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The importance of input interactions in the uncertainty and sensitivity analysis of nuclear fuel behavior

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HIGHLIGHTS

- Uncertainty and sensitivity analysis of modeled nuclear fuel behavior is performed.
- Burnup dependency of the uncertainties and sensitivities is characterized.
- Input interactions significantly increase output uncertainties for irradiated fuel.
- Identification of uncertainty sources is greatly improved with higher order methods.
- Results stress the importance of using methods that take interactions into account.

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ABSTRACT

The propagation of uncertainties in a PWR fuel rod under steady-state irradiation is analyzed by computational means. A hypothetical steady-state scenario of the Three Mile Island 1 reactor fuel rod is modeled with the fuel performance FRAPCON, using realistic input uncertainties for the fabrication and model parameters, boundary conditions and material properties. The uncertainty and sensitivity analysis is performed by extensive Monte Carlo sampling of the inputs' probability distribution and by applying correlation coefficient and Sobol' variance decomposition analyses. The latter includes evaluation of the second order and total effect sensitivity indices, allowing the study of interactions between input variables. The results show that the interactions play a large role in the propagation of uncertainties, and first order methods such as the correlation coefficient analyses are in general insufficient for sensitivity analysis of the fuel rod. Significant improvement over the first order methods can be achieved by using higher order methods. The results also show that both the magnitude of the uncertainties and their propagation depends not only on the output in question, but also on burnup. The latter is due to onset of new phenomena (such as the fission gas release) and the gradual closure of the pellet-cladding gap with increasing burnup. Increasing burnup also affects the importance of input interactions. Interaction effects are typically highest in the moderate burnup (of the order of 10–40 MWd/kgU) regime, which covers a large portion of the operating regime of typical nuclear power plants. The results highlight the importance of using appropriate methods that can account for input interactions in the sensitivity analysis of the fuel rod behavior.

1. Introduction

The nuclear fuel rod of a light-water reactor consists of an oxide fuel pellet stack enclosed inside a metallic cladding tube. The pellet stack is held in place by a spring and the rod is pressurized with heat-conducting gas, facilitating heat transfer across the gap between the pellets and the cladding. Analysis of the fuel rod's behavior under irradiation in a nuclear reactor involves solving the transfer of heat from the pellet into the surrounding coolant through the gap and the cladding, the mechanical response of the pellet and the cladding to thermal and mechanical stresses, the irradiation-induced changes in materials, and the release of gaseous fission products into the gas gap. All the phenomena are interconnected, constituting an extremely complex system with rich behavior in different operating regimes and strong dependency on burnup (Bailly et al., 1999; Cacuci, 2010).

The system can be modeled numerically with dedicated fuel performance codes, which traditionally focus either on the steady

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state irradiation or the simulation of transient scenarios. In both cases, one of the main purposes of numerical modeling is to provide understanding on how the individual phenomena interact and create the overall response of the rod to external conditions. An important aspect of such a study is the acquisition of detailed data that can be used to ensure that the rod performs within the safety and regulatory guidelines, and also to guide in revising the safety regulations (Rashid et al., 2011).

There are several sources of uncertainty in fuel performance analysis. The rod's fabrication parameters, experimentally determined material properties and system parameters are never precise, but introduce various amounts of uncertainties into the system. These are propagated to model outputs such as the fuel centerline temperature, internal pressure and gap conductance. To ensure safe operation of the fuel rod, these uncertainties must be taken into account, either by conservative analysis or best estimate analysis accompanied by evaluation of the related uncertainties. In best estimate analysis quantifying both the magnitude and the source of the output uncertainties is necessary. The latter involves determining the contribution of the uncertainty of each model input to the overall output uncertainty, and is called sensitivity analysis.

Sensitivity analysis in nuclear engineering has recently concentrated mostly on the reactor physics and neutronics and, on the other hand, on thermal hydraulic modeling. Sensitivity analysis of the fuel behavior models has received considerably less attention, although it is known that uncertainties related to fuel modeling can be significant and may also have broader impact on the thermal hydraulics and neutronics modeling (Christensen et al., 1981; Wilderman and Was, 1984; Syrjälahti, 2006; Boulore et al., 2012). In addition, most studies focus on the direct (first-order) effects of the input on the model output. A common approach is to evaluate, *e.g.*, the Spearman correlation coefficients or the first-order Sobol' indices from the model output (Boulore et al., 2012; Glaeser, 2008), which neglect the higher order interactions between the input variables. However, for a complex system such as the fuel rod (Rashid et al., 2011), these interactions can play a major role in the overall output uncertainties, and should not be neglected *a priori* (Saltelli et al., 2008).

In this paper, we investigate the role of input interactions in the uncertainty and sensitivity analysis of the nuclear fuel rod. For this purpose, we use the FRAPCON fuel performance code (Geelhood et al., 2011a,b) and perform statistical analysis of the code's output by evaluating both the conventionally used Spearman correlation coefficients (Draper and Smith, 1998; Kvam and Vidakovic, 2007) and the Sobol' sensitivity indices (Saltelli et al., 2008; Sobol', 1993). We consider a steady state scenario, and focus on identifying the major sources of uncertainties, characterizing interactions between inputs and their dependencies on burnup. Since the initial states of transient calculations with non-fresh fuel are usually generated by such steady state simulations, our results have direct relevance for transient analyses also.

The structure of the paper is as follows. In Section 2, we discuss the fuel performance code FRAPCON-3.4 used in the analysis of the scenario, and in Section 3 we describe the statistical analysis methods. The specifications of the modeled scenario and the input uncertainties are given in Section 4. In Section 5, we first show the best estimate plus uncertainty results of the scenario. Then, we illustrate how the data can be analyzed with the conventional methods and show that such an analysis remains incomplete in most cases. In Section 5.4, we repeat the analysis using the Sobol' variance decomposition method. We show that the variance decomposition analysis gives a consistent and much more complete picture of the system's response to the input uncertainties. According to the analysis, the shortcomings of the Spearman correlation method are due to non-additive interactions between the input variables. These produce uncertainties, whose source can be identified only with higher order methods such as the variance decomposition. We also compare the quantitative effectiveness of the variance decomposition to the Spearman correlation method, showing significant improvement. Using the evaluated total effects, we rank the input uncertainties with respect to their overall importance. We summarize the results of the paper in Section 6.

2. The fuel performance code FRAPCON

The fuel performance model used in this study is the FRAPCON-3.4 code that is maintained by the Pacific Northwest National Laboratory (Geelhood et al., 2011a,b). FRAPCON is a deterministic fuel performance code that calculates the steady-state response of light-water reactor fuel rods during long-term burn-up. Boundary conditions such as the power history and the coolant properties, in addition to the rod fabrication parameters, are supplied as input. The output of the code comprises several observables, including the fuel and cladding temperature distributions in the radial and axial directions, mechanical deformations of the pellet and the cladding, internal pressure, gap conductance, and so on. The physical phenomena described in the code include heat conduction through the fuel, gap and cladding into the coolant, fuel densification and swelling, cladding elastic and plastic deformations, fission gas release and cladding oxidation. The thermal and mechanical solutions are strongly coupled through the gap conductance, which is a function of, for example, pellet and cladding dimensions, and has a strong influence on the radial heat conduction. In addition to the gap conductance, the different submodels become interconnected via phenomena such as the fission gas release, which influences both the thermal and mechanical solutions through gap conductance and pressure and is itself affected by the fuel temperature.

The strong coupling between the different phenomena imply that the model is highly nonlinear and interactions between input variables are very likely. To analyze the output of such a model, we use the statistical methods described in Section 3, which involve Monte Carlo sampling of the input variables from given probability distributions. As FRAPCON is a deterministic code (one input producing exactly one deterministic solution), the sampling of input variables has to be done by external means. For this purpose we have developed a Python script that performs the sampling using methods discussed in Section 3 and generates the FRAPCON input files (Ikonen, 2012). The code is then run and the results post-processed on a Linux cluster.

3. Statistical analysis methods

3.1. General considerations

A fuel performance code is typically very complex, with dozens of input variables and several output observables of interest. In the analysis of such a complex system, determining the propagation of uncertainties from input to output can be a delicate task. Numerous alternative methods of uncertainty and sensitivity analysis exist, each having their advantages and disadvantages. Two ways to group different methods is to distinguish between deterministic and statistical methods, and local and global sensitivity analysis (Saltelli et al., 2008; Ionescu-Bujor and Cacuci, 2004; Cacuci and Ionescu-Bujor, 2004).

With deterministic methods, one estimates the response of the model output to the changes in the value of an input variable either by analytical means, or by deterministic sampling of individual values. With statistical methods, on the other hand, one samples the values of the input variables from a distribution and calculates the