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Extension of the reactor dynamics code DYN3D to SFR applications – Part II: Validation against the Phenix EOL control rod withdrawal tests

E. Nikitin [a,b,*], E. Fridman [a]

[a] Helmholtz-Zentrum Dresden-Rossendorf, Bautzner Landstraße 400, 01328 Dresden, Germany
[b] Ecole Polytechnique Fédérale de Lausanne, CH-1015 Lausanne, Switzerland

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ABSTRACT

The reactor dynamics code DYN3D, initially developed for LWR applications, is being extended for steady state and transient analyses of Sodium cooled Fast Reactor (SFR) cores. The extension includes the development of the few-group cross section generation methodology, updating of the thermal-hydraulic database with thermal-physical properties of sodium, and development of the thermal-mechanical model to account for thermal expansion effects of the core components.

Part I of the paper provided a detailed description of the recently implemented thermal expansion models able to treat axial expansion of fuel rod and radial expansion of diagrid. The results of the initial verification tests were also presented in Part I of the paper.

The capability of the extended version of DYN3D to perform steady state and transient analyses of SFR cores was validated using selected tests from the end-of-life experiments conducted at the Phenix reactor. Part II of the paper reports on the results of the steady state analysis of the control rod withdrawal tests from the Phenix end-of-life experiments. The transient analysis of the initial stage of the natural circulation test is covered in Part III of the paper.

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1. Introduction

The reactor dynamics code DYN3D (Rohde et al., 2016), initially developed for LWR applications, is being extended for steady-state and transient analyses of Sodium cooled Fast Reactor (SFR) cores. The extension includes the development of the few-group cross section (XS) generation methodology (Fridman and Shwageraus, 2013; Rachamin et al., 2013; Nikitin et al., 2015a, 2015b), the updating of the thermal-hydraulic (TH) database with thermal-physical properties of sodium, and the development of the thermal-mechanical (TM) model to account for thermal expansion effects of the core components.

Part I of the paper (Nikitin et al., 2018a) provided a detailed description of the thermal expansion models that have recently been implemented in DYN3D. The models enable the treatment of important thermal expansion effects occurring within the SFR core, in particular axial expansion of fuel rod and radial expansion of diagrid. The axial expansion model is capable of modeling non-uniform core expansions by using the spatial temperature distribution of the fuel rods. The radial diagrid expansion model utilizes the average inlet sodium temperature to uniformly expand the core in the radial direction. The initial verification study, summarized in Part I of the paper, demonstrated an adequate performance of the newly implemented models.

Part II of the paper focuses on the verification and validation of the extended version of DYN3D against the IAEA benchmark on the control rod (CR) withdrawal tests from the end-of-life (EOL) experiments conducted at the Phenix reactor (IAEA, 2014). The benchmark was designed to assess the capability of neutronic codes, used for SFR analyses, to model deformations in radial power distribution due to the various asymmetric arrangements of CRs. In order to take a full advantage of the available experimental data, the benchmark tasks were solved by DYN3D in two different ways: 1) neutronic solution without feedbacks, and 2) coupled solution with TH and TM feedbacks. The first one was obtained using the fixed core geometry while the dimensions of the fuel assemblies and other core components were explicitly expanded based on the average temperatures provided in the benchmark specifications. In contrast to the previous case, the second solution was obtained by invoking the thermal expansion models to obtain the actual core dimensions based on the temperatures provided by the TH module. In this study, the first solution is used to validate the few-group XS generation methodology and the neutronic performance of DYN3D in general, while the second one served to

* Corresponding author at: Helmholtz-Zentrum Dresden-Rossendorf, Bautzner Landstraße 400, 01328 Dresden, Germany.

E-mail address: e.nikitin@hzdr.de (E. Nikitin).

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demonstrate and assess the overall capabilities of the extended DYN3D version.

The following section provides a brief overview of the CR withdrawal tests. The computational methodology and more detailed modeling assumptions are presented in Section 3. The numerical results obtained with the DYN3D are compared to the experimental data and other codes in Section 4. Section 5 summarizes the results of this part of the paper.

2. Description of the control rod withdrawal tests

The control rod withdrawal tests, documented in (IAEA, 2014), were part of the several EOL tests performed in 2009 at the Phenix reactor. The Phenix EOL core consists of 54 inner and 56 outer MOX fuel assemblies surrounded, first, by 86 blanket assemblies and, secondly, by 252 reflector assemblies on the periphery (see Fig. 1). Furthermore, the core comprises 6 primary CRs, one secondary CR, and 14 reflector-type assemblies inside the core and blanket region as depicted in Fig. 1.

During the control rod withdrawal tests, a series of steady-state measurements were conducted with four different CRs arrangements (hereafter referred to as the CR shift tests). The main goal of the test was to investigate deformations of the radial power shape due to the asymmetric axial positioning of the shifted CRs. At the reference state, all six primary CRs were kept on equivalent level of elevation (so-called “rod bank” position). In three additional steps, CR #1 and #4 shown in Fig. 1 were either withdrawn or inserted relatively to the “rod bank” according to the sequence presented in Fig. 2. The rod bank position was adjusted to keep the reactor at constant power of about 335 MWth. The total sodium flow rate remained constant along all steps.

During the CR shift test, the sodium temperature was measured at the core outlet. Thermocouples were positioned at the head of each assembly located in the first seven assembly rows of the core starting from the core center (120 thermocouples in total). Using these, the sodium heating (temperature difference between inlet and outlet) was measured assembly-wise at each step. The heat transfer between assemblies was assumed to be insignificant. Moreover, the change in the flowrate distribution due to the change of sodium heating was also assumed negligible. Based on these assumptions, the radial distribution of the relative power deviations in respect to the reference state was calculated for each step from the variation of sodium heating (IAEA, 2014):

$$\delta_{rel}P_i = \frac{P_i - P_i^{ref.}}{P_i^{ref.}} = \frac{Qc_P(\Delta T_i - \Delta T_i^{ref.})}{Qc_P\Delta T_i^{ref.}} = \delta_{rel}(\Delta T_i), \quad (1)$$

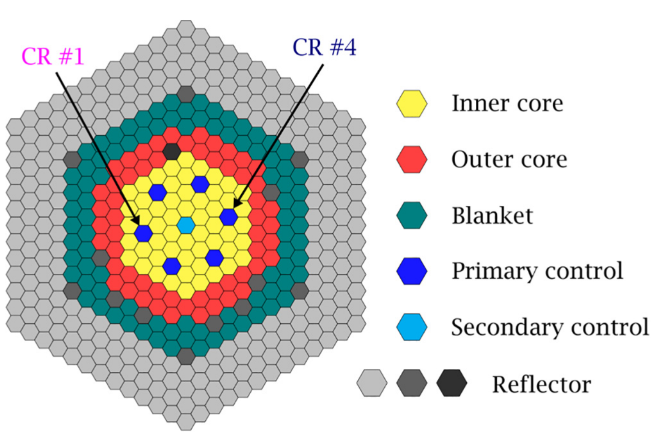

Fig. 1. Phenix EOL core: radial core layout and location of the shifted CRs.

where P is the assembly power, Q is the sodium mass flowrate, c_P is the assembly-average specific heat capacity, and ΔT is the sodium heating in the assembly i. The *Ref.* index denotes the values taken from the reference state. In this study, the deviation in radial power distribution estimated from the temperature measurements for Steps 1–3 were used for the validation of the DYN3D code.

The reactivity worth of the two involved CRs was measured before the CR shift test. The balancing method was used to acquire the integral rod worth over the full range of withdrawal (S-curve) for the CR #1 and #4. The test was initiated at critical low power state (~50 kW), where CR #1 and CR #4 was in complete inserted and withdrawn position, respectively. By using a succession of elementary steps, the CR #4 was moved from bottom to top, while the CR #1 in the opposite direction. As presented in Fig. 3, the CR #4 was firstly moved downwards by a small increment (Step I in Fig. 3) and secondly the CR #1 was withdrawn to compensate the negative change of reactivity (Step II in Fig. 3). Between each elementary CR displacement, the differential worth was measured for each CR based on the application of the inverse kinetics method. When the CR #4 had been totally withdrawn to the top and CR #1 inserted to the bottom (Step N in Fig. 3), the full S-curves were obtained for both rods.

3. Computational methodology

The calculations were done in a two-step approach using the Serpent-DYN3D codes sequence. In the first step, the homogenized few-group XS were generated on lattice level with Serpent, and in the second step, the full core nodal calculations were performed with DYN3D.

3.1. Generation of parametrized cross section libraries

A parametrized XS library was generated for DYN3D that covers the full range of reactor conditions of the CR withdrawal tests.

The XS were calculated with Serpent at different fuel temperatures, coolant temperatures, axial expansion and radial diagrid expansion states. Table 1 presents the selected states that span the parameter space of the XS library. In Table 1, the temperature-dependent expansion coefficients are defined as:

$$\varepsilon(T) = \frac{L(T)}{L(T_0)} = 1 + \alpha(T) \cdot (T - T_0), \quad (2)$$

where L is the linear dimension and α is the linear expansion coefficient corresponding to the temperature T, and T_0 is the reference temperature of the used linear expansion correlation. In this case, the correlation for the temperature-dependent linear expansion coefficients was provided in the benchmark specification.

The coolant density variation is implicitly considered in the coolant temperature variation. In case of the axial expansion effect, the change of dimensions and densities is considered for both fuel and cladding. Furthermore, the radial expansion of the pins, the reduction in the liquid sodium amount between the pins, and the temperature effect of the cladding are taken into account. The axial expansion relies on a closed gap hypothesis and is assumed to be driven by the cladding temperature, which is directly calculated by the internal TH module of DYN3D. When the diagrid is radially expanded, the assembly pitch size is increased (Fig. 4). At the same time, the dimensions of the pins and assembly wrapper remain unchanged, i.e. the increase of sodium gap between assemblies is modeled when the homogenized XS are generated.

In the benchmark specifications, the EOL isotopic densities for the six average core zones were provided. However, atomic densities of the fission products (FPs) were not explicitly given but rather represented by six lumped FPs (also known as pseudo FPs)

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