line break accidents was one of the major drivers for the development of coupling of 3D neutron kinetic core models and thermal hydraulic system codes. The hot zero power MSLB analyses using the STAR and the RETRAN codes reported in (Feltus, 1994) were one of the earliest works in that area. The first international benchmarks for coupled code systems were based on postulated MSLB accidents, the TMI-1 benchmark (Ivanov et al., 2003) for a reactor core with rectangular fuel assembly cross section and the VVER-440 benchmark (Kliem et al., 2007) for hexagonal fuel assemblies.

The current MSLB benchmark was defined and calculated within the NURESAFE project with the intention to test the multi-physics features of the NURESIM platform with respect to 3D neutron kinetics of the whole core and system thermal hydraulics and to serve as a basis for the verification and use of the advanced NURESIM features such as embedded hybrid coupling with different resolution in the core, the direct coupling with computational fluid dynamics codes in the pressure vessel and in the core (Grahn et al., 2017) or the envisaged coupling with fuel performance codes.

In chapter 2, the description of the benchmark is shortly outlined. Chapter 3 gives an overview on the used codes for the solution of the different parts of the benchmark. In chapter 4 the models developed for the solution of the benchmark are described and chapter 5 contains the comparison between the different platform codes.

2. Benchmark specification

A benchmark for the evaluation of the possibilities of codes integrated into the NURESIM platform was defined. It concerns a main steam line break (MSLB) in a pressurized water reactor. The benchmark specification is given below.

The reference reactor is the 4-loop Westinghouse PWR ZION. The reactor core configuration includes fuel assemblies (FAs) of four different types (MOX and UOX FAs, each with two different initial enrichments). Their geometries and initial compositions given in (Kozlowski and Downar, 2003) have been used in preparation of the cross-section library for benchmark simulations (see Section 4.1 of Chapter 4). To maximize the reactivity effect of the core overcooling due to the MSLB, the transient is assumed to start at the end of cycle (EOC) when the boron content of coolant is negligible. One cycle calculation has been performed under nominal conditions with CRONOS2 (see Section 3.2.1) to compute the EOC state (radial and axial burnup distributions) (Bernard et al., 2014). The resulting scheme of reactor core model configuration and control rod locations is given in Fig. 2.1 (radial map and axial burnup profile).

The initial steady-state corresponds to the hot zero power (HZP) conditions at the end of a fuel cycle (EOC) summarized in Table 2.1.

A double-ended MSLB at the outlet nozzle of the steam generator (SG) in the first reactor loop is assumed to be the initial event of the accident. The break size is the actual size of the main steam line (MSL) with the diameter of 0.89 m. The flow limiter with the cross

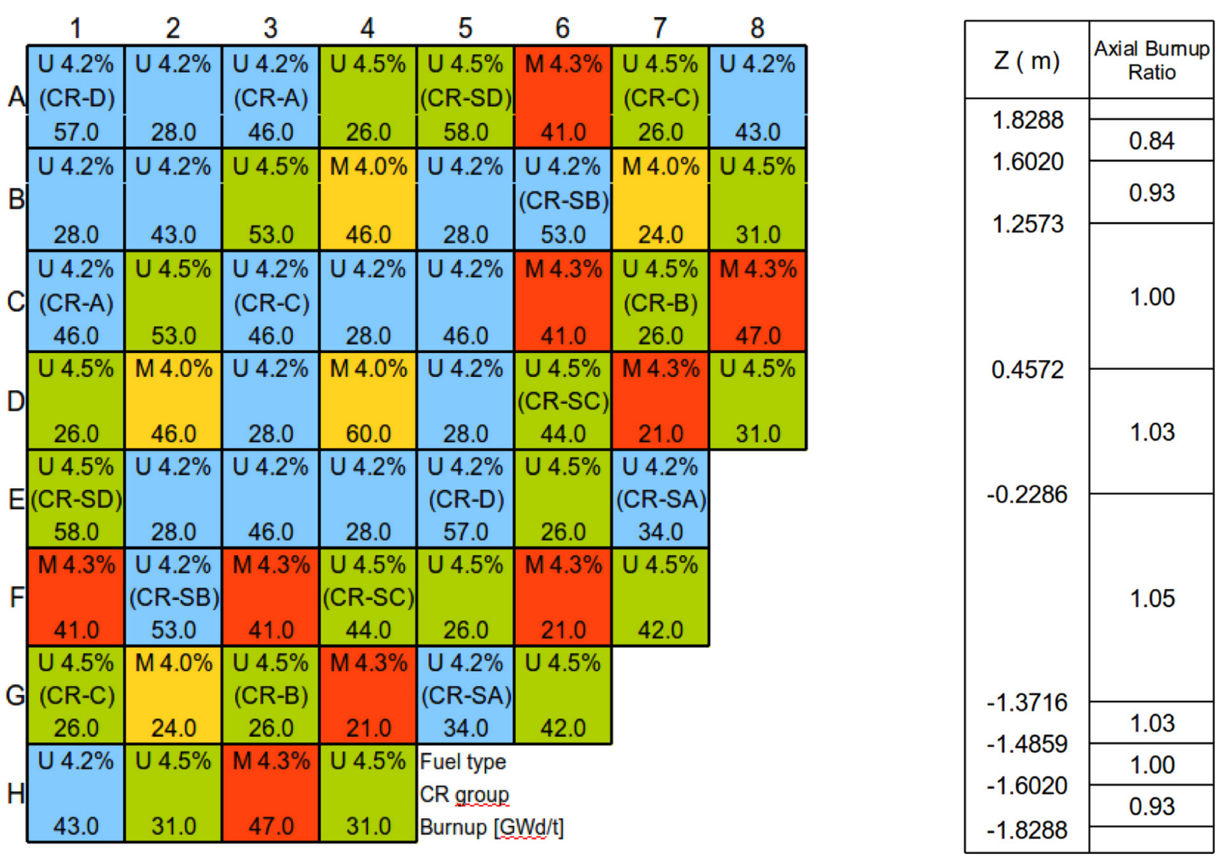

Fig. 2.1. Reactor ¼ core map (fuel assembly type, initial enrichment, mean burnup at EOC, and location of control rods) and axial burnup profile.

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