Nuclear Engineering and Design 241 (2011) 3697–3706

Contents lists available at ScienceDirect

Nuclear Engineering and Design

journal homepage: www.elsevier.com/locate/nucengdes

Uncertainty and sensitivity analyses as a validation tool for BWR bundle thermal–hydraulic predictions

Augusto Hernandez-Solis[a,b,*], Christian Ekberg[b], Christophe Demaziere[a], Arvid Ödegård Jensen[b], Ulf Bredolt[c]

[a] Department of Nuclear Engineering, Chalmers University of Technology, SE-412 96 Göteborg, Sweden
[b] Department of Chemical and Biological Engineering, Chalmers University of Technology, SE-412 96 Göteborg, Sweden
[c] Westinghouse Electric Sweden AB, Nuclear Services, SE-721 63 Västerås, Sweden

ARTICLE INFO

Article history:
Received 7 May 2010
Received in revised form 14 June 2011
Accepted 27 June 2011

ABSTRACT

In recent years, more realistic safety analyses of nuclear reactors have been based on best estimate (BE) computer codes. The need to validate and refine BE codes that are used in the predictions of relevant reactor safety parameters, led to the organization of international benchmarks based on high quality experimental data. The OECD/NRC BWR full-size fine-mesh bundle test (BFBT) benchmark offers a good opportunity to assess the accuracy of thermal–hydraulic codes in predicting, among other parameters, single and two phase bundle pressure drop, cross-sectional averaged void fraction distributions and critical powers under a wide range of system conditions. The BFBT is based on a multi-rod assembly integral test facility which is able to simulate the high pressure, high temperature fluid conditions found in BWRs through electrically heated rod bundles. Since code accuracy is unavoidably affected by models and experimental uncertainties, an uncertainty analysis is fundamental in order to have a complete validation study. In this paper, statistical uncertainty and sensitivity analyses are used to validate the thermal–hydraulic features of the POLCA-T code, based on a one dimensional model of the following macroscopic BFBT exercises: (1) single and two phase bundle pressure drop, (2) steady-state cross-sectional averaged void fraction, (3) transient cross-sectional averaged void fraction and (4) steady-state critical power tests. The Latin hypercube sampling (LHS) strategy was chosen since it densely stratifies across the range of each uncertain input probability distribution, allowing a much better coverage of the input uncertainties than simple random sampling (SRS). The results show that POLCA-T predictions on pressure drop and void fractions under a wide range of conditions are within the validation limits imposed by the uncertainty analysis, while the accuracy of critical power predictions depends much on the boundary and input conditions.

1. Introduction

At the end of the year 2009, nuclear energy provided about 15% of the world's electricity as a continuous and reliable based-load power. Nowadays, nuclear energy is experiencing a renaissance because it represents a very good option to fulfill the growing demand for electricity around the globe. Concerns over climate change and dependence on overseas supplies of fossil fuels are the main reasons for such a renaissance. Therefore, with an increased number of light water reactors (LWRs) in the world, there is a huge interest on improving deterministic safety analysis as an essential tool for demonstrating the safety of nuclear power plants.

BE computer codes constitute one of the actual acceptable options for demonstrating that safety is ensured with an adequate margin. Nowadays, they are employed to calculate postulated accidents and transient scenarios in a realistic way, replacing the conservative approach used in the past in computational reactor safety analysis. However, code predictions are uncertain due to several sources of uncertainty, like code models as well as uncertainties of plant and fuel parameters. Therefore, BE calculations should be supplemented by a quantitative uncertainty analysis. On the other hand, the study of how output uncertainty can be apportioned to the different input sources, known as sensitivity analysis, is an important complement to uncertainty quantification since it identifies the most dominant model parameters.

* Corresponding author at: Department of Nuclear Engineering, Chalmers University of Technology, SE-412 96 Goteborg, Sweden. Tel.: +46 317723089; fax: +46 317723079.
E-mail address: augusto@nephy.chalmers.se (A. Hernandez-Solis).

0029-5493/$ – see front matter © 2011 Elsevier B.V. All rights reserved.
doi:10.1016/j.nucengdes.2011.06.034

The need to validate and refine BE codes that are used in the predictions of relevant reactor safety parameters, led to the organization of international benchmarks based on high quality experimental data. The OECD/NRC BFBT benchmark was established in 2002 based on available data from the Nuclear Power Engineering Corporation (NUPEC), and offers a good opportunity to assess the accuracy of thermal–hydraulic codes in predicting, among other parameters, single and two phase bundle pressure drop, cross-sectional averaged void fraction distributions and critical powers under a wide range of system conditions. With respect to the void distribution inside a fuel assembly, which has been regarded as an important factor in the determination of boiling transition in boiling water reactors (BWRs), NUPEC performed from 1987 to 1990 a series of radial void measurements at four axial locations in a full-size mock-up test facility able to simulate the high pressure, high temperature fluid conditions found in BWRs through electrically heated rod bundles. Therefore, since other important parameters such as system pressure, inlet sub-cooling and power input conditions were also supplied, these test series form a substantial database for the assessment of the accuracy of thermal hydraulic codes in predicting radial and axial assembly void distributions, under both steady-state and transient conditions.

Nevertheless, due to uncertainties coming, for example, from approximations in the physical and mathematical models, variation and imprecise knowledge of initial and boundary conditions, scatter of measured experimental data, etc., it has been recognized in the last years that uncertainty analysis would not only be necessary if useful conclusions are to be obtained from BE calculations, but would also complete the validation process of BE models (Mehta, 1998; Glaeser, 1995). Among the different approaches to perform uncertainty analysis, the one based on statistical techniques begins with the treatment of the code input uncertain parameters as random variables. Thereafter, values of these parameters are selected according to a random or quasi-random sampling strategy and then propagated through the code in order to assess the output uncertainty in the corresponding calculations. This framework has been highly accepted by many scientific disciplines not only because of its solid statistical foundations, but also because it is affordable in practice and relatively easy to implement thanks to the tremendous advances in computing capabilities.

In this paper, statistical uncertainty and sensitivity analyses using the POLCA-T code and based on the following BFBT benchmark exercises are presented:

1. *Ex. 0, Phase II*. Steady-state single and two phase pressure drop
2. *Ex. 2, Phase I*. Steady-state cross-sectional averaged void fraction
3. *Ex. 3, Phase I*. Transient cross-sectional averaged void fraction
4. *Ex. 1, Phase II*. Steady-state critical power benchmark

The chosen sampling strategy for the current studies is Latin hypercube sampling. The quasi-random LHS allows a much better coverage of the input uncertainties than SRS because it densely stratifies across the range of each input probability distribution. A comparison of both sampling methodologies applied to uncertainty analysis of cross-sectional averaged void fraction predictions based on the BFBT benchmark, can be found in (Hernandez-Solis et al., 2010), where the advantages of LHS over SRS are clearly shown. Other work comparing such advantages on uncertainty propagation in analysis of complex systems can be found in (Helton and Davis, 2003).

In the following sections, a description of the NUPEC integral test facility (ITF) and the systems employed for the different test measurements is given, along with a brief explanation of the thermal–hydraulic models used in the Westinghouse POLCA-T code. Thereafter, a deeper review on how to perform a statistical uncertainty analysis is presented, with emphasis on how it can be applied as a code validation process. Finally, quantitative and qualitative uncertainty and sensitivity analyses are shown for the different BFBT exercises.

2. Description of the NUPEC test facility

The pressure drop, void fraction distribution and critical power in a BWR multi-rod assembly, have been measured in a large scale facility operated by NUPEC in Japan. The facility is able to simulate the high pressure, high temperature fluid conditions found in nuclear reactors. An electrically heated rod bundle has been used to simulate a full scale BWR fuel assembly.

The maximum operating conditions of the ITF are 10.3 MPa in pressure, 315 °C in temperature, 12 MW in test power and 75 t/h in flow rate. It also has the capability to simulate time dependent characteristics of complicated BWR operational transients such as a turbine trip without bypass, as well as a re-circulation pump trip. The full-scale fuel assembly inside the pressure vessel corresponds to a general electric 8×8 rod design, where each rod is electrically heated to simulate an actual reactor fuel rod. The cladding, the insulator and the heater are made of inconel, boron nitride and nichrome, respectively. The heated length of the bundle corresponds to 3.7 m.

Two types of void measurement systems were employed as shown in Fig. 1: an X-ray computed tomography (CT) scanner and an X-ray densitometer.

Under steady-state conditions, fine mesh radial void distributions were measured using the X-ray CT scanner located 50 mm above the heated length (i.e. at the assembly outlet). However, the X-ray densitometer measurements of void distributions around each rod were performed at three different axial elevations from the bottom (i.e. 682 mm, 1706 mm and 2730 mm) under both steady-state and transient conditions. For the each of the four different axial locations, the cross-sectional averaged void fraction was also measured.

Absolute and differential pressures were measured using diaphragm transducers. The inlet flow rate was measured using a

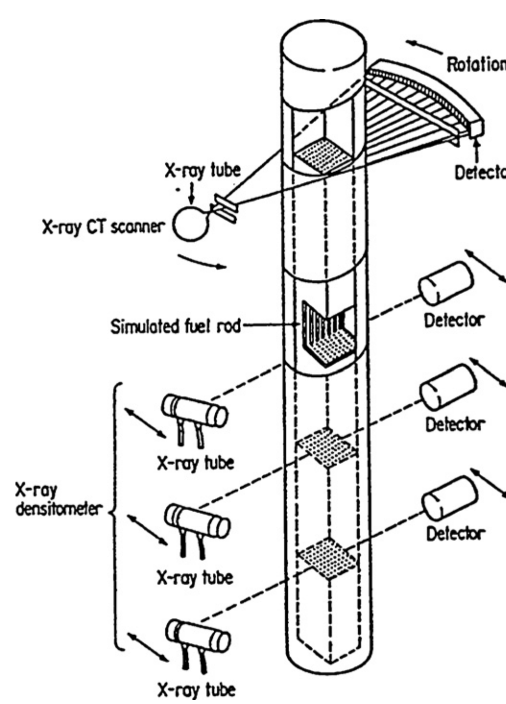

Fig. 1. Void fraction measurement system (Neykov et al., 2005).