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Detailed neutronic study of the power evolution for the European Sodium Fast Reactor during a positive insertion of reactivity

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HIGHLIGHTS

- This paper studies the effect of an unexpected runway of a control rod in the ESFR.
- The power peaked fuel pin within the core was identified.
- The increase of the fission power density of the fuel pin has been evaluated.
- Radial/axial fission power density of the power peaked fuel pin has been evaluated.

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ABSTRACT

The new reactor concepts proposed in the Generation IV International Forum require the development and validation of new components and new materials. Inside the Collaborative Project on the European Sodium Fast Reactor, several accidental scenario have been studied. Nevertheless, none of them coped with mechanical safety assessment of the fuel cladding under accidental conditions. Among the accidental conditions considered, there is the unprotected transient of overpower (UTOP), due to the insertion, at the end of the first fuel cycle, of a positive reactivity into the reactor core as a consequence of the unexpected runaway of one control rod. The goal of the study was the search for a detailed distribution of the fission power, in the radial and axial directions, within the power peaked fuel pin under the above accidental conditions. Results show that after the control rod ejection an increase from 658 W/cm^3 to 894 W/cm^3, i.e. of some 36%, is expected for the power peaked fuel pin. This information will represent the base to investigate, in a future work, the fuel cladding safety margin.

1. Introduction

The Generation IV International Forum (GIF) Roadmap (Kelly, 2014) identifies fast reactors as an exceptional, potentially sustainable energy source, particularly in terms of waste management and nuclear fuel optimization. Nearly 55 years of technological experience gained from related projects in many countries have placed the Sodium-cooled Fast Reactor (SFR) in a unique position among the different systems promoted by the GIF. Many countries demonstrated significant advancements on SFRs technology not only in terms of design but also in terms of operation. The Experimental Breeder Reactor (EBR) and the Fast Flux Test Facility (FFTF) in USA, the BN series reactors in Russia and the prototype Phénix and commercial SuperPhénix in France have added over 400 reactor-years of operational experience in the SFR technology. Latest examples of SFRs are the recently connected to the grid China Experimental Fast Reactor (CEFR) (Mi, 1999), the Russian BN-800 (Saraev et al., 2010) and the Indian Prototype Fast Breeder Reactor (PFBR) (Chetal et al., 2006). Also in Europe there are research activities on the SFR field. The European Sustainable Nuclear Industrial Initiative (ESNII), under the umbrella of Sustainable Nuclear Energy Technology Platform (SNETP), has planned an industrial project for demonstration purposes called Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID). The present work is part of the Collaborative Project on the European Sodium Fast Reactor (CP-ESFR), which has been initiated as part of the EURATOM FP7 contribution to the GIF and an attempt to create a common European framework to support the SFR technology, establishing the technical basis of a European Sodium Fast Reactor with improved safety performance, resource efficiency and cost efficacy (Fiorini and Vasile, 2011). In particular, the study here presented is focused on the determination of the fission power distribution within the power peaked fuel pin at the

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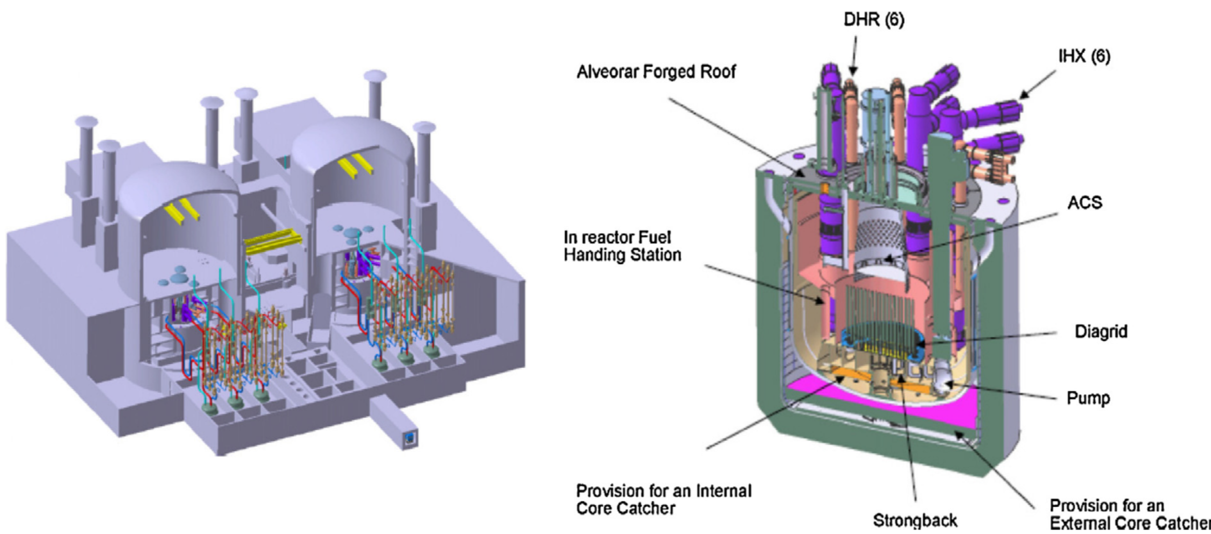

Fig. 1. ESFR plant design and pool type concept for the reactor core (Lazaro et al., 2014).

end of the first fuel cycle (EoC) under accidental conditions. The accidental scenario taken into consideration is the unprotected transient of overpower (UTOP) due to an unexpected runaway of one control rod (a scenario which is also part of the classical design basis accidents). The results of this analysis, part of the preliminary safety analysis of the reactor core, will drive a future work to investigate the pellet-cladding mechanical interaction (PCMI).

The Monte Carlo code MCNP6 (Goorley et al., 2012) has been used for all the calculations here presented. This code differs from its predecessors being the first which integrates all the features of MCNP5 and MCNPX providing, among the others, the capability to perform burnup calculations with the depletion code CINDER90 (Wilson et al., 1995). However, being MCNP6 a steady-state code, the analysis of the UTOP accident will be limited to the instant at which the control rod is ejected from the core. On the other hands, no transient kinetics codes would allow to describe the geometry and perform the neutron transport with the same degree of accuracy achievable with MCNP6. Moreover, the results will be conservative as the neutronic feedback due to the Doppler effect will not be present, not being considered the increase of the fuel temperature during the transient.

2. Core, specifications and MCNP6 model of the European Sodium Fast Reactor

2.1. Core description and specifications

A detailed description of the plant design of the European Sodium Fast Reactor (ESFR) can be found in Fig. 1, designed in Ammirabile and Tsige-Tamirat (2013). The present core design makes use of a fuel based on mixed oxides of uranium and plutonium and it refers to a reactor power of 3600 MW_{th}. The main parameters of the reactor core are reported in Table 1. This section aims at providing the major characteristics of this core design useful for the neutronic modeling purposes.

Fig. 2 represents a horizontal cross section of the core, showing different zones: the inner and outer fuel assembly zones (purple and light blue[1], respectively) and the reflector assembly zone (yellow, to be noted that there is also a reflector assembly in the center of the reactor core). The inner and outer fuel zones are, respectively, made by 225 and 228 assemblies, each one containing 271 fuel pins, and, in order to flatten the core power distribution at the EoC, are characterized by different plutonium mass content (12.80% and 14.90%, respectively) and uranium mass content (75.28% and 73.18%, respectively). The assumed fresh fuel composition for both the inner and outer zone is reported in Table 2. It can be noted that to take into account the beta decay of ^{241}Pu a small fraction of ^{241}Am is also present in the fresh fuel composition.

The fuel assembly consists of a hexagonal wrapper tube, made of a chromium ferritic/martensitic steel (EM10, 9Cr-1Mo), that contains a triangular arrangement of fuel pins with a helical wire wrap spacers to minimize their displacement. The MOX fuel pin is made by pellets with an oxide dispersion strengthened (ODS) steel cladding. Finally, the main characteristics of the fuel assembly are summarized in Table 3.

Fig. 2 shows also the 24 control shutdown devices (CSD) and the 9 diverse shutdown devices (DSD). The CSD rods contain natural boron carbide (~20% of ^{10}B) whereas the DSD rods are made of enriched boron carbide (~90% of ^{10}B). Both the CSD and the DSD rods have a follower which is made of steel and sodium (8% and 92% by weight, respectively). The CSD rods are located, according to a symmetric pattern within the two fuel zones, on two different rings while a single ring of DSD rods is placed between them. The CSD rods are expected to be used during the normal operation to control the long term reactivity changes, the DSD rods are instead

Table 1
ESFR pool-type concept core design specifications (Ammirabile and Tsige-Tamirat, 2013).

ESFR core parameters	
Thermal power	3600 MW_{th}
Volume	17.5 m^3
Lattice pitch	21.08 cm
Fuel type	Pins/Pellets
Active height	1 m
Cladding material	ODS steel
Pin diameter	9.43 mm
Pin per assembly	271
Fuel assemblies	453
Control shutdown devices (CSD)	24
Diverse shutdown devices (DSD)	9
Fraction of delayed neutrons	390 pcm
Core inlet temperature	395 °C
Core outlet temperature	545 °C
Fuel pellet material	$(U, Pu)O_2$

[1] For interpretation of color in Fig. 2, the reader is referred to the web version of this article.