Methodologies for Sensitivity/Uncertainty Analysis Using Reactor Core Simulators with Application to Pressurized Water Reactors

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ABSTRACT

The results presented here are part of a PhD thesis that participates in the 12th ENEN PhD Event & Prize 2018. This thesis work has been done in the framework of the “Uncertainties Analysis in Modeling” Benchmark using the core simulators developed at Universidad Politécnica de Madrid (SEANAP and COBAYA4), with the aim of transforming these best-estimate codes to best-estimate plus uncertainty methodologies. One application of each system is presented.

First, uncertainty quantification and data assimilation techniques have been implemented in SEANAP, and applied to the core design of a PWR cycle. Predictions of the critical boron concentration along burnup are improved by considering measurements of previous cycles, obtaining best-estimate results in better agreement with the actual values, and an impressive reduction in the uncertainty.

Second, using COBAYA4 and a sampling-based uncertainty quantification methodology, the impact of the homogenization level, nodal or pin-by-pin, on the uncertainty predicted for the power peaking factors is analyzed. The impact of the mesh size is very significant, yielding non-normal probability density functions and highlighting the need of performing full core calculations at the pin-level to get reliable estimates of the uncertainties in the peaking factors.

1 INTRODUCTION

Traditionally, safety analyses of nuclear power plants were carried out using pessimistic methods and hypotheses, forcing the results to be conservative. In recent years, realistic (best-estimate) simulations are being considered for application in safety analysis. However, if the best-estimate predictions are to be compared with limits of acceptance, they should be complemented with a rigorous quantification of the impact of the input parameters uncertainties on
the responses of interest. These approaches are named best-estimate plus uncertainty (BEPU) methods.

Aware of the importance of the role of uncertainties, the OECD/NEA launched the “OECD Benchmark for Uncertainty Analysis in Best Estimate Modeling for Design, Operation and Safety Analysis of LWRs (UAM)”, with the aim of developing uncertainty quantification methods for light water reactors simulations [1].

This work has been done in the framework of the mentioned UAM Benchmark, with the main objective of implementing uncertainty analysis techniques in the core simulators developed at Universidad Politécnica de Madrid (SEANAP and COBAYA4), in order to convert those best-estimate approaches in BEPU methodologies. One application of each system (SEANAP and COBAYA4) is presented in this paper.

The results presented here are part of a PhD thesis [2] that participates in the 12th ENEN PhD Event & Prize 2018.

2 UNCERTAINTY QUANTIFICATION AND DATA ASSIMILATION USING SEANAP

The first system, SEANAP, is a complete set of tools for the simulation of PWR, which has demonstrated a good agreement between calculated and measured values in several Spanish nuclear power plants [3]. A sampling-based uncertainty quantification methodology has been implemented, where a set of perturbed cross section libraries is applied in random SEANAP calculations to obtain a set of responses. Then, this set of responses is statistically analyzed to obtain the uncertainty information of interest.

After the uncertainty quantification, a data assimilation method has been implemented, in collaboration with AREVA GmbH, applying MOCABA [4]. Using this Monte Carlo-based Bayesian inference procedure, measurements of previous cycles of a power plant are used to improve the predictions of subsequent cycles.

2.1 Methodology

This uncertainty quantification and data assimilation method has been applied to the core design of a PWR cycle, and in particular, to the boron letdown curve. Using measurements of one cycle, predictions of the next cycle are updated.

Figure [1] shows a scheme of the applied methodology, which is divided in three steps. First, 200 random nuclear data libraries are generated with NUDUNA [5] using covariance data from the ENDF/B-VII.1 evaluation for the following isotopes: $^{235}\text{U}$, $^{238}\text{U}$, $^{239}\text{Pu}$, $^{1}\text{H}$, $^{16}\text{O}$ and $^{10}\text{B}$. These random libraries are combined in the reference library of SEANAP yielding a set of random nuclear data libraries that will be applied in the second step in the core simulations. The core simulation results are then used in a statistical analysis to estimate the prior mean values and covariances of the cycle I and II observables.

In the last step, measurement information of cycle I is added by applying MOCABA, updating the prior cycle I and cycle II predictions. Hence, cycle I is used as a benchmark to improve the predictions of cycle II. This way of applying MOCABA will be referred to as direct updating.

There is a different way of obtaining updated results using MOCABA, which is updating the SEANAP nuclear data library using the measurements of cycle I, and performing after that the simulation for cycle II with the adjusted library as input. This will be referred as nuclear data updating results.
2.2 Core description and results

The analysis is performed using two consecutive cycles (I and II) of a 157 fuel assemblies core. The refueling pattern (figure 2) for both cycles is quite similar, where all fresh fuel assemblies have an initial enrichment of 3.6 w/o except for the fuel assemblies with 20 WABA rods in cycle I, whose enrichment is 3.24 w/o.

The results of the application of the methodology are represented in figure 3. For all burnup steps, considering the previous cycle measurements leads to a better prediction of the boron letdown curve, that is, the difference between predicted and measured values is significantly reduced. Using direct updating or the nuclear data updating procedure yields equivalent results. Second, the uncertainty in the boron calculation is reduced by one order of magnitude, from values larger than 40 ppm to values around 3 ppm.

This impressive improvement in the prediction of the boron letdown curve by including

![Figure 2: Refueling pattern for the two cycles In blue: fresh fuel without burnable absorber. In red: fresh fuel with burnable absorber, the number indicates the number of rods with poison. In white: old fuel](image-url)
measurements of the previous cycle is explained by Figure 4. It shows large correlations between the boron values of cycle I and cycle II in the range 0.92-0.99, which reflects a high similarity between both cycles.

A comprehensive analysis of the SEANAP/MOCABA system for data assimilation is available at [6].

![Figure 3: Differences between measured boron concentrations and respective prior predictions, posterior predictions, and predictions obtained with a MOCABA-updated nuclear data library. The error bars represent the respective 1σ uncertainties](image)

![Figure 4: Prior correlation matrix of boron concentration values for Cycle I and Cycle II at different burnup steps](image)
3 SENSITIVITIES OF POWER PEAKING FACTORS TO THE HOMOGENIZATION LEVEL USING COBAYA4

The second application presented here makes use of the core simulator COBAYA4, which is the evolution of the core simulator in SEANAP and is able to perform full core 3D simulations at nodal and pin-by-pin scales [7].

3.1 Methodology

A sampling-based uncertainty quantification methodology has been implemented. First, a set of random few-group macroscopic cross section libraries is generated using SAMPLER, the NEWT lattice code and the 56 group covariance matrix of SCALE 6.2 [8]. NEWT was selected as lattice code because it can provide cross sections for COBAYA4 homogenized at both nodal and pin levels. This ability of COBAYA4 to perform full core calculations at nodal and pin levels will allow to analyze the impact of the spatial mesh used in the diffusion calculations on the uncertainty results of core parameters.

3.2 Core description and results

Uncertainties in the nuclear data libraries of SCALE 6.2 have been propagated to full core parameters for the PWR TMI-1 Exercise of UAM Benchmark. The core is composed of 177 fresh fuel assembly with homogeneous axial compositions and the considered scenario is all-rods-in (ARI) and hot zero power (HZP).

Emphasis was put on assessing the impact of the homogenization level (nodal or pin-by-pin) on the uncertainties predicted by core simulators on the power peaking factors. In the following, results will be presented for the assembly-wise enthalpy hot channel factor \( F_{\Delta H} \), which is defined as the maximum assembly power (axially integrated) divided by the average assembly power in the core.

The results for this power peaking factor obtained using both solvers are shown in table I. Mean values using both solvers are in good agreement. However, uncertainties computed using the pin-by-pin solver are much higher than the values computed using the nodal solver, reflecting the strong influence of the spatial discretization on the uncertainties of the peaking factor.

<table>
<thead>
<tr>
<th></th>
<th>Mean value</th>
<th>Standard deviation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nodal</td>
<td>1.643</td>
<td>0.498%</td>
</tr>
<tr>
<td>Pin-by-pin</td>
<td>1.606</td>
<td>1.151%</td>
</tr>
</tbody>
</table>

To better understand this behavior, the histogram of this parameter is represented in figure 5. While the values computed with a nodal mesh follow a normal distribution, the ones provided using a pin mesh are far from normality.

An in-depth analysis showed that some nuclear data perturbations in the pin-by-pin calculations predicted a power distribution with the hottest channel located at different core positions. The hottest assembly was found at two different positions depending on the random sample as shown in figure 6: at position A in the 77% of the samples and at position B in the remaining 22%. However, using the nodal solver, the hottest channel was always located at position A. In figure 7 the probability density functions of the assembly power for assemblies in positions A
and B are displayed, being overlapped. Since this peaking factor is the maximum value of those random variables, it can correspond to position A or B depending on the simulation.

Figure 5: Histograms of the assembly-wise enthalpy hot channel factor

Figure 6: Positions of the hottest fuel assembly predicted by the pin-by-pin solver

Figure 7: Probability density functions of the assembly power (axially integrated) for assemblies at positions A and B, predicted by pin-by-pin simulations
A comprehensive analysis of this and other responses using COBAYA4 uncertainty quantification capabilities is available at [9].

4 CONCLUSIONS

The work performed in this thesis consisted on analyzing and implementing uncertainty analysis techniques in core simulators, in particular in the computational tools developed at UPM for PWR analysis (SEANAP and COBAYA4), converting this best-estimate approaches in BEPU methodologies.

Two applications of the systems have been presented here. First, uncertainty quantification and data assimilation techniques in SEANAP have been developed and applied to the core design of a cycle, and in particular to the boron letdown curve. The application of the MOCABA updating leads to major improvements for the best-estimates and to massive uncertainty reductions. SEANAP system was lacking uncertainty quantification techniques that have been now incorporated.

The second application presented uses the COBAYA4 core simulator to propagate uncertainties in nuclear data in full core standalone neutronics calculations. In particular, for the presented application, it was concluded that the spatial mesh has a large impact on the probability density functions of the power peaking factors. This behavior highlights the need of performing full core calculations at the pin-level to get reliable estimates of the uncertainties in peaking factors.

ACKNOWLEDGMENTS

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REFERENCES


