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Code assessment and modelling for Design Basis Accident Analysis of the European sodium fast reactor design. Part I: System description, modelling and benchmarking

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HIGHLIGHTS

- Ten system-code models of the ESFR were developed in the frame of the CP-ESFR project.
- Eight different thermohydraulic system codes adapted to sodium fast reactor's technology.
- Benchmarking exercise settled to check the consistency of the calculations.
- Upgraded system codes able to simulate the reactivity feedback and key safety parameters.

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ABSTRACT

The new reactor concepts proposed in the Generation IV International Forum (GIF) are conceived to improve the use of natural resources, reduce the amount of high-level radioactive waste and excel in their reliability and safe operation. Among these novel designs sodium fast reactors (SFRs) stand out due to their technological feasibility as demonstrated in several countries during the last decades. As part of the contribution of EURATOM to GIF the CP-ESFR is a collaborative project with the objective, among others, to perform extensive analysis on safety issues involving renewed SFR demonstrator designs. The verification of computational tools able to simulate the plant behaviour under postulated accidental conditions by code-to-code comparison was identified as a key point to ensure reactor safety. In this line, several organizations employed coupled neutronic and thermal-hydraulic system codes able to simulate complex and specific phenomena involving multi-physics studies adapted to this particular fast reactor technology. In the "Introduction" of this paper the framework of this study is discussed, the second section describes the envisaged plant design and the commonly agreed upon modelling guidelines. The third section presents a comparative analysis of the calculations performed by each organisation applying their models and codes to a common agreed transient with the objective to harmonize the models as well as validating the implementation of all relevant physical phenomena in the different system codes.

1. Introduction

Fast reactors are identified in the Generation IV International Forum Roadmap (GIF; Gen and Roadmap, 2002) as a unique, potentially sustainable energy source in terms of waste management, fuel optimisation, economic competitiveness and proliferation

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resistance. Among the proposed designs, Sodium fast reactors (SFR) have the advantage of the extensive technological experience gained during past and current projects as developed in many different countries for nearly 50 years. In particular, the design and operation of several prototypical sodium-cooled fast reactors, such as the Experimental Breeder Reactor (EBR) and the Fast Flux Test Facility (FFTF) in Hanford (USA), the prototype Phénix in France (Sauvage, 2005), the BN-350 in Kazakhstan, and Monju in Japan (Matsuura et al., 2007, Kondo et al., 2013) have accumulated a total of about 400 reactor-years of operational experience in SFR technology. Currently, the sodium-cooled China Experimental Fast Reactor (CEFR) (Xu, 2000) has been connected to the grid in July 2011, while the Russian BN-800 (Saraev et al., 2012) and the Prototype Fast Breeder Reactor (PFBR) in India (Chetal et al., 2006) are both under construction. In Europe the advanced sodium technological reactor for industrial demonstration ASTRID (Le Coz, 2013) is planned as demonstration project within the "European Sustainable Nuclear Industrial Initiative" (ESNII, 2009; European Utility Requirements, 2001). Several European countries are currently active in research programmes for the development of fast reactors innovative (GENIV) concepts.

In order to create a common European framework to support the sodium fast reactor technology and as part of the EURATOM contribution to the GIF, the Collaborative Project on the European Sodium Fast Reactor (CP-ESFR) was conducted within the 7th EURATOM Framework programme (Vasile et al., 2011).

The project merged the contribution of 24 European partners with the objective to establish the technical basis of a European sodium fast reactor plant with enhanced safety performance, resource efficiency and cost-effectiveness.

The project was divided into several work packages (Vasile et al., 2011) covering the different technical aspects of the reactor design. In particular the work package number three (WP3) focused on the safety assessment of the plant and its subsystems. Its main objectives were the definition of an adequate safety approach with safety objectives and principles for the design; the proposal of preliminary assessment of provisions related to the implementation of the whole set of defence in depth levels; studies of representative transients; and scenarios and the evaluation of design strategies to limit the consequences of a partial core disruptive situation.

Innovative nuclear reactor design concepts require specific tools to assess reactor performance and safety. Compared to the traditional water-cooled thermal nuclear reactors the use of sodium as coolant, the fast neutron kinetics, and the pool-type design are some of the design options that call for the development and validation of computational tools able to analyse the safety features of these innovative systems that may affect the global plant safety. A specific task (Task 3.3.1) was established with the objective to develop multi-physics models for the plant design and perform calculations to evaluate the behaviour of the plant during certain postulated transients.

The task group consisted of eight organisations namely, CEA (France), ENEA (Italy), EDF (France), UPM (Spain), PSI (Switzerland), KIT (Germany), NRG (Netherlands) and JRC-IET (European Commission). Each organisation provided its expertise on the use of dedicated system codes adapted for sodium-cooled fast reactor applications. Some computational tools, originally developed for commercial light water nuclear reactor safety analyses, have undergone extensive modifications (coolant material properties, heat transfer correlations and pressure drop models) to be applied for SFR studies.

The approach followed was to set up a benchmarking exercise. First, each organisation amended the modelling of their employed system codes according to the ESFR plant specifics (primary, secondary, tertiary loops). Calculated results of a defined transient were then compared with those of the other partners allowing collective modelling iteration and enhancements in the various codes. In this respect, this benchmark exercise provided the unique opportunity to compare the performance of the current state-of-the-art system codes for applications related to sodium-cooled fast reactor safety analyses.In Section 2 we present the reference European sodium fast reactor design (Ehster, 2009) adopted in the safety studies: an oxide-fuelled, pool-type reactor. The system codes used in the task are shortly described in Section 3 while some common modeling aspects are given in Section 4. In Section 5 we present the main results of the comparison case. Specific aspects and phenomena that required refinement in the modeling in order to achieve higher accuracy in the results are identified, and the main conclusions of the study are outlined. Discussion and conclusions are presented in Section 6.

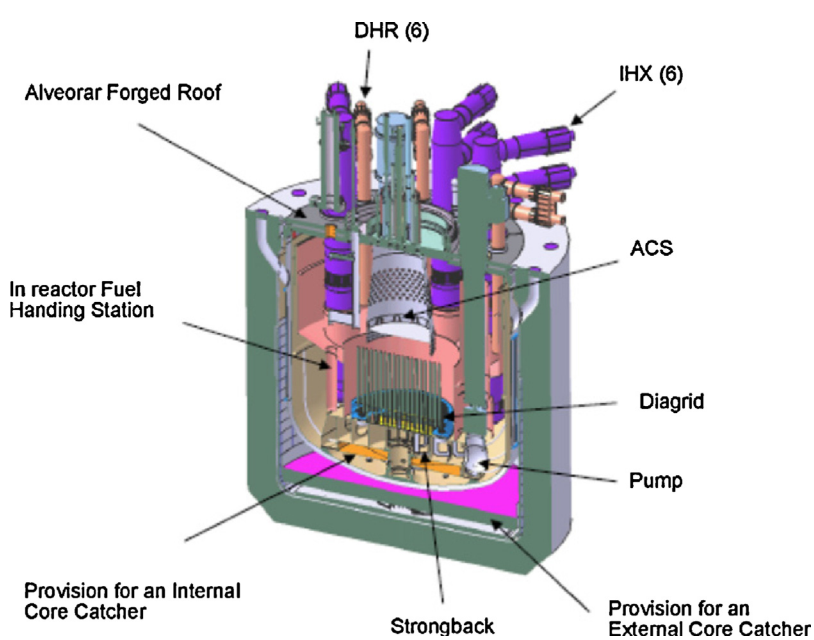

Fig. 1. ESFR pool-type concept (Vasile et al., 2011).