Investigation of dynamic behavior in generation IV fast spectrum reactors during unprotected transients

PhD Thesis booklet

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Introduction

Fast reactor technologies are actively investigated to ensure the long-term sustainability of nuclear power. The new systems' main goal is to produce less radioactive waste and operate safer than traditional thermal reactors [1]. The Generation IV International Forum (GIF) has selected the sodium-, lead-, and gas-cooled fast reactors as the most promising concepts and aims to develop fourth-generation nuclear systems for industrial use by 2030 [2]. The most knowledge is accumulated for sodium-cooled fast reactors (SFRs) since several operated before (e.g., Phénix, Monju) or still works today (e.g., BN-800, CEFR). Lead-cooled fast reactors (LFRs) were first developed for submarine propulsion but also explored today, for example, in Europe (e.g., MYRRHA, ALFRED) and in Russia (e.g., BREST-300). Gas-cooled fast reactors (GFRs) are also actively investigated, but GFRs have never been built before. The first GFR could be the ALLEGRO helium-cooled fast reactor, developed by the CEA until 2009 and further developed by the V4G4 Centre of Excellence [3]. The ALLEGRO's objective is to demonstrate the safe applicability of GFR specific systems and innovative fuel materials.

Before building a nuclear reactor, safety analyses must be performed to prove the fulfillment of the safety requirements. A necessary part of the safety analyses deals with unprotected transients, where it is assumed that the primary reactor shutdown system is not working. The investigation of the reactors' dynamic behavior during unprotected transients requires the correct calculation of reactivity effects. For this, appropriate calculation tools should be used by considering that neutronics and thermal-hydraulics are closely coupled. The straightforward approach uses a thermal-hydraulics code with point-kinetics methodology. This method is satisfactory in specific cases, but it can lead to distorted results because essential 3D effects are often neglected in the point-kinetics approach. A more precise and to this day, the state-of-the-art practical method for full-core analysis uses a coupled three-dimensional nodal neutronics/thermal-hydraulics code to consider changes in power distribution and complex spatial dependence of reactivity feedbacks during unprotected transients. It is challenging that nodal neutronics tools such as the KIKO3D [4] and DYN3D [5] well-tried for analyzing thermal reactors are not suitable for fast spectrum reactors, mainly because fast and thermal reactor reactivity feedbacks considerably differ, requiring different modeling methods.

Several nodal neutronics codes have been developed in the last couple of years and are currently under development for dynamic analyses of fast reactors. The DYN3D was recently extended for the investigation of SFRs [6]. The PARCS code, coupled with the ATHLET thermal-hydraulics code, was also applied to simulate an SFR transient [7]. The KIKO3DMG – based on the KIKO3D – is developed to analyze fast reactors in the Centre for Energy Research [8]. The main aim of my PhD research was to prepare and extend the KIKO3DMG code for the investigation of the dynamic behavior of fast reactors and perform the first-ever coupled neutronics/thermal-hydraulics analysis of an unprotected transient of the ALLEGRO.

Objectives

Recent studies of the ALLEGRO [9] and a 3600 MW_{th} SFR [10] brought to the fore that the calculation of the essential core safety parameters (e.g., reactivity coefficients, power peaking factors) are loaded with significant uncertainties originated from nuclear data and modeling uncertainties. Because of the uncertainties and the complex processes, it was not apparent what the relevant core safety parameters are during unprotected transients. The relevant core safety parameters can be identified by performing sensitivity and uncertainty analyses using the ATHLET thermal-hydraulics code with point-kinetics methodology. So the first aim of my PhD research was to determine the most important core safety parameters of the ALLEGRO and a 3600 MW_{th} SFR during an unprotected transient overpower (UTOP) and an unprotected loss of flow transient (ULOF).

For more precise analyses of unprotected transients, a coupled nodal neutronics/thermal-hydraulics code can be used. Nodal neutronics codes require homogenized cross-sections (group constants) as a function of several state parameters (e.g., fuel temperature, coolant density). The necessary state parameters can be selected knowing the relevant core safety parameters for the transients to be examined. The Serpent Monte Carlo code is an increasingly popular tool for group constant generation. The Serpent code can also perform full-core calculations, which allows the verification of group constants and nodal codes for criticality calculations. Therefore, my second aim was to develop and verify a group constant generation and parameterization methodology for the ALLEGRO using the Serpent and KIKO3DMG codes.

Neutronics modeling of fast reactors must include thermal expansion of fuel and structural elements due to significant reactivity effects, which are practically negligible in thermal reactors. The reactivity effect of the axial fuel thermal expansion is a defining prompt negative feedback in fast reactors. The main challenge of modeling this phenomenon in nodal neutronics codes is that the axial fuel expansion is radially non-uniform in reality. Accordingly, the third aim of my

PhD was to develop a methodology for modeling the radially non-uniform axial thermal expansion of fuel with the KIKO3DMG and verify the new model on the ALLEGRO reactor by comparison with Serpent Monte Carlo calculations.

After the above-described improvements, the improved KIKO3DMG code coupled with a thermal-hydraulics code, for example, with the ATHLET, can be used to analyze the dynamic behavior of fast reactor cores. Application of a three-dimensional neutronics code instead of the point-kinetics methodology is essential for transients, during which the nuclear power distribution changes significantly. An example is an unintentional control rod withdrawal transient. So the fourth aim of my PhD research was to analyze an unprotected control rod withdrawal transient of the ALLEGRO reactor core using a coupled neutronics/thermal-hydraulics code and identify the limitations of the point-kinetics approach by a comparative analysis.

A reliable calculation tool needs to be validated by measurements, but it is challenging for fast spectrum reactors because only limited experimental results are available. In 2018, the IAEA launched the Neutronics Benchmark of China Experimental Fast Reactor (CEFR) Start-Up Tests Coordinated Research Project [11]. The project provided an excellent opportunity to validate the criticality calculations of the KIKO3DMG code by experimental measurements. Thus, the final aim of the PhD research was to validate the group constant generation methodology and the criticality calculations of the improved KIKO3DMG code on experimental measurements of the CEFR start-up test.

New scientific results

1. I have first-ever performed best estimate plus uncertainty (BEPU) analyses of an unprotected transient overpower (UTOP) and an unprotected loss of flow transient (ULOF) of the ALLEGRO core. For this, I have taken into account the uncertainties of the most important selected core safety parameters originated from modeling uncertainties, and nuclear data uncertainties. I have calculated the power distribution and reactivity coefficients assuming uniform temperature distributions using the Serpent Monte Carlo code and applied the results for thermal-hydraulics analyses using the ATHLET code with point-kinetics methodology. I have found that the peak cladding temperature is sensitive to all the selected uncertain core safety parameters during the ULOF transient. During the UTOP, the Doppler and the fuel thermal expansion reactivity coefficients are the dominant factors. I have shown that, by considering the uncertainties, the peak cladding temperature exceeds the cladding melting point (1300 °C) during the UTOP, initiated by a potential 1$

reactivity insertion. This result adumbrates that additional safety shutdown systems are to be applied to mitigate the consequences of reactivity transients. [P1], [P4], [P5], [P6]

2. I have first-ever performed best estimate plus uncertainty (BEPU) analyses of an unprotected transient overpower (UTOP) and an unprotected loss of flow (ULOF) transient of a 3600 MW_{th} SFR core. For this, I have taken into account the uncertainties of the most important selected core safety parameters originated from modeling uncertainties and nuclear data uncertainties. I have calculated the power distribution and reactivity coefficients assuming uniform temperature distributions using the Serpent Monte Carlo code and applied the results for thermal-hydraulics analyses using the ATHLET code with point-kinetics methodology. I have found that the peak cladding temperature is almost equally sensitive to the power, the coolant plus wrapper temperature coefficient of reactivity, the radial power peaking factor, and the Doppler coefficient during the ULOF transient. During the UTOP, the Doppler coefficient is the most dominant factor. [P1], [P4], [P5], [P8], [P9]

3. I have created a computational model of the ALLEGRO reactor core for coupled neutronics/thermal-hydraulics calculations and developed a practically applicable methodology to flatten the outlet coolant temperature distribution. Using the coupled KIKO3DMG/ATHLET code, I have shown that the difference between the minimum and maximum temperatures of the outlet coolant temperatures can be decreased from 137 °C to 34 °C for the fresh core of ALLEGRO. I have investigated a station-blackout transient of the ALLEGRO and compared the results to RELAP5 calculation results performed by Boris Kvizda. I have found that the ATHLET core model is more conservative considering the peak cladding temperature. The deviation is because the ATHLET model neglects the uncertainly computable effect of the radial heat transfer from the fuel assemblies to the bypass and the structural elements. [P1], [P3], [P6]

4. I have developed a group constant generation and parameterization methodology for the ALLEGRO reactor and verified the method using the Serpent Monte Carlo and KIKO3DMG codes. I have shown that the fuel temperature reactivity coefficient calculated in the KIKO3DMG code differs less than 10% from the reference Serpent results when a cubic fit is applied for the parameterization as a function of fuel pellet temperature. The void and diagrid expansion reactivity coefficients are underestimated in the KIKO3DMG by 12% and 16%, respectively, compared to the Serpent results, when applying a linear fit as a function of coolant density and diagrid temperature for the parameterization. I have found that the

average absolute difference in the relative fuel assembly power between Serpent and KIKO3DMG is only 0.33%. [P2], [P7], [P10]

5. I have developed a new methodology for modeling the axial thermal expansion of fuel with the KIKO3DMG by introducing mixed nodes. Beyond the straightforward radially uniform case, the method is applicable for the much more complex radially non-uniform case. I have verified the new model of the KIKO3DMG on the ALLEGRO reactor by comparison with Serpent Monte Carlo calculations. I have shown that volume-weighted group constants provide correct results in the introduced small mixed nodes. However, it is essential to weight the inverse of the diffusion coefficients instead of the diffusion coefficients, which leads to distorted results. Moreover, I have derived from the one-dimensional diffusion equation that the inverse volume-weighted diffusion coefficient aims to keep the neutron current at the interface of the merged regions. [P2]

6. I have first-ever analyzed an unprotected control rod withdrawal transient of the ALLEGRO reactor core using a coupled neutronics/thermal-hydraulics code, namely the KIKO3DMG/ATHLET. I have shown that the KIKO3DMG/ATHLET overestimated the assembly-wise peak cladding temperature only by 5-13 °C using 6-group constants instead of the widely used 24-group constants. With this, the computation time of the transient is reduced by 95%. Moreover, I have analyzed the transient using the point-kinetics method with fixed power distribution and approximate spatial reactivity effects calculated by the KIKO3DMG without thermal feedback. I have found that this point-kinetics approach underestimates the assembly-wise peak temperatures nearby the withdrawn control assembly up to 23 °C compared to the coupled KIKO3DMG/ATHLET results. [P3]

7. I have validated the criticality calculations of the improved KIKO3DMG code and the developed group constant generation method on experimental measurements of start-up tests of the China Experimental Fast Reactor (CEFR). I have found that the KIKO3DMG overestimates the effective multiplication factor by only 79 pcm in the critical position. The reactivity worths of control rods calculated by the KIKO3DMG are within the standard deviations of measurement values. The void and assembly swap reactivities are calculated within twice the standard deviation of the measurement values. The calculated isothermal temperature reactivity coefficient (-4.06 pcm/K) is within the two measurement values (-3.77 and -4.4 pcm/K).

List of publications

[P1] B. Batki, A. Keresztúri, I. Panka, Calculation of core safety parameters and uncertainty analyses during unprotected transients for the ALLEGRO and a sodium-cooled fast reactor, ***Annals of Nuclear Energy***, Volume 118, 2018, Pages 260-271

[P2] B. Batki, I. Pataki, A. Keresztúri, I. Panka, Extension and application of the KIKO3DMG nodal code for fast reactor core analyses, ***Annals of Nuclear Energy***, Volume 140, 2020, Article 107295

[P3] B. Batki, I. Pataki, A. Keresztúri, I. Panka, Simulation of an unprotected transient of the ALLEGRO reactor using the coupled neutronics/thermal-hydraulics system code KIKO3DMG/ATHLET3.0, ***Annals of Nuclear Energy***, Volume 154, 2021, Article 108086

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