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Best estimate plus uncertainty analysis of departure from nucleate boiling limiting case with CASL core simulator VERA-CS in response to PWR main steam line break event

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HIGHLIGHTS

- Best estimate plus uncertainty (BEPU) analyses of PWR core responses under main steam line break (MSLB) accident.
- CASL's coupled neutron transport/subchannel code VERA-CS.
- Wilks' nonparametric statistical method.
- MDNBR 95/95 tolerance limit.

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ABSTRACT

VERA-CS (Virtual Environment for Reactor Applications, Core Simulator) is a coupled neutron transport and thermal-hydraulics subchannel code under development by the Consortium for Advanced Simulation of Light Water Reactors (CASL). VERA-CS was applied to simulate core behavior of a typical Westinghouse-designed 4-loop pressurized water reactor (PWR) with 17×17 fuel assemblies in response to two main steam line break (MSLB) accident scenarios initiated at hot zero power (HZP) at the end of the first fuel cycle with the most reactive rod cluster control assembly stuck out of the core. The reactor core boundary conditions at the most DNB limiting time step were determined by a system analysis code. The core inlet flow and temperature distributions were obtained from computational fluid dynamics (CFD) simulations. The two MSLB scenarios consisted of the high and low flow situations, where reactor coolant pumps either continue to operate with offsite power or do not continue to operate since offsite power is unavailable. The best estimate plus uncertainty (BEPU) analysis method was applied using Wilks' nonparametric statistical approach. In this demonstration of BEPU application, 59 full core simulations were performed for each accident scenario to provide the minimum departure from nucleate boiling ratio (MDNBR) at the 95/95 (95% probability with 95% confidence level) tolerance limit. A parametric goodness-of-fit approach was also applied to the results to obtain the MDNBR value at the 95/95 tolerance limit. Initial sensitivity analysis was performed with the 59 cases per accident scenario by use of Pearson correlation coefficients. The results show that this typical PWR core retains design margin with respect to the MDNBR safety limit for both of the MSLB accident scenarios. The scenario with available offsite power was more restrictive in terms of MDNBR than the scenario without offsite power.

Abbreviations: BEPU, best estimate plus uncertainty; CASL, Consortium for Advanced Simulation of Light Water Reactors; CC, correlation coefficient; CFD, computational fluid dynamics; CHF, critical heat flux; CTF, COBRA-TF; DNB, departure from nucleate boiling; DNBR, departure from nucleate boiling ratio; EOC, end of cycle; FOM, figure of merit; HF, high flow; HFP, hot full power; HPC, high performance computing; HZP, hot zero power; INL, Idaho National Laboratory; LF, low flow; LOCA, loss-of-coolant accident; MDNBR, minimum departure from nucleate boiling ratio; MSLB, main steam line break; OECD, organisation for economic co-operation and development; PWR, pressurized water reactor; RCCA, rod cluster control assembly; SA, sensitivity analysis; UQ, uncertainty quantification; VERA-CS, Virtual Environment for Reactor Applications, Core Simulator; VUSAT, VERA-CS Uncertainty and Sensitivity Analysis Toolkit; WBN1, Watts Bar Nuclear Unit 1.

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1. Introduction

The increasing capabilities of high performance computing (HPC) continue to improve the fidelity of nuclear reactor safety and operational calculations in multi-physics and multi-scales. For such high fidelity calculations, proper uncertainty quantification (UQ) of the computing models of reactor neutronics and thermal-hydraulics is important to provide reliable and accurate results. UQ becomes a necessary step in reactor safety calculations that refers to the determination of uncertainty in model outputs based on the uncertainty in model inputs (Helton et al. 2006). Uncertainties in safety calculations with computer codes for nuclear reactor modeling and simulation result from code limitations, scaling inaccuracies embedded in the experimental data used for benchmarking, and uncertainties associated with the state of the reactor at the initiation of the transient (Boyack et al. 1990). UQ seeks to characterize the uncertainties associated with the reactor model and its input values. Specifically, UQ addresses epistemic uncertainty that results from the inability to know the correct value for a model input that is assumed constant (Marcum and Brigantic 2015).

The best estimate plus uncertainty (BEPU) analysis method (Boyack et al. 1990) has been developed for UQ in support of regulatory rulemaking changes of safety analyses in the nuclear industry (Zhang et al. 2016) and has been applied to loss-of-coolant accidents (LOCA) (Boyack et al., 1990; Frepoli, 2007; Martin and O'Dell, 2005; Perez et al., 2011). The BEPU method can also be applied to non-LOCA nuclear reactor accident scenarios such as a PWR main steam line break (MSLB) accident. Benchmark simulations for the MSLB scenario with coupled neutronic/thermal-hydraulic considerations have been performed under the OECD Nuclear Energy Agency (Ivanov et al. 1999). During a postulated PWR MSLB event initiated from the hot zero power (HZP) condition, increased steam flow from the broken steam pipe in one of the steam generators results in a significant reduction in the primary coolant temperature and the reactor core is returned to power with a high peak fuel rod power, thus imposing a challenge to the fuel thermal limit with respect to Departure from Nucleate Boiling (DNB). The safety analysis further assumes that the most reactive rod cluster control assembly (RCCA) is in its fully withdrawn position during the transient. A return to power following a steam line rupture is a potential problem mainly because of the high power peaking factors due to the stuck RCCA assumption. The event is terminated after the boric acid is delivered to the reactor core by the safety injection system. Although it is classified as a Condition IV event that allows fuel cladding failures for radiological dose evaluation, many plant safety analyses conservatively show no fuel rod failure by meeting the DNB Ratio (DNBR) limit with a 95% probability at a 95% confidence level (Sherder and McHugh, 1998).

The BEPU methodology consists of several sequential and logical steps in the evaluation process. Some of the key steps include: (1) Selection of a plant/core model, accident scenario, safety analysis tools, safety acceptance criteria, and figure of merit (FOM); (2) Selection of relevant physical phenomena and uncertain parameters with their respective probability distribution functions; (3) Construction of accident simulation models; (4) Random sampling of uncertain input parameters and performing accident progression simulations; (5) Determination of the 95/95 (95% probability with 95% confidence level) tolerance limits for the FOM. A BEPU analysis can be based on either the Monte Carlo approach or Wilks' nonparametric statistical approach (Wilks, 1941). The Monte Carlo approach requires a large number of computer simulations, which presents a challenge when computationally intensive high fidelity and coupled multi-physics codes are used. On the other hand, the Wilks' approach uses a relatively small number of samples to provide the safety metric at the 95/95 tolerance limit. For the Wilks' nonparametric statistical approach, all of the selected uncertain input parameters are sampled simultaneously in the space defined by their respective uncertain ranges (Zhang et al., 2016). The combined effect of the uncertain parameters on the selected FOM can then be quantified in the safety analysis. When only one FOM is considered, the number of samples required can be determined by:

$$\beta = 1 - \gamma^N \tag{1}$$

where β is the confidence level, N is the number of samples, and γ is the percentile of probability (Zhang et al., 2016). For example, using Eq. (1) to determine the 95/95 value of a single FOM, i.e. $\gamma = 0.95$ and $\beta = 0.95$, would require that the number of samples be 59.

In this work, a BEPU analysis is demonstrated using the Wilks' nonparametric statistical approach and a parametric approach with goodness-of-fit test to analyze the reactor response for two MSLB accident scenarios. The reactor core responses to the MSLB accident scenarios were simulated using the Virtual Environment for Reactor Applications, Core Simulator (VERA-CS) under development by the Consortium for Advanced Simulation of Light Water Reactors (CASL). Details on VERA-CS are provided in Section 2. The transient response of the reactor coolant system was predicted using a system analysis code RETRAN and the associated plant model for the Westinghouse-designed 4-loop PWR (Huegel et al., 1999). The reactor core boundary condition consisting of loop flow and temperature, system pressure and core average power at the DNB limiting time step of the transient, referred to as the state point, were determined from the system transient simulation. The core response was analyzed with VERA-CS at the most limiting state point determined by the MSLB accident analysis using RETRAN. In the MSLB accident, the minimum departure from nucleate boiling ratio (MDNBR) is the considered FOM. In general, UQ studies of reactor response require reactor power characteristics (power shape, peaking factors, etc.) as inputs to the thermal-hydraulic code (Marcum and Brigantic, 2015). Marcum and Brigantic (2015) found the axial and radial power factors to be the most influential parameters on the minimum departure from nucleate boiling ratio (MDNBR). Predictions of the core thermal response and power distributions could be improved with neutronic/thermal-hydraulic coupling. Therefore, it is of interest to use fully-coupled neutronics and core thermal-hydraulic calculations to perform nuclear reactor core uncertainty quantification. VERA-CS provides this fully-coupled neutronic/thermal-hydraulic capability as well as isotopic depletion. Coupled calculations pose a challenge to UQ since, in general, they require more computational power and longer simulation times than stand-alone thermal-hydraulic calculations.

2. VERA-CS

The CASL VERA-CS code currently includes three main components: MPACT for reactor physics and neutron transport, COBRA-TF for thermal-hydraulics, and ORIGEN for isotopic depletion (Collins and Godfrey, 2015). VERA-CS has been tested and applied to an array of problems including core physics analysis (Franceshini et al., 2015), full-core modeling for all of the fuel cycles of Watts Bar Nuclear Unit 1 (WBN1) (Kochunas et al., 2015; Godfrey et al., 2015), and startup core modeling for the AP1000 PWR (Franceshini et al., 2015).

MPACT is a 3D pin-resolved reactor transport code developed by the Oak Ridge National Laboratory (ORNL) and the University of Michigan. MPACT uses the 2D/1D method to solve the neutron