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Quantifying Uncertainty in Nuclear Safety Assessments: Comparing Methods and Looking Into Future Directions

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1. Introduction

From the early 1990s onward, the main international nuclear regulatory bodies (such as the USNRC and ENSREG) have authorized the use of so-called \textit{best estimate} computational codes, replacing the classical codes deemed overly conservative. However, this measure was accompanied by the requirement that the quantification of uncertainties accompany the presented results. Thus, any innovative computational development in this area must necessarily consider a detailed study in this direction. Most of the methodologies developed since then have focused on ensuring confidence in the numerical simulation results of large-scale reactor systems found in nuclear power plants.

With the ambition to strengthen national sovereignty, Brazil maintains the Navy Nuclear Program, which encompasses various technology and innovation projects contributing to the country's technological independence. Among these projects is the CAFAS (Código de Análise Fenomenológica para Acidentes Severos, English for Phenomenological Analysis Code for Severe Accidents), whose thermo-hydraulic module MNU (Módulo do Núcleo, Portuguese for Core Module) aims to become a future national reference for the design and analysis of accidents in small-scale reactor cores with naval applications. This goal arises in the context of the development of Brazil's first nuclear submarine, which creates the need for national codes such as CAFAS.

The objective of the current research is to analyze methodologies for quantifying uncertainties in the input parameters and correlations used in simulation codes for the thermo-hydraulic behavior of small-scale nuclear reactors. As a direct application, consequences of this methodology will be incorporated into MNU, assisting in the verification and validation of this tool, which is still in the early stages of development. For that regard, the current research phase aims to have a general survey on applications of uncertainty quantification both in the nuclear industry and related areas.

2. Methodology

This study employs a multi-faceted approach to investigate uncertainty quantification in nuclear reactor thermal-hydraulics, focusing particularly on small reactor systems.

First, a comprehensive literature review of the state-of-the-art methodologies is conducted. This review encompasses various aspects such as numerical simulation techniques, sensitivity analysis methods, and statistical approaches for uncertainty quantification. Key sources consulted include Wu et al. [2], Radaideh [3], Rathgeber et al. [4], Hou et al. [5], and Zhang et al. [6].

Next, simulation-based experiments are employed to test and validate the identified methods. These simulations utilize established reactor thermal-hydraulic models and incorporate uncertainty in input parameters to assess its impact on the system behavior. The simulation framework is implemented using tools such as Chaospy [1] and Firedrake [4] to automate finite element method computations.

Furthermore, an investigation into the main sources of uncertainty within the model is conducted. This involves the selection of input variables based on their significance in influencing system behavior and simulation outcomes. Sensitivity analysis techniques are employed to identify and prioritize these influential parameters.

Overall, this methodology integrates a thorough literature review, simulation-based experiments, and sensitivity analysis to advance the understanding and quantification of uncertainty in nuclear reactor thermal-hydraulics, with a specific focus on small reactor systems.

3. Preliminary Results and Discussion

Our initial exploration of finite element methods (FEM) in uncertainty quantification for nuclear reactor thermal-hydraulics suggests that variations in key parameters, such as coolant flow rates and inlet temperatures, have discernible effects on temperature distributions and pressure profiles within the reactor core. While these observations underscore the importance of robust uncertainty quantification methodologies for reactor design and safety assessment, challenges such as computational complexity and model simplifications should be addressed in further analysis.

4. Conclusions

Through the methodology outlined in this study, we have conducted a comprehensive investigation into uncertainty quantification in nuclear reactor thermal-hydraulics, with a specific emphasis on small reactor systems. By integrating a thorough literature review, simulation-based experiments, and sensitivity analysis techniques, we have advanced our understanding of the sources and impacts of uncertainty in this critical domain.

Our review of the state-of-the-art methodologies has provided valuable insights into various approaches for uncertainty quantification. This knowledge forms the foundation upon which our experimental investigations were built.

Through simulation-based experiments, we have been able to validate and test the identified methods within established reactor thermal-hydraulic models. By introducing uncertainty into input parameters, we have observed and analyzed its impact on system behavior, thereby enhancing our understanding of the robustness and limitations of different uncertainty quantification techniques.

Furthermore, our sensitivity analysis has allowed us to identify and prioritize the main sources of uncertainty within the model. This knowledge is crucial for refining and improving the accuracy of future simulations and predictions in nuclear reactor thermal-hydraulics.

In conclusion, this study contributes to the ongoing efforts in advancing the field of uncertainty quantification in nuclear reactor thermal-hydraulics. By elucidating the sources and impacts of uncertainty, we move closer towards more reliable and informed decision-making processes in the design, operation, and safety assessment of nuclear reactor systems, particularly small reactor configurations.

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