Annals of Nuclear Energy 85 (2015) 115–126

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Space–time effects in the initiating phase of sodium fast reactors and their evaluation using a three-dimensional neutron kinetics model

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ARTICLE INFO

Article history:
Received 29 October 2014
Received in revised form 22 April 2015
Accepted 27 April 2015

Keywords:
Severe accident
Sodium fast reactor
Space–time neutron kinetics

ABSTRACT

The treatment of low probability events leading to core disruption is one of the key issues of R&D plans for the advanced reactor systems in general, and for sodium fast reactors in particular. Regarding sodium fast reactor systems, a major concern is that the core, at nominal, is not necessarily in its neutronically most reactive state. The simulation of the initiating phase of such accidents is of particular interest both for the prevention and the mitigation of routes leading to a large core disruption and recriticalities. Current analysis of the initiating phase relies on a Point Kinetics model, which neglects spatial variations of the neutron flux (*i.e.* the flux shape). Because the core geometry may be modified during such accidental conditions, the constant-shape approximation becomes questionable when applied to initiating phase simulations. The present paper investigates space–time effects during the degradation of a sodium fast reactor core. To this aim, a three-dimensional neutron kinetics model has been coupled to a multiple-channel thermal–hydraulics model and applied to evaluate spatial effects during a severe accident transient. As expected, it was found that spatial effects are not likely to be of importance in the pre-boiling portion of the transient reinforcing the use of a Point Kinetics model for the simulation of design-basis transient up to the boiling point. Departure from the Point Kinetics approach is noticeable after the fuel-pin breakup. The fuel being the source of neutrons, its motion within the core boundaries tends to create important spatial changes in the fission source. From this event, the Point Kinetics model fails in reproducing results obtained by the three-dimensional model and the study conducted in the present paper assesses this discrepancy.

1. Introduction

A key research goal of sodium fast reactors (SFR) is an enhanced safety compared to former nuclear reactor concepts (*i.e.* GEN-II systems and former SFR designs) and thus safety requirements at least identical to GEN-III standards are required. To a large extent the safety considerations will dictate the SFR design characteristics and play a major role in establishing the economic attractiveness of this concept. In particular, the treatment of low probability events leading to core disruption (*i.e.* severe accident conditions) is one of the key issues of R&D plans for the advanced reactor systems in general, and for SFR in particular. Regarding SFR systems, a major concern is that the core, at nominal, is not necessarily in its neutronically most reactive state.

A severe accident scenario is generally broken down into subphases as the initiating phase, the transition phase, the core-material relocation phase and the decay heat-removal phase for applying specific code to each phase. During the initiating phase, the subassembly wrapper tubes keep their mechanical integrity. Material disruption and dispersal is primarily one-dimensional. For this reason, evaluation methodology for the initiating phase relies on a multiple-channel approach. Typically a channel represents an average pin in a subassembly or a group of similar subassemblies. The simulation of the initiating phase of such accidents is of particular interest both for the prevention and the mitigation of routes leading to a large core disruption and recriticalities.

Current analysis of the initiating phase relies on a Point Kinetics (PK) model, which neglects spatial variations of the neutron flux (*i.e.* the flux shape). Because the core geometry may be modified during such accidental conditions, the constant-shape approximation becomes questionable when applied to initiating phase simulations. Also, few or no justification is available in the literature to support the use of the PK approximation. Most advanced attempts to perform a space–time analysis of neutronics effects during initiating phase on SFR are described herein.

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http://dx.doi.org/10.1016/j.anucene.2015.04.033
0306-4549/

Acronyms

PK Point Kinetics
RDV Rendezvous

Neutron diffusion and transport theory capabilities have been added to the SAS4A/SASSYS-1 code (Tentner et al., 1985) and tested for the analysis of thermal reactors (Taiwo et al., 1997) and accelerator driven systems (Cahalan et al., 2000). However, these calculations do not model fuel-pin melting and relocation which may induce local flux shape modifications and corresponding reactivity effects. It should be noted that a coupling between the DIF3D-K spatial kinetics computer code, a two-phase thermal–hydraulics model, and the MARTINS fuel relocation model has been applied to the analysis of space–time effects during a severe accident on a large Heavy Water Reactor (Morris, 1994). The results and conclusions obtained in the paper are typical of what would be expected for any loosely coupled core, and therefore does not apply to fast reactors. As a consequence, a specific space–time analysis of neutronics effects during initiating phase on SFR is needed.

In a preceding thesis defended in 1972 (Bezella, 1972), a multi-group accident analysis model has been developed which approximates in two dimensions the fast reactors spatial effects as well as the major thermal–hydraulics feedback processes including sodium voiding. Based on simple models for two-phase sodium and fuel relocation dynamics and a two-dimensional neutron kinetics code, sodium voiding effects have been calculated on a medium-size SFR. Core design considered in Bezella (1972) presents a strong positive sodium voiding effect leading to prompt critical conditions during the voiding phase. As no fuel-pin degradation model is included in this code, accident results found in Bezella (1972) are unrealistic after prompt-criticality is reached due to the neglection of fuel-pin failures and associated neutronics effects. During the voiding process, spatial effects have been calculated showing a close agreement between 3D and PK model.

To overcome this issue and improve the safety assessment of SFR, the SNATCH spatial neutron kinetics model (Le Tellier et al., 2010) has been extended and coupled to the SIMMER-III fluid dynamics code (Yamano et al., 2003) in a multiple-channel approach (Guyot et al., 2014; Guyot, 2014). The proposed model allows one to perform the evaluation of local effects regarding neutronics and thermal–hydraulics quantities.

The present paper investigates space–time effects during the core degradation. To this aim, a comparison of PK and space–time models is conducted on a large SFR and spatial effects are examined. A description of the reactor as well as the computer model are first provided. Axial and radial divergence factors are then introduced to evaluate the flux distortion during the transient. Finally, numerical results are analyzed including several sensitivity studies and conclusions are formulated.

2. Reactor description

The reactor core used in the present study pertains to a large sodium-cooled oxide-fueled reactor. A core map for a one-third sector is shown in Fig. 1. This core is made of three different coolant flow rate zones and two different enrichment zones (inner and outer cores). A beginning-of-life state (fresh fuel) is selected as the core state of this evaluation (*i.e.* no pre-irradiation phase). Main design parameters of the core are listed in Table 1.

For the selected core with 453 hexagonal fueled subassemblies arranged in a one-third symmetry, a complete description of the reactor requires the modeling of 151 fueled subassemblies. In the following study, a 151 subassembly-to-channel arrangement has been used. In this representation, each channel models a fuel subassembly (3 subassemblies from the view point of reactivity effects due to the one-third symmetry).

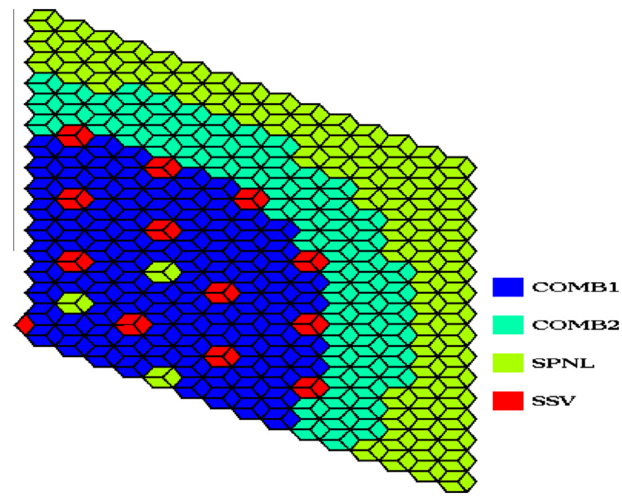

Fig. 1. One-third core sector showing inner core (COMB1), outer core (COMB2), control rod guide tube (SSV) and steel blanket (SPNL) areas.

Table 1
Core design parameters.

Item	Specification
Core power (MW th.)	3600
Inlet/Outlet Temp. (K)	650/800
Core height (m)	1
Number of fuel subassemblies	453
Number of pins/subassembly	271
Total effective delayed neutron fraction	3.69E−03

The initial power distribution calculation has been performed using the SNATCH solver. Six energy groups are employed in the multi-group treatment (see Table A.6 for the complete description of the energy meshing). Microscopic cross sections are based on JEFF2.2 data and processed using the ECCO cell code of the ERANOS distribution (Rimpault, 1995). Fig. 2 shows the spatial distribution of the initial power at the core median plan.

Total reactivity worths of the core are summarized in Table 2.[1] First-order reactivity worths are computed by assuming a complete removal of sodium, cladding or fuel from the core. The sodium and cladding reactivity coefficients[2] of the core show negative values at the upper and lower end of the active core region and in the outer core where a leakage of neutrons prevails. Otherwise, the coefficients show positive values. The value of the fuel Doppler reactivity worth is computed by assuming a variation between steady-state

[1] Reactivity worths are expressed in units of total effective delayed neutron fraction ($).
[2] Reactivity coefficients are expressed in units of pcm per gram.