***Abstract*:**

Nuclear energy has been identified as a potential contributor to reduce the greenhouse effect gases emissions and consequently it should play an important role in the electricity production mix of every country. Developing countries can also find in nuclear energy the solution to power their expected growth. The current fleet of nuclear reactors is based on a technology designed in the early stages of the nuclear power development and presents as major concerns its associated nuclear waste disposal, its high safety requirements and its financial hurdles.

The Generation IV International Forum (GIF) is an international R&D platform with the objective to coordinate the envisaged efforts needed to develop a new generation of nuclear reactors. The organisation has identified a selection of nuclear designs with potential to meet the highest technological objectives; improved fuel management capabilities, higher reliability and safety standards and economic competitiveness regarding any other energy source.

Among these designs, Liquid Metal cooled Fast Breeder Reactors (LMFBRs) stand out due to its potentiality to achieve the ambitious goals presumed for the new generation of nuclear reactors. The Sodium Fast Reactor (SFR) and Lead Fast Reactor (LFR) are the preferred technologies due to their benign thermo-mechanical and chemical coolant characteristics.

In order to assess the compliance of the proposed designs with the highest safety standards it is needed to apply computational tools able to simulate the system behaviour under conditions that may overtake the reactor safety limits from the early stages of its design process. These computational tools should also be detailed enough to take into account the particular phenomena related with fast reactors including three-dimensional effects that may occur during certain transients.

The objective of the research work in this PhD thesis has been to develop, assess and apply advanced computational tools and models for the design basis analysis of liquid metal cooled fast breeder reactors.

The first part of the thesis outlines the development of a one-dimensional thermal-hydraulic model of the European Sodium Fast Reactor (ESFR) design with point kinetic neutronic feedback, which has been benchmarked with its peers in the framework of the FP7-CP-ESFR project, using a state-of-the-art thermal-hydraulic system code TRACE. The model is applied to analyse the system behaviour withstanding the most severe Design Basis Accident initiator identified in the safety assessment of the design, the Unprotected Loss of Flow (ULOF) transient. The same process is applied to develop an equivalent model of the Lead Fast Reactor (LFR) demonstrator called ALFRED in the framework of the FP7-CP-LEADER project.

The last and most important part of the research work focuses on the development and application of the extension of the one-dimensional model into a three-dimensional thermal hydraulic model, which is coupled with a spatial neutronic model that upgrades the point kinetic neutronic feedback used previously. The coupled TRACE-PARCS system codes are used. These coupled tools allow performing calculations with a higher detail level, and especially, reproducing asymmetric phenomena, such as the coastdown of single primary and secondary pumps or the withdrawal of a peripheral control rod bank, demonstrating the unique capability

of the code to simulate such transients and the capability of the design to withstand them under safety limits.

The final chapter of the PhD thesis presents the conclusions and main contributions of the research work and envisage topics for further research.

Key-words: Generation IV, Sodium Fast Reactor, ESFR, CP-ESFR, Lead Fast Reactor, ALFRED, LEADER, Multi-physics, Thermal-hydraulics, Neutronics, Asymmetric transients, ULOF, UTOP, TRACE, PARCS.