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Multi-physics models for design basis accident analysis of sodium fast reactors. Part I: Validation of three-dimensional TRACE thermal-hydraulics model using Phenix end-of-life experiments

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ABSTRACT

The demonstrated technological feasibility of Sodium-cooled Fast Reactors (SFRs) makes them stand out among the other reactor concepts proposed by Generation IV International Forum (GIF) for short-term deployment. The availability of reliable computational tools in support of safety analyses and plant simulations under complex transient scenarios is essential to assure SFR's compliance with the highest safety goals.

Answering this need, a multi-physics three-dimensional core and system model is being developed to enable a more detailed representation of the physics of the plant and to anticipate more accurately plant behaviour, even under wider three-dimensional scenarios, such as asymmetric transients. The coupling will be performed using the U.S.NRC system codes TRACE-PARCS, modified to simulate more accurately when using sodium as coolant.

The publicly available end-of-life tests conducted in the French SFR Phenix were chosen as baseline to perform a first validation of the computational model. The development of the tool started with a three-dimensional thermal-hydraulic nodal system of Phenix using the TRACE system code.

The system simulates the Phenix end-of-life natural circulation test and the result have been compared with published experimental and benchmark results. The main physical phenomena of the 3 phases of the transient (rise in temperature in the low part of the reactor vessel, establishment of natural convection and subsequent cooling of the lower and upper part of the vessel) are predicted by the developed nodal system. More specifically, the analysis of parameters such as Intermediate Heat Exchangers (IHX), primary pumps and core temperatures, shows that the developed system is able to predict and study natural convection phenomena in Phenix-type reactors.

The three-dimensional nodal system is able to clearly illustrate the existing thermal stratification in the hot pool, which is neglected by one-dimensional systems and enables the modelling of thermal hydraulic asymmetric behaviour, as it is shown by the uneven flow distribution in Phenix's primary IHXs as they are asymmetrically located in the reactor vessel.

1. Introduction

As an energy source and demonstrated value of more than 50 years of commercial power production, nuclear power may play an important role in the future low-carbon energy mix. Recognizing it, several countries are carrying out R&D programs to prepare the deployment of a new generation (Generation-IV) of nuclear advanced systems, that could excel in terms of sustainability, safety and reliability, economics and proliferation resistance and physical protection.

The Generation-IV International Forum (GIF) was launched in 2001 focusing on collaborative R&D programs for selected innovative reactor systems. GIF has selected six reactor concepts for near future large scale implementation, conceived to excel in their reliability, sustainability, safe operation, economic competitiveness and proliferation resistance (Generation-IV International Forum, 2014).

The Sodium-cooled Fast Reactor (SFR) design stands out among the reactor concepts selected by GIF based on its technological feasibility demonstrated in several countries during the last decades. In order to

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assess the compliance of the SFR design with Generation IV safety goals, it is necessary to identify and analyze plant behavior against potential accident scenarios. In this respect, the development and validation of computational tools able to perform reliable plant behavior simulations is essential to the safety case.

To enable a more detailed representation of the physics of the plant and to simulate more accurately plant behavior, even under 3-dimensional scenarios such as asymmetric transients, a doctoral work is aiming at the development and validation of a multi-physics three-dimensional core and system model of a SFR, combined features still not yet validated by current computational tools. This paper describes the first step of the tool development, specifically the development of a three-dimensional thermal-hydraulics nodal system of the French SFR Phenix and its validation against Phenix end-of-life (EOL) natural circulation test. The computational tool used in the present study is an adaptation of the U.S. Nuclear Regulatory Commission (NRC) thermal-hydraulics system code TRACE to simulate more accurately when using sodium as coolant. The second step of the development features the coupling of the U.S. NRC system codes TRACE-PARCS, matter that will be described in a future paper.

1.1. Sodium fast reactors

As the name describes, the SFR operates with a fast neutron spectrum, which not only allows energy production but also breeding, as the conversion rate of fertile material into fissile (^{238}U into ^{239}Pu) can be greater than the fissile consumption rate from the fission chain reaction. Additionally, it also features transmutation capabilities, contributing positively to nuclear waste management. Typically, a SFR operates at a near to atmospheric pressure in the primary system and at temperature of 550 °C, contrasting with the 15 MPa and 350 °C of the PWR operation conditions. This results in producing a higher temperature steam, increasing the energy conversion efficiency.

Throughout the past 50 years, the development of Liquid Metal Fast Reactors (LMFRs) enabled to acquire more than 300 reactor-years of operating experience. The first SFR, the EBR-I, was built in the U.S. and reached criticality in 1951 (International Atomic Energy Agency, 2006). Since then, several fast programs were launched in Asia and Europe (International Atomic Energy Agency, 2006). In Europe, France is the European country that invested the most in SFR development up to now. Its history with SFRs started with the experimental reactor Rapsodie, which started operation in 1967 till 1983 (International Atomic Energy Agency, 2012). This was continued by Phenix prototype, reaching criticality in 1973 and shutdown in 2009 (International Atomic Energy Agency, 2013). Its 35 years of successful operation demonstrated the viability of sodium-cooled fast breeder reactors. In July 2009, Phenix was disconnected from the grid and was prepared for a set of final tests. These tests were projected and carried out as a unique opportunity to learn more about the prototype and contribute to the future development of SFRs. For this reason, Phenix prototype reactor was chosen to be the reference reactor for the model development in this study, validated against some of the Phenix end-of-life tests. The success of Rapsodie and Phenix led to the design and construction of the first large industrial fast breeder worldwide, the Superphenix, having reached criticality in 1985 (International Atomic Energy Agency, 2012). Technical incidents and public opposition dictated the fate of Superphenix that was shut down in 1997, 20 years before the end of the plant's designed lifetime. Nonetheless, Superphenix was still able to demonstrate that large scale fast reactors are possible to operate neutronics wise, as the large thermal inertia limits greatly the reactivity feedback effects. In addition, the European Fast Reactor (EFR) project was launched in 1988 by the European Fast Reactor Utilities Group (EFRUG) (Lefèvre et al., 1996), having the French electric utility company EDF as leader. The EFR design aimed to gather the combined experience of France, Germany and the UK for liquid metal reactor technology based on pool-type reactors. The developed design was based on established technology and with realistic cost estimates (Lillington, 2004). Although its construction is not foreseen in the near future, the project constituted an important step towards the road of the commercial utilization of fast reactors. Presently, France is working on the Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) Project (International Atomic Energy Agency, 2012). The project aims to develop a SFR industrial prototype (ASTRID) expected to serve as a precursor of a first-of-its-kind commercial reactor that can meet GIF's criteria.

Joint European efforts have also been made towards SFR R&D. In 2009 10 different countries engaged in a collaborative project within EURATOM's 7th Framework program for nuclear and research training for the development of a Gen-IV European Sodium-cooled Fast Reactor (ESFR) (CEA, 2003–2011). The project targeted the deployment of ESFR technology by mid-century and finished successfully in 2013. A continuation of ESFR Project was launched in 2017 with the collaborative Project ESFR-SMART (Mikityuk, 26-29 June 2017).

1.1.1. Calculation tools for sodium fast reactors

Many computational tools dedicated to SFR analysis have been developed throughout the years of SFR R&D. Code developments have been focusing on different type of accident analysis, simulating different relevant phenomena. Among the available computational tools, it is possible to distinguish purely thermal-hydraulic codes, purely neutronic codes and coupled thermal-hydraulic/neutronics codes. Additionally, it is possible to distinguish codes of different types, e.g. deterministic versus statistic, the scale of each computational tool, steady state versus transient, etc., and if they are applicable to the full core or to the full reactor. The analysis of Generation-IV SFRs would benefit by the use of multi-physics models, like the coupling between thermal-hydraulic and neutronics computation tools.

The most important experimental and theoretical findings related to sodium have been integrated to state-of-the-art codes, of which SAS4A/SASSYS-1 (Cahalan et al., 2000) and SIMMER-III/IV (Tobita et al., 2006) appear as reference tools. SAS4A and SASSYS-1 computer codes were first developed at ANL for the transient analysis of LMFRs. SAS4A analysis has the objective to quantify severe accident consequences as measured by the generation of energy sufficient to challenge reactor vessel integrity; whereas SASSYS-1 analysis is to quantify accident consequences as measured by the transient behavior of system performance parameters, such as fuel and cladding temperatures, reactivity and cladding strain. In spatial terms, each SAS4A/SASSYS-1 channel models a fuel pin and its associated coolant. These two codes have been additionally coupled to nodal spatial kinetics computer codes (VARIANT-K and DIF3D-K) for accurate analysis of coupled spatial neutron kinetics and thermal hydraulic effects. For Core Disruptive Accidents (CDA), SIMMER has been developed continuously since its first release (Bohl and Luck, 1990). The current code version allows an evaluation of the transition phase of a CDA, through the simulation of the entire core and the modelling of key thermodynamic and neutronics phenomena occurring during the accident progression.

The FAST (Fast-spectrum Advanced Systems for power production and resource managemenT) code system is currently being developed at PSI for static and transient analysis of Design Basis Accidents (DBA) of the main Generation-IV fast-neutron spectrum reactor concepts: sodium, gas and lead-cooled fast reactors. The main goal is to allow an analysis of advanced fast spectrum systems including different coolants and fuel types. The reactor modelling includes an integral representation of the core neutronics, thermal- hydraulics and fuel behavior, and the reactor primary and secondary systems. The code system has been assembled from well-established existing codes, extended to the simulation of fast reactor features when necessary, namely: ERANOS, for static neutronics, PARCS, for reactor kinetics, TRACE, for system thermal-hydraulics, and FRED, for thermal mechanics.

The procedure followed in this study has been based on the adaptation of computational tools verified, validated and extensively used in

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