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Uncertainty and sensitivity analysis of the nuclear fuel thermal behavior

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HIGHLIGHTS

- ► A complete quantitative method for uncertainty propagation and sensitivity analysis is applied.
- ► The thermal conductivity of UO_2 is modeled as a random variable.
- ► The first source of uncertainty is the linear heat rate.
- ► The second source of uncertainty is the thermal conductivity of the fuel.

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ABSTRACT

In the global framework of nuclear fuel behavior simulation, the response of the models describing the physical phenomena occurring during the irradiation in reactor is mainly conditioned by the confidence in the calculated temperature of the fuel.

Amongst all parameters influencing the temperature calculation in our fuel rod simulation code (METEOR V2), several sources of uncertainty have been identified as being the most sensitive: thermal conductivity of UO_2, radial distribution of power in the fuel pellet, local linear heat rate in the fuel rod, geometry of the pellet and thermal transfer in the gap. Expert judgment and inverse methods have been used to model the uncertainty of these parameters using theoretical distributions and correlation matrices.

Propagation of these uncertainties in the METEOR V2 code using the URANIE framework and a Monte-Carlo technique has been performed in different experimental irradiations of UO_2 fuel. At every time step of the simulated experiments, we get a temperature statistical distribution which results from the initial distributions of the uncertain parameters. We then can estimate confidence intervals of the calculated temperature. In order to quantify the sensitivity of the calculated temperature to each of the uncertain input parameters and data, we have also performed a sensitivity analysis using the Sobol' indices at first order.

1. Introduction

1.1. General context

Most of the physical phenomena occurring in nuclear fuel elements during their irradiation in nuclear power plants are controlled by the temperature. As a matter of interest, we can mention fission product release, fuel densification and swelling, creep of cladding and fuel, etc. In our fuel behavior simulation code, the models corresponding to all these phenomena are very much sensitive to the temperature calculated in the fuel. The confidence we have in the calculated behavior of the code depends mainly on the confidence in the calculated temperature.

1.2. Methodology

In the general framework of uncertainty analysis in fuel simulation codes, CEA follows the methodology recommended by the ESReDA organization (European Safety, Reliability and Data Association) (De Rocquigny et al., 2008). These recommendations consist of a decomposition of the global problem of uncertainties into four steps which have been adapted to our specific problem (Fig. 1).

First of all stands the specification of the problem (Step A). The fuel performance code we consider is the CEA code METEOR V2. It is a 1D-radial thermal–mechanical simulation code. The variable of interest is the central temperature in the fuel pellet which is calculated by a thermal model embedded in the METEOR V2 code (Struzik et al., 1997). The quantity of interest of this output variable

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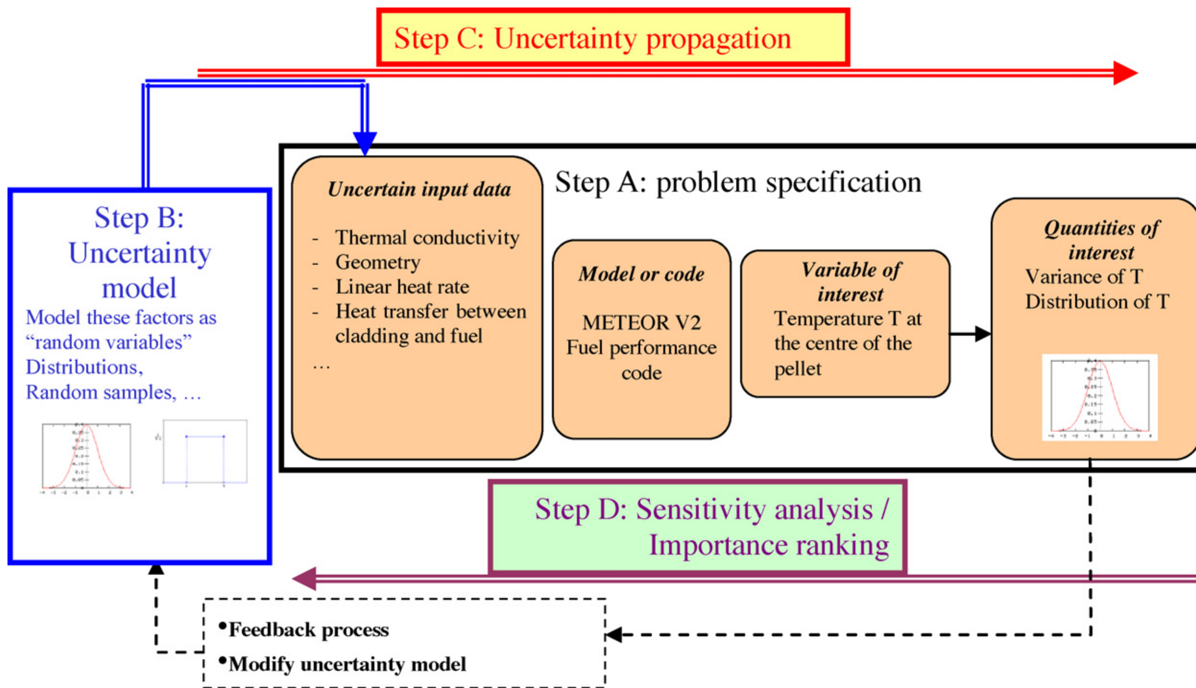

Fig. 1. General recommended methodology adapted to the thermal behavior simulation.

is mainly its variance, but also its whole distribution for different levels of temperature. Because the thermal model depends on the irradiation regime, in this paper we are interested in 2 different regimes:

- low burn-up, nominal PWR power;
- high burn-up, and transient regime.

To cover these 2 regimes, 2 experiments of interest are considered, and they are presented below (Section 1.3).

The calculation depends on a large number of input data such as geometry of the fuel elements, material thermal properties, filling gas composition in the free volume of the rod, irradiation power history, coolant temperature in the reactor, porosity of the pellet, corrosion behavior of the cladding, etc.

In the specification phase, the uncertain input parameters and data have to be listed. Depending on the considered experiment, we have limited the uncertain input parameters and data to:

- fuel thermal conductivity;
- linear heat rate;
- heat transfer between fuel pellet and cladding;
- divided into the transfer model itself, and the fuel pellet relocation model that controls the size of the gap;
- geometry of the pellet (inner radius in case of hollow pellet with a thermocouple);
- radial distribution of power.

The second step consists of the modeling of the uncertainties as random variables (Step B). The uncertainty model will be a set of parametric distributions (e.g. Uniform or Gaussian) with some independence hypotheses or approximate linear or rank correlations. This modeling has to be done as far as possible using all the experimental data available, and several methods can be used: direct observations, expert judgment, inverse methods, etc.

Once all the uncertain input parameters and data are modeled, the propagation through the code can be performed (Step C). This needs first the building of a design of experiment (DoE) of the input data (e.g. by a Monte-Carlo Sampling) which is propagated through the code to result in a representation of the variable of interest and a quantification of the quantities of interest mentioned above. Ideally, this step has to be completed by a sensitivity analysis step (Step D) which refers to the computation of sensitivity or importance indices of the uncertain input parameters with respect to the output variable. A large variety of methods are available, and in our case, we have focused on variance based methods (Sobol' indices). They quantify the part of the variance of the output variable due to the variance of each input parameters and data.

1.3. Experiments of interest

1.3.1. GRIMOX2

GRIMOX2 was an experiment conducted in the SILOE nuclear test reactor in Grenoble during the nineties (Caillot and Delette, 1998). It had been performed in order to study the thermal behavior of MOX fuel compared to the UO_2 fuel in the beginning of irradiation.

For that purpose a short fuel rod of PWR diametrical geometry including MOX and UO_2 pellets was irradiated in a boiling capsule pressurized to 13 MPa placed at the edge of the SILOE reactor core. At both ends of the experimental rod, the pellets were drilled in order to introduce a thermocouple. So this device allowed the measurement of the centerline temperature of both fresh fuel, UO_2 and MOX, during an irradiation up to 0.5 at%. The time duration of the experiment was about 4000 h (Fig. 2). The uncertainty on temperature measurement announced by the operator is $\pm 10\,^{\circ}C$.

The experimental fissile power was determined by the mean of neutron flux detector during the experiment and it was cross checked by quantitative gamma spectrometry after irradiation. The order of magnitude of power level is comparable to the PWR one in standard and incidental conditions.

Although two types of fuel had been tested in this experiment this paper only deals with the UO_2 stack.