Group: Transmutation

3-D Simulation of Fuel Assembly Blockage in MYRRHA

The paper is based on the presentation by X.N. Chen et al. at TCADS-4, Fourth International Workshop on Technology and Components of Accelerator-Driven Systems, Antwerp, Belgium, 14-17 October 2019.

Xue-Nong Chen, Andrei Rineiski*

Abstract

In the framework of the KIT and SCK•CEN R&D cooperation and as continuation of earlier studies performed for the EURATOM FP7 MAXSIMA project, accidental transients caused by a single fuel assembly (FA) blockage were simulated with the SIMMER-IV (3-D) code for the MYRRHA core, while assuming no power variation during the accident. A 7-FA model that includes mesh cells for inter-wrapper gaps between FAs was applied, with the blockage of the central FA. Sensitivity analyses on the gap flow rate, fuel chunk jamming fraction, insulator pellet material were performed in order to identify a conservative case that maximises the chance of damage propagation from the blocked FA to the neighbouring ones. All calculations with different options and parameters did show the same sequence in the blocked FA, including melting of pin cladding, fuel pellet failure, small can-wall break-up, steel particle and fuel chunk accumulation leading to additional blockages, and large can-wall break-up. Finally, fuel chunks are swept out from this FA through the inter-wrapper gaps. In the calculations performed for several tens of seconds and longer, no canwall break-up in the neighbouring FAs has been observed. Nevertheless, different options for simulation of the insulator break-up lead to significantly different results in the later phases of calculations. If the insulator pellet breaks up when the cladding is lost, a fuel/steel blockage is formed, which results in a large canwall break-up, but this blockage is dissolved as soon as the upper steel structure melts. If no insulator pellet breaks-up, the fuel/steel blockage is kept in place by the ceramic insulator, which has a very high melting temperature. This observation supports the use of an insulating material with low melting temperature; this option may prevent or reduce blockage of fuel/steel particles coming from failed pins that eventually may prevent or reduce the risk of damage propagation to the intact core.

Keywords

Reactor safety study, LBE cooled reactor, MYRRHA, SIMMER code, 3-D simulations, Fuel assembly blockage accident

Introduction

The MYRRHA reactor designed and developed by the Belgian Nuclear Research Centre (SCK-CEN) was investigated in the European 7th framework project MAXSIMA [1] and earlier projects. The present work is dedicated to analyses of fuel assembly (FA) blockage accidents in MYRRHA studied at KIT in cooperation with SCK-CEN.

SIMMER-III/IV is a two-/three-dimensional, multi-velocity-field, multi-phase, multicomponent, Eulerian, fluid-dynamics code system coupled with a structure model including fuel pins, hexcans, etc., and a space-, time- and energy-dependent transport theory neutron dynamics model [2, 3]. The overall fluid-dynamics solution algorithm is based on a time-factorization approach, in which intra-cell interfacial

area source terms, heat and mass transfers, and the momentum exchange functions are determined separately from inter-cell fluid convection. In addition, an analytical equation-of-state (EOS) model is available to close and complete the fluid-dynamics conservation equations. The code was originally developed and applied for severe accident analyses of fast sodium cooled reactors. However, the philosophy behind the SIMMER development was to generate a versatile and flexible deterministic tool, applicable for the safety analysis of various reactor types with different neutron spectra and coolants, up to the new accelerator driven systems for waste transmutation. In particular, SIMMER was extended at KIT to studies of heavy-metal-cooled, gas-cooled and molten-salt reactors.

The SIMMER IV code, validated by KIT experimental KIT [6], see Fig. 1, was applied for the described studies.

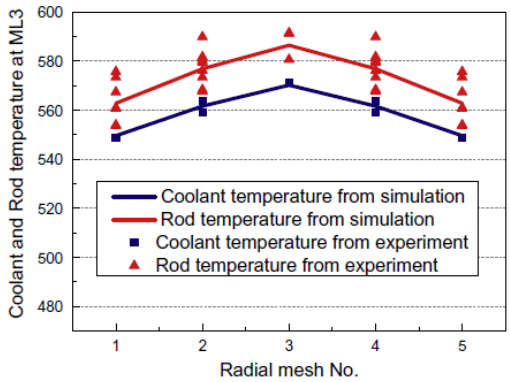

Figure 1: Comparison of the coolant and rod temperatures from SIMMER-IV simulation and experiment [6].

A geometrical model for the chosen 7-FA configuration of a recent design version of the MYRRHA critical reactor has been set up, where the interwrapper gaps have been explicitly taken into account. Central FA blockage accidents are studied, where the configuration of central blocked FA surrounded by other 6 FAs are asymmetric. Results are presented and analysed in this paper.

SIMMER-IV modeling for the 7-FA conficuration

Geometrical Modeling

The coolant flow through gaps (inter-wrapper flow) between FAs cools the FA canwall and therefore can delay or even prevent the FA canwall failure and its propagation. As SIMMER was developed initially for simulations of core disruptive accidents (CDAs) in sodium fast reactor and in simulations of such accidents explicit – i.e. via allocation of special meshes in plane - treatment of gaps between FAs is obviously not important, the the mainly used modelling approach with SIMMER was not to not consider gaps explicitly. However when the SIMMER code is applied to other scenarios than CDAs, the modelling of gaps may reduce the modelling uncertainties. In the past, the gaps were modelled explicitly in a 2D case providing quite different simulation results as compared to the implicit option [4, 5]. Moreover the gaps were explicitly modelled also in 3D with SIMMER-IV for an experimenat case [6]. Due use of the XYZ-geomtery in SIMMER-IV, the hexcan geometry is transformed into a rectangle cubic geometry. Taking 7 FAs as an example, the hexcan geometry could be converted into a rectangular one, as shown in Fig. 2. Each FA has to be divided into 2 parts with suitable parameters. The upper part ‘‘(p)” contains left, right, and back canwall. The lower part ‘‘(q)” contains left, right, and the front canwall. Additionally the gaps between the subassemblies need to be explicitly modelled as part ‘‘(r)”. As long as the explicit meshes are set up in the computational system, SIMMER code treats the flow in the gaps with the hydraulic diameter based on the actual geometry, the heat transfer between the canwall (steel) and the flow (LBE) is calculated with the Nussult numbers based on empirical heat transfer correlations.

Power Amplitude and Distributions

The MYRRHA critical core has 108 FAs. 7 FAs have been selected for the study. The selected

7 FAs are numbered and their power peaking factors are indicated in Fig. 3. The selection of the central blocked FA is representative for a heterogeneous combination of subassemblies, with a maximum risk of a local failure of the canwall of the blocked FA. In this case the possible following fuel sweep-out is anisotropic with a high probability of damaging the neighboring FAs. The FA-wise and axial power distributions are imposed on each single assembly according to the reference results [7]. The power is assumed to be constant during the blockage transients. The thermal boundary conditions outside the system are adiabatic.

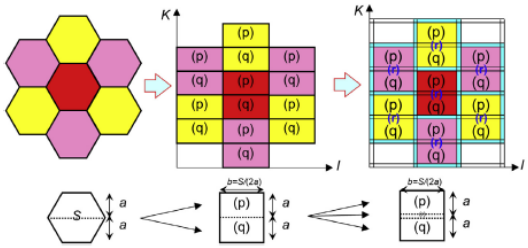

Figure 2: Explicit meshes for gaps flow between subassemblies using SIMMER-IV code.

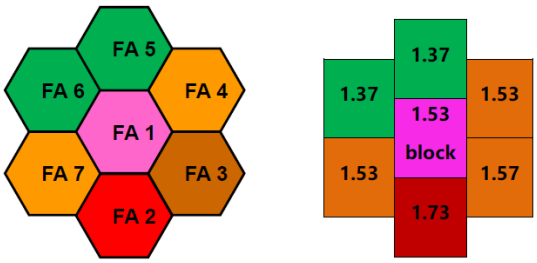

Figure 3: 7 fuel assembly arrangement with power peaking factors

The steady states at the BOC and EOC conditions are quite similar, where only the peaking factors at BOC are slightly higher than those at EOC, mainly due to different axial positions of CRs. Therefore, to be conservative, the BOC condition was used for steady state and transient analyses.

Key steady-state thermohydraulic parameters calculated by SIMMER-IV are calculated for every FA, which match the design values [8]. The coolant inlet temperature is 270 °C. The average thermal power per FA is 0.9259 MW. The pressure boundary conditions are set up and the orifice coefficients for the seven FAs and gaps (in the FA upper position) are adjusted, so that certain coolant flow rates (velocities) are achieved in FAs and gaps respectively. It is noticed that the flows in the gaps are heated up by the can wall.

FA Blockage Transients

FA blockage conditions and parametric studies

The FA blockage is a postulated envelope scenario that evolves from local pin failures which propagate at a FA scale. The blockage is simulated by a reduction of the FA inlet flow rate. Its flow rate is reduced from 72 kg/s to 3.2 kg/s (i.e. 95.6% blockage) for FA1, the central FA, as it is more realistic to assume a porous blockage, allowing a residual cooling flow. The fuel particle diameter is 5.6 mm. For these calculations the fission gas blowout model is deactivated, because we assume, the fission gases contained in the pin gap are all released before the blockage extends to the whole FA flow area.

Different parameters, such as gap velocity, insulator material, particle packing fraction limit in the particle jamming model, and other breakup model options, were employed, some results are given in the following. The reduction of the gap velocity has no signicant effects on the clad melting, but it causes a numerical instability problem. The simulations with different insulator materials as steel and B4C give very similar results. The increase of the particle packing fraction limit shows no significant differences in the calculated results except a slightly earlier canwall breakup and larger particle blockage coefficients.

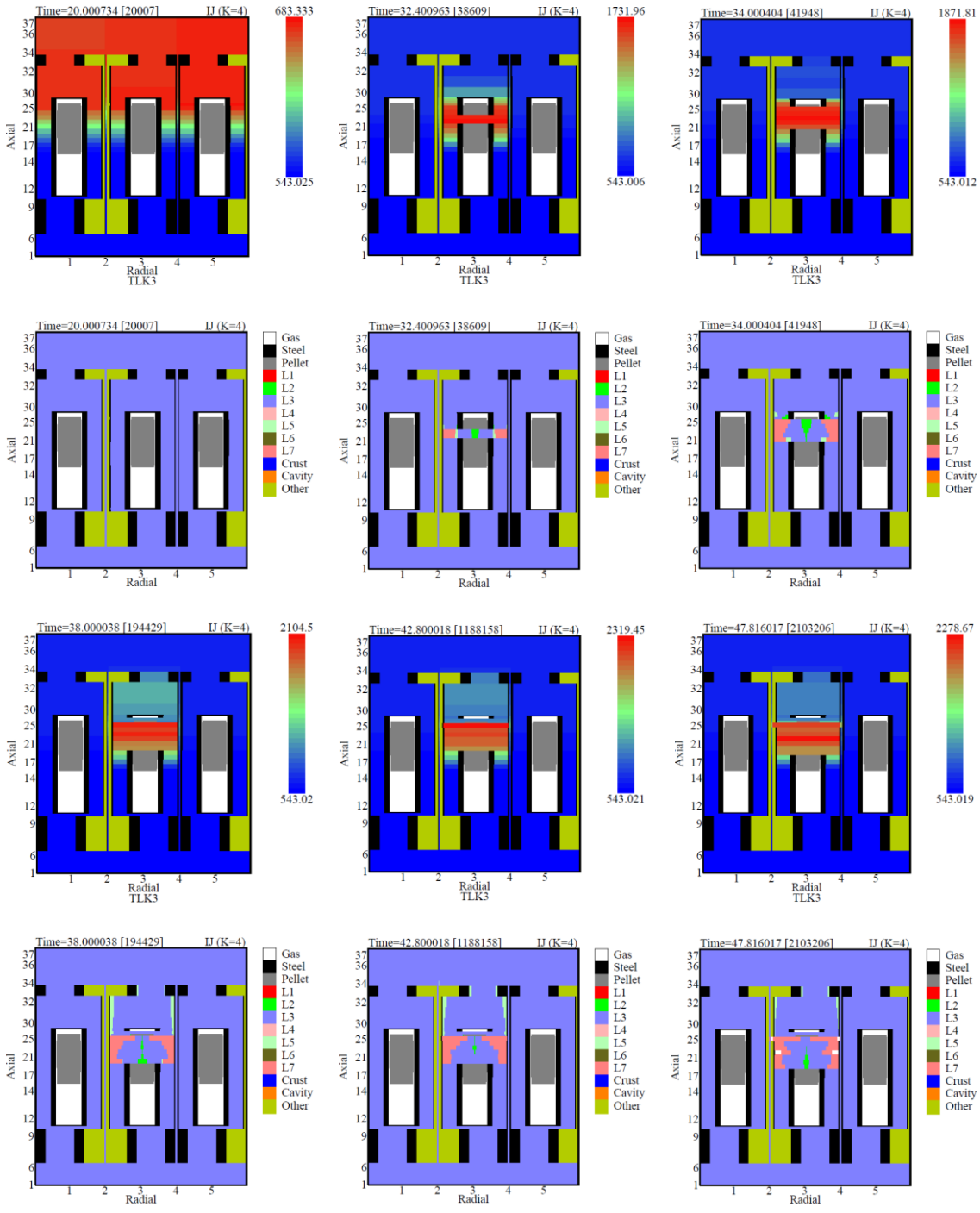

Figure 4: Fuel pin breaks up and fuel chunk blockage: coolant temperature TLK3 and material distribution, where L1 liquid fuel, L2 liquid steel, L3 liquid sodium, L4 fuel particle, L5, steel particle, L6 control particle, L7 fuel chunks.

FA blockage transient results in a typical case

In this paper we just present the calculated blockage results in a typical case**,** i.e. gap velocity = 0.22 m/s, insulator material is B4C, the particle packing fraction limit APJ = 0.8.

Fig. 4 shows the fuel pin breaks up, coolant temperature (TLK3) and material distributions after 20 seconds, the starting time of the blockage. As long as the blockage occurs, the coolant temperature reaches about 1000 °C after less than 10 s. As the cladding temperature reaches the steel melting point limit (1430 °C), the cladding starts to melt. The fuel dispersion starts when the cladding is broken up after 12.4 seconds of instantaneous blockage. A large fuel chunk blockage takes place then aflter 14 seconds at the upper part of the active core. Then the canwall starts to break up after 16.2 seconds. The fuel chunk blockage is not dissolved during the transient calculated. The canwall breakup takes place only at the middle small FA cells. Afterwards, the fuel in this FA escapes from this break-up through the gap to the upper part of the core. No neighboring canwall break-up takes place, i.e. the neighboring FAs stay intact during the transient. Nevertheless, additional blockages in the interwrapper space cannot be excluded, as they cannot be detected with the current model. It is noticed that, if the blockage is a little bit smaller, the canwall in the blocked FA would not break up, thanks to the cooling provided by the flow in gaps, as reported in [9]. This means that the gap flow can prevent the canwall break-up and its break-up propagation.

Conclusions

Calculations with different options and parameters show the same sequences, i.e. the pin melting, the small canwall break-up, fuel/steel particle accumulation and its blockage and the canwall break-up in the blocked FA and finally fuel particle swept out from the blockage FA through the inter-wrapper gaps. The conservative calculations show there is no canwall break-up in the neighboring FAs, even if additional studies need to be carried out to exclude the formation of fuel blockages in the interwrapper space. The variation of the maximum packing fraction APJ shows that it has only slight effects on the later sequences of particle blockage, which does not change major results and conclusion. The variation of the interwrapper gap velocity shows that it has some effects on the clad melting time and later sequences.

References

[1] Baeten, P.; “MYRRHA—Multipurpose hYbrid Research Reactor for High-tech Applications,” SEARCH/MAXSIMA 2014 International Workshop, Location, Karlsruhe, Germany, 7-10 Oct. 2014.

[2] Kondo, S.; Tobita, Y.; Morita, K.; Shirakawa, N.; “SIMMER-III: an advanced computer program for LMFBR severe accident analysis,” Proceedings of the International Conference on Design and Safety of Advanced Nuclear Power Plant (ANP’92), vol. IV, Tokyo, Japan, Oct. 25–29, 1992, pp. 40.5.1–40.5.11 (1992).

[3] Kondo, S.; Yamano, H.; Tobita, Y.; Fujita, S.; Morita, K.; “SIMMER-IV: a Three-Dimensional Computer Program for LMFR Core Disruptive Accident Analysis,” Japan Nuclear Cycle, Development Institute, JNCTN9400 (2000).

[4] Suzuki, T.; Chen, X.-N.; Rineiski, A.; Maschek, W.; “Transient analyses for accelerator driven system PDS-XADS using the extended SIMMER-III code,” Nuclear Engineering and Design 235, pp. 2594-2611 (2005).

[5] Li, R.; Chen, X.-N.; Rineiski, A.; Moreau, V.; “Studies of fuel dispersion after pin failure: analysis of assumed blockage accidents for the MYRRHA–FASTEF Critical Core,” Annals of Nuclear Energy 79, pp. 31-42 (2015).

[6] Li, R.; Chen, X.-N.; Andriolo, L.; Rineiski, A.; “3D numerical study of LBE-cooled fuel assembly inMYRRHA using SIMMER-IV code,” Annals of Nuclear Energy 104, pp. 42-52 (2017).

[7] Sarotto, M.; et al., “The MYRRHA-FASTEF cores design for critical and sub-critical operational modes (EU FP7 Central Design Team project),” Nuclear Engineering and Design 265, pp. 184-200 (2013).

[8] Castelliti, D.; “Comparison of MYRRHA RELAP5 mod 3.3 and RELAP5-3D models on steady state and PLOF transient,” Proceedings of IRUG 2013 Meeting, Idaho, USA, 12-13 Sept. 2013.

[9] Chen, X.-N.; et al., “Recent MYRRHA safety studies with the SIMMER code,” NUTHOS-11, Gyeongju, Korea, October 9-13, 2016, Paper N11A0234 (2016).