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Uncertainty Propagation and Sensitivity Analysis of Coupled Thermalhydraulic-Neutronic Nuclear Power Plant Simulations: Influence of Uncertainty in Neutronic Data

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Abstract

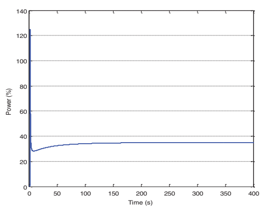
This paper presents a study of the influence of the uncertainty in the macroscopic neutronic information for a three-dimensional PWR core model on the core power and reactivity during a Reactivity Induced Accident (RIA). The analysis uses a coupled thermal-hydraulic and neutronic model for RELAP5-PARCS. The SIMTAB methodology provides neutronic information. The statistical methodology assumes uncertainty in the macroscopic cross sections, whose *pdf*s are sampled and the Noether-Wilks formulas for Tolerance Intervals determine the sample size. Non-parametric statistical methods determine the intervals of tolerance and sensitivities of the sample for core power, fuel enthalpy, fuel temperatures and reactivity values.
Key Words: Uncertainty, Sensitivity, Coupled TH-3DNK codes, Wilks Methodology

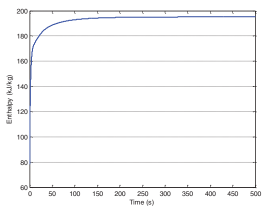
1. Introduction and Simulation Model Description

The model object of the uncertainty and sensitivity analysis presented in this paper simulates the behavior of the Almaraz NPP core configuration in a REA (Rod Ejection Accident) with all the control rods inserted (ARI). The REA is initiated by the ejection of a control rod caused by the failure of its driving mechanism. The physical description of the reactor response is based on the coupled neutronic-thermalhydraulic systems analysis program standard in the industry RELAP5/PARCS v2.7.

The data needed for the complete neutronic description of the core behavior, the cross-sections sets, are obtained by using the SIMTAB methodology developed at the Polytechnic University of Valencia (UPV). In order to qualify the results obtained using these cross-sections sets, the model´s results for steady-state conditions have been compared to the results of a stand-alone core simulation performed with SIMULATE3, a standard and extensively validated core analysis tool in the industry.

The transient is started by the ejection of rod with the maximum reactivity worth until it is completely extracted in 0.1s. A continuous reactivity insertion drives the transient behavior of the reactor (see Fig. 1). As a result, the amount of energy deposited in the nuclear fuel (see Fig.2) stays below the maximum values accepted as safety limits, and assuring that the fuel will not be damaged.





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In this paper, we present the cross-sections sensibility analyses performed using the SIMTAB methodology. The process consists of the perturbation of each one of the cross-sections and the simulation in each perturbation of the RIA case explained.

2. Uncertainty and Sensitivity Analysis Methodology

The results presented in Figures 1 and 2, are the so-called Best Estimate evolution of the parameters shown. The quantification of the uncertainty of the input neutronic parameters is carried out through the assignment of the so-called Subjective Probability Density Functions (SPDF) to reflect how well this uncertainty is known. The process of uncertainty starts with the generation of a random sample of size N, $(\boldsymbol{X})_N$, from these SPDFs, where $\boldsymbol{X}$ is a vector containing the sampled values for all the m neutronic data inputs $X_i=1,..m$. The mathematical and physical model, i.e., the computer program, is then executed N times, thus generating a sample of the output variable Y, $(\boldsymbol{Y})_N$. An statistical analysis of $(\boldsymbol{Y})_N$ with non-parametric methods can produce tolerance intervals, which are able to quantify the uncertainty of Y.

The use of Tolerance Intervals based on non-parametric methods opens the possibility to the determination of the minimum number of code calculations, i.e., the sample size N, by the Noether-Wilks' formula[13, 14], which determines the value of N according to the desired p and γ. Thus, the number of required calculations ***does not depend*** on the number of input parameters or on any assumption about the probability distribution of the results[15].

We have also used statistical linear measures of sensitivity based on the Simple Correlation Coefficient SCC or Pearson's moment product, and the Partial Correlation Coefficient (PCC)[16, 17]; the latter eliminates the linear influence of the remaining input variables on the output, leaving only that of the input variable whose sensitivity is being calculated. In order to deal with models which are not clearly linear, Simple (SRCC) or Partial Rank Correlation (PRCC) coefficients have also been calculated[17]. The sensitivity analysis allowed us to determine the most influential neutronic parameters which influence in the uncertainty of the power and reactivity of the reactor.

3. Results and Conclusions

The uncertainty analysis has shown that variations about 1% have little influence in the output variables of interest, e.g. similar peak power and time at which it is reached. Comparison of the type of *pdf* used has shown that a normal *pdf* results in more "conservative" results from the point of view of maximum peak value and spread of the time of the power transient, if compared to an uncertainty quantified with a uniform *pdf*. Similar conclusions can also be extracted for the uncertainty in the reactivities. Increasing the uncertainty, there is a spread in the time at which the power rises and falls, but the maximum value is not greatly changed, even for uncertainty as large as 10%.

The sensitivity analysis has shown that the most influential uncertainties correspond to the fast diffusion coefficient D_F, which determines the leakage, the scattering cross section $\Sigma_{scattering}$, which determines the moderation, and both fission cross sections, which determine the rate of fission power release.

4. References

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