Benchmark Between EDF And IPPE On The Behavior Of Low Sodium Void Reactivity Effect Sodium Fast Reactor During An Unprotected Loss Of Flow

D.Lemasson[1], F.Barjot[1], K.Raskach[2], A.Volkov[2]

[1]Electricite De France (EDF), Saclay, France

[2]Institute of Physics and Power Engineering, (IPPE), Obninsk, Russian Federation

david.lemasson@edf.fr

Abstract. The validation of severe accident analysis codes for Sodium Fast Reactors (SFR) is a difficult task as it is not possible to carry out full scale integral experiments. Therefore, in addition to the validation of specific models with dedicated experiments, it is of the utmost importance to increase the confidence we have in these codes by performing benchmarking exercises with independent codes and by independent teams. As EDF R&D and IPPE are both interested in the analysis of the behavior of low Sodium Void Reactivity Effect (SVRE) cores during severe accidents, whether to support R&D on the ASTRID project (conducted by CEA) or to support R&D on the BN family reactors, a benchmarking exercise has been launched in this purpose.

A low SVRE core design has been developed. Its main neutronics properties related to severe accident behavior - sodium density and void effect and fuel Doppler effect - have been evaluated with the CEA and IPPE codes. Finally, an Unprotected Loss Of Flow (ULOF) accident has been simulated by each partner. On EDF side, the SIMMER code has been used with the CEA support whereas IPPE performed its calculations with its code COREMELT. In this article main results concerning power evolution and sodium boiling are presented and compared.

Key Words: SFR, severe accident, boiling stabilization.

1. Introduction

Sodium Fast Reactor technology is one of the most mature among GenIV concepts. Industrial demonstrators have been constructed, operated, and new ones are planned. France and the Russian Federation have already operated such reactors (Superphénix, BN-600 and lately BN-800) and expect to construct new ones (ASTRID developed by CEA in France, BN-1200 developed by OKBM in the Russian Federation). Significant efforts have been granted in both countries to increase the safety of SFR compared to designs developed in the 1980s. Sodium void reactivity effect (SVRE) is an indicator of core performance in dealing with ULOF (Unprotected Loss Of Flow) transients and recent designs have been optimized to reach low SVRE. Severe accident simulations are necessary to confirm the safety improvement suggested by the optimisation of this indicator. Results published by different organisations, with different codes on different low SVRE reactors give a large variety of behaviours. A benchmark has been run by EDF, with SIMMER code system, and IPPE, with the COREMELT code, in order to understand where those differences come from.

2. Description of codes

2.1. COREMELT

The integral code COREMELT [1] was developed for coupled thermohydraulics, neutronics and thermomechanics calculations. It is used for modelling severe accidents (ULOF, UTOP, TIB, etc.) for sodium cooled fast reactors. It can simulate sodium boiling, damage and melting of the fuel rod clads, fuel melting and displacement, thermal interaction between fuel and sodium, freezing of melted steel and fuel. By the moment, the simulation of CDA accidents can be performed up to boiling temperatures of fuel and steel. COREMELT has been used in safety analyses of all Russian projects of sodium cooled fast reactors: BN-600, BN-800, BN-1200 and MBIR.

The thermohydraulics model of COREMELT includes:

- The 4-velocity model of the multi-component multi-phase flow in the cylindrical R-Z geometry based on the porous body approximation;
- 1D or 2D structural models of the core and reactor elements, where the change of the geometry owing to their melting is taken into account;
- The point models of frozen steel and fuel adjusted to the main calculation mesh.

The thermohydraulics part of COREMELT uses the 2D cylindrical geometry (3D version is being currently developed) and calculates two-dimensional distributions of velocity, pressure, internal energy, and volume fractions of four components: sodium (liquid and vapor fractions), particles of melted steel and fuel. Groups of core subassemblies are represented as cylindrical “thermohydraulics channels”. The calculation model includes basic elements of primary circuit such as heat exchangers and pumps.

The neutronics part of COREMELT has several options: 2D (R-Z) and 3D (hexagonal-Z and triangular-Z) transient diffusion options and a 2D quasi-static transport option based on the PnSn-approximation (3D transport option based on the PnSn-approximation has been developed for the next version of COREMELT with 3D thermohydraulics). The most often used neutronics option was the 3D diffusion option. Yet, the calculation results by COREMELT presented in this paper were obtained with the 2D transport option in order to be compared with the calculation results by SIMMER-III, where a similar neutronics model is used. These calculations were performed with the 21-group neutron constants processed by the neutron constant processing code CONSYST [2] on the base of the ABBN93 library [3]. The P_1S_8 approximation is used.

The thermomechanics part of COREMELT calculates stresses and deformations of fuel rods, simulates fuel behavior under thermal and radiation conditions given, initial stages of fuel rod degradation, gas release, etc.

2.2. SIMMER code system

The SIMMER code system (Sn Implicit Multifield Multicomponent Eulerian Recriticality) is specialized in the study of the secondary phase of accidents such as unprotected loss of flow or transient of power. Its range of application has been extended lately to the whole transients, including the primary phase [4]. SIMMER is an Eulerian, 2D/3D multi-velocity-field, multi-phase, multi-component, fluid-dynamics code [5][6]. It is coupled with a fuel-pin model and a space and energy dependent neutron kinetics model.

The code can be divided into three main parts (see Fig. 1):

- Fluid dynamics: intra-cell transfer and inter-cell convection.
- Pin structure: heat transfer in the pin.
- Neutronics: reactivity calculation, flux calculation and associated power distribution calculation.

The main part of the code is fluid-dynamics. It exchanges heat and mass at structure surfaces with the structure model. The neutronics model gives nuclear heat sources to the other models.

One average pin stands for all the pins in the average sub-assembly of a core region.

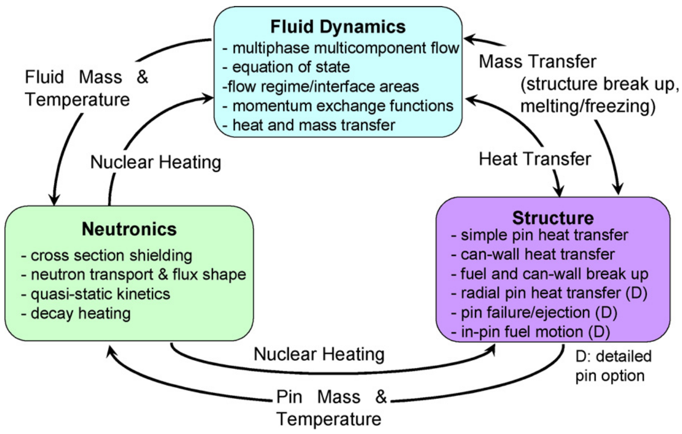

Fig. 1. Overall framework of the SIMMER code [5]

SIMMER can calculate the radial motion of the materials once the hexcans have broken up, and deals with the movements of materials in a molten pool (e.g. sloshing).

Concerning the neutronics, the SIMMER code system features the quasi-static method.

It uses a homogeneous cell calculation, at a given number of neutron energy groups and number of delayed neutron groups. The Boltzmann equation is solved with a Sn transport code.

SIMMER code exists in a 2D (SIMMER-III) and in a 3D (SIMMER-IV) version. In this paper, 2D (R-Z) calculations are performed with SIMMER-III (release 3.E). Fluids (liquid sodium, liquid fuel, liquid clad, gaz, fuel particles, clad particles, absorber material particles and fuel chunks) are assigned to 5 velocity fields.

P_1S_4 approximation is used and the neutron flux is discretized in 16 energy-groups.

3. Description and basic hypotheses of the benchmark. Steady-State Calculations

The benchmark reactor is a hypothetical sodium cooled fast reactor with MOX fuel with thermal power 1500 MW. The core of the reactor consists of two zones with different Pu contents, surrounded by a steel reflector (see Fig. 2). The core is made of hexagonal subassemblies, with a subassembly pitch equal to 19.3 cm. In the axial direction fuel subassemblies consist of a lower expansion zone, a lower fertile blanket, a fissile part, a sodium plenum and an upper shielding. Some basic characteristics of the reactor are presented in Table 1.

The thermohydraulics model of the benchmark reactor is shown in Fig. 3. The core region is simulated by 17 thermohydraulics channels (a description of the channels is presented in Table 2). The primary circuit is represented by 45 mesh points in the radial direction and by

58 mesh points in the axial direction in COREMELT, and by 50 radial mesh points and 58 axial mesh points in SIMMER.

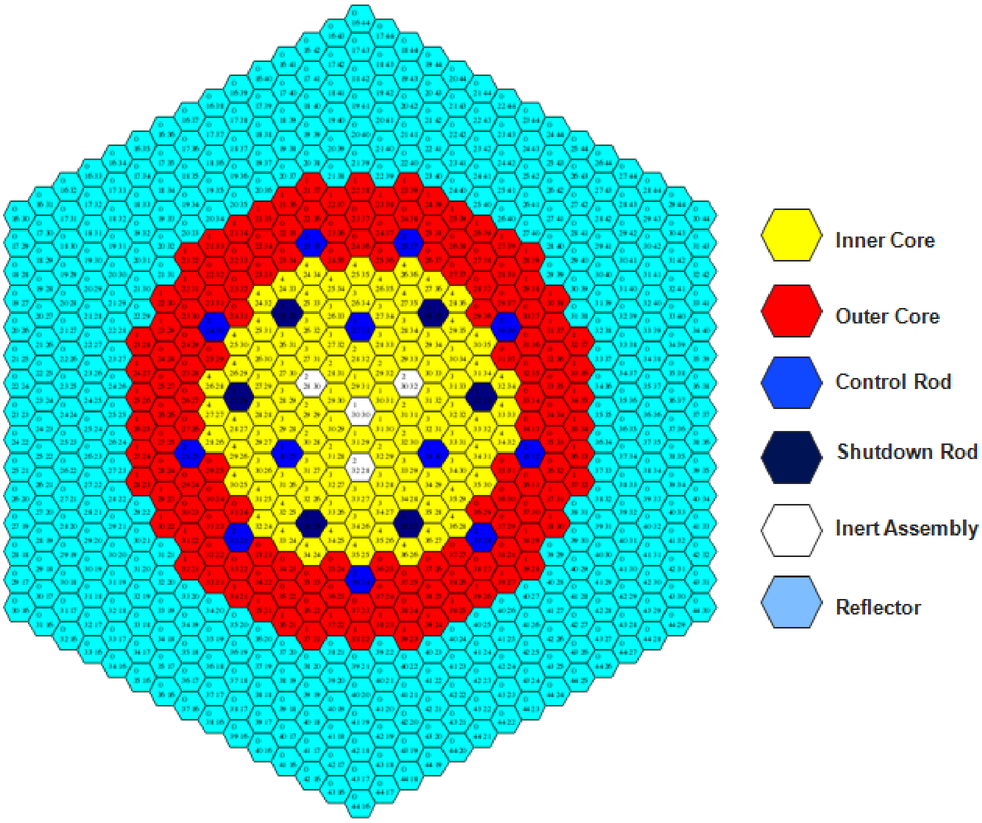

Fig. 2. Layout of the Benchmark Reactor Core

Table 1. Basic Characteristics of the Benchmark Reactor

Data	Values	Units
General Characteristics		
Total Power	1500	MWth
T inlet	395	°C
T outlet (fissile part)	545	°C
Core Zones for Pu content, number of SA	2 zones 96/135	
CSD rods	12	
DSD rods	6	
Inert assembly	4	
Fuel		
Fuel type	Oxide core	
Burn-up status	EOEC[1]	
Porosity	95.5	%(TD)
Pu content		%vol
Cycle		
Frequency	4	
Cycle length	350	EFPD

Rods position at EOEC	Top of the upper expansion zone	

[1]EOEC – End of Equilibrium Cycle

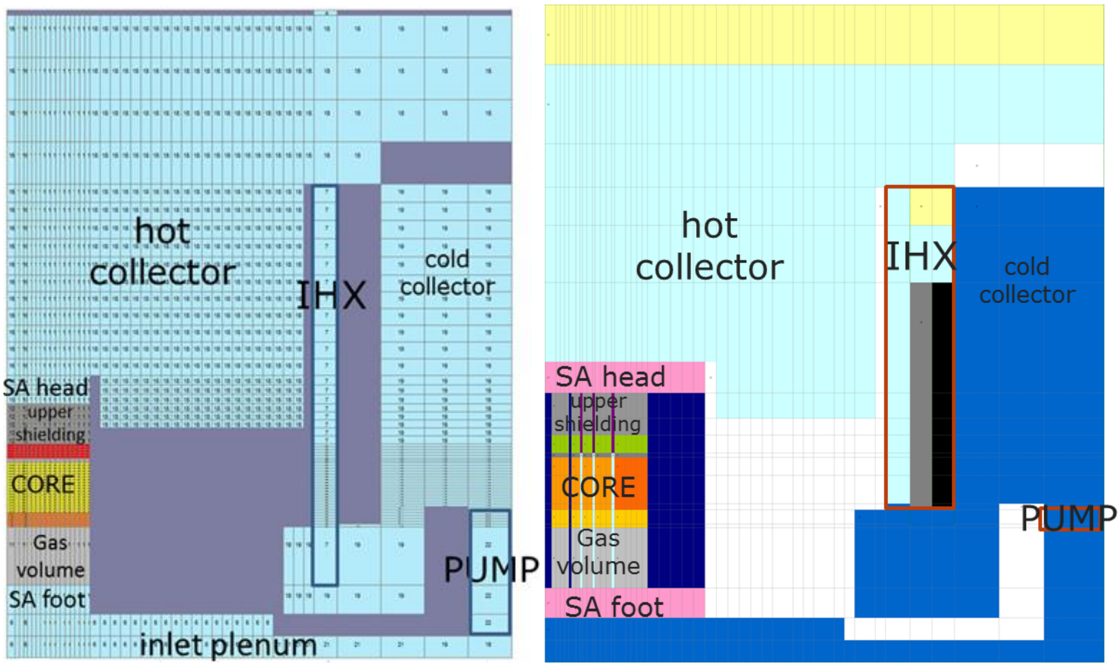

Fig. 3. Thermohydraulics Model (left: COREMELT; right: SIMMER)

Table 2. Description of Thermohydraulics Channels (COREMELT / SIMMER)

Channel Number	**Number of SAs**	**The average flow rate in the channel, kg/s**	**Power per pin, W**	**Sodium heating, °C**
2 / 2	2	38.72 / 38.73	35010 / 35129	154 / 155
3 / 3	3	38.72 / 38.74	34609 / 34700	153 / 153
4 / 4	4	38.72 / 38.74	34615 / 34762	153 / 153
5 / 5	5	38.72 / 38.61	34649 / 34556	153 / 153
6 / 7	7	38.72 / 38.74	34537 / 34386	152 / 151
7 / 8	8	38.72 / 38.57	34455 / 34575	152 / 153
8 / 10	10	38.72 / 38.73	34376 / 34453	152 / 152
9 / 11	12	38.72 / 38.53	34283 / 34383	151 / 152
10 / 13	14	38.72 / 38.72	34060 / 34104	150 / 150
11 / 14	15	38.72 / 38.71	33702 / 33629	149 / 148
12 / 15	16	38.72 / 38.42	33257 / 33301	147 / 148
13 / 17	19	38.72 / 38.68	34604 / 34684	153 / 153
14 / 18	20	38.72 / 38.67	32857 / 32860	145 / 145
15 / 19	21	38.72 / 38.60	30177 / 30239	133 / 134
16 / 20	23	30.23 / 29.91	26454 / 26643	149 / 152
17 / 21	25	24.18 / 23.84	21815 / 22177	154 / 159
18 / 22	27	20.78 / 20.34	17300 / 17905	142 / 150

Steady-state calculations were performed within the framework of the following assumptions:

- Fuel pellet divided in 11 rings;
- Blasius correlation for friction pressure drop coefficient $f=0.316Re^{-0.25}$;
- Fuel-conductivity – Philipponneau correlation [7]:

$$\lambda = \left(\frac{1}{1{,}320\sqrt{x+0{,}0093} - 0{,}091 + 0{,}038B + 2{,}493 \cdot 10^{-4}T + 88{,}4 \cdot 10^{-12}T^3}\right) \times \frac{1-p}{1+2p}$$

- Gas (Argon) volume in the upper chamber of the reactor: 220 m^3;
- Gas pressure in upper chamber of the reactor: 10^5 Pa;
- Pump thrust for nominal conditions: $4.956 \cdot 10^5$ Pa;
- Sodium flow rate through calculation channels (see Table 2) was adjusted by varying hydraulic resistance of the channel inlet. Fuel-clad gap conductance dependent on linear power q (W/cm) [8]:

$$hgap = min\left[3\left(1000 - q + \left(\frac{q}{10}\right)^2 + \left(\frac{q}{100}\right)^3\right), 23000\right]$$

Thermal conductance of the gap between fuel and clad of the fuel pins is presented in Fig. 4 for the hottest channel.

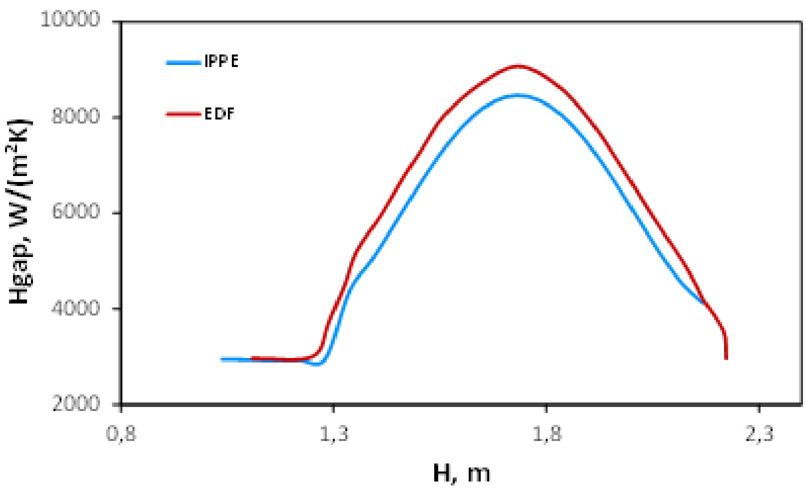

Fig. 4. Axial profile of fuel-clad gap thermal conducance

The power distribution and temperature profiles by SIMMER-III and COREMELT are presented in Figs. 5 and 6. As can be seen, a very good agreement between the two codes is achieved.

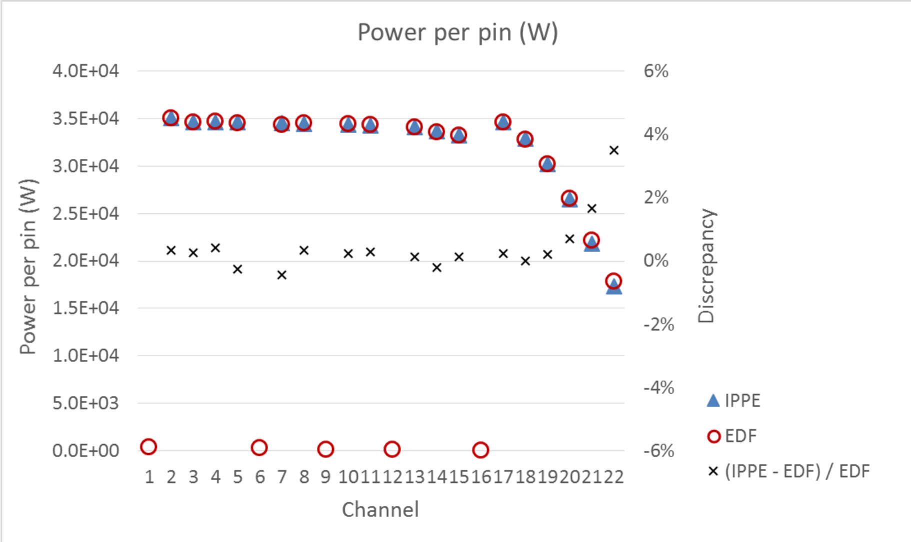

Fig. 5. Radial Power Distribution by COREMELT and SIMMER-III at the Beginning of Transient (t=0 s)

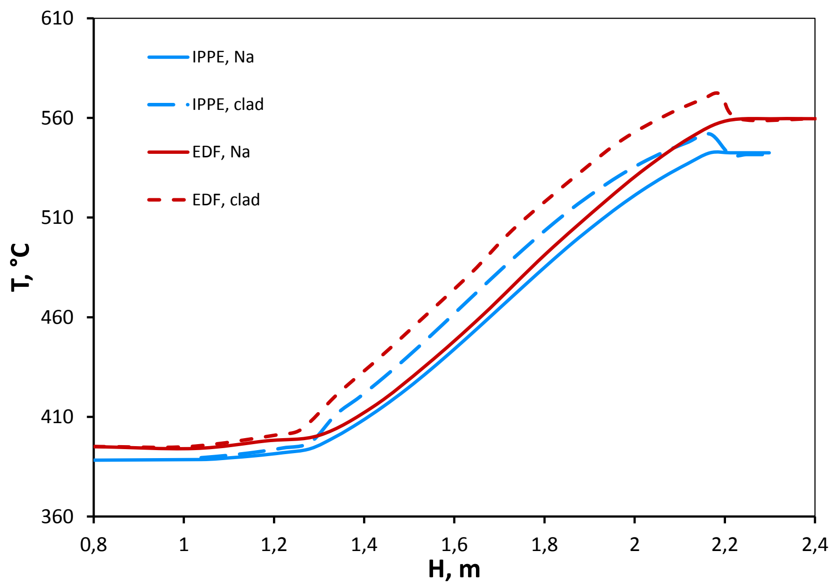

Fig. 6. Axial Temperature Profiles

- Initial data for IHX:

  - Number of IHX pipes: 7000;

  - Inner diameter of pipe: 14 mm, thickness: 1 mm;

  - Length of pipe: 7.2 m;

- The secondary circuit flow rate was adjusted to satisfy boundary conditions for IHX (see Fig. 7).

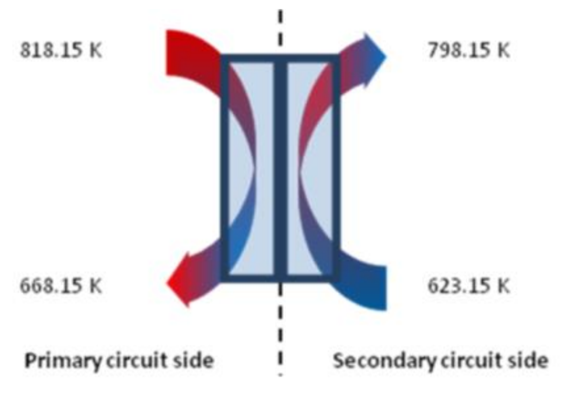

Fig. 7. Sodium Temperature in IHX

A good agreement has been achieved for the steady-state and major parameters influencing the transient behavior have been carefully chosen and shared in order to decrease as much as possible the sources of discrepancies for the transient.

4. Results of Transient Calculations

The scenario of the ULOF accident:

- Up to the moment of the transient initiation the reactor runs with nominal thermal power at the end of equilibrium cycle (all control rods are in upper position);
- At the beginning of transient electricity supply fails and failure of all systems of reactivity control is also postulated (safety rods do not move into the core).
- The pump coast down is simulated by imposing the evolution of the pump head (H is the pump head, t is the time and τ is the mass-flow halving time, here 10s):

$$H = H(t = 0) \times \left(\frac{1}{1 + {}^{t}/_{\tau}}\right)^2$$

Figs. 8 and 9 give change of the basic reactor characteristics with time: the reactor power, the inlet and outlet sodium mass flow rates through the core and through the 2nd thermohydraulics channel. The results obtained by COREMELT and SIMMER-III prove that at least for the first stage of the ULOF accident, for the first 150 s of transient, there is no dry-out and degradation of the core. Instead, boiling stabilization is observed. The most important fact is that boiling with large amount of sodium vapor is located above the core – in the sodium plenum and upper shielding (see Fig. 10) – where the sodium void effect is negative. Boiling reveals itself in oscillations of reactivity and reactor power. The amplitude of these oscillations by SIMMER is greater and their frequency is less. In this SIMMER-III simulation the feedback associated with the axial expansion of the fuel part of the core is not accounted for whereas in COREMELT simulation this feedback is investigated. From Fig. 8 one can see that accounting for this feedback leads to more rapid power decrease for the initial stage of the accident and some delay in the beginning of boiling because the feedback is negative. The reason of the above mentioned differences in oscillation properties is not clear now and is to be investigated in additional calculations. It is to be noticed that, to EDF knowledge, such a behavior was never observed with SIMMER before. There is no explanation up to now for this change of behavior. This will need to be investigated further.

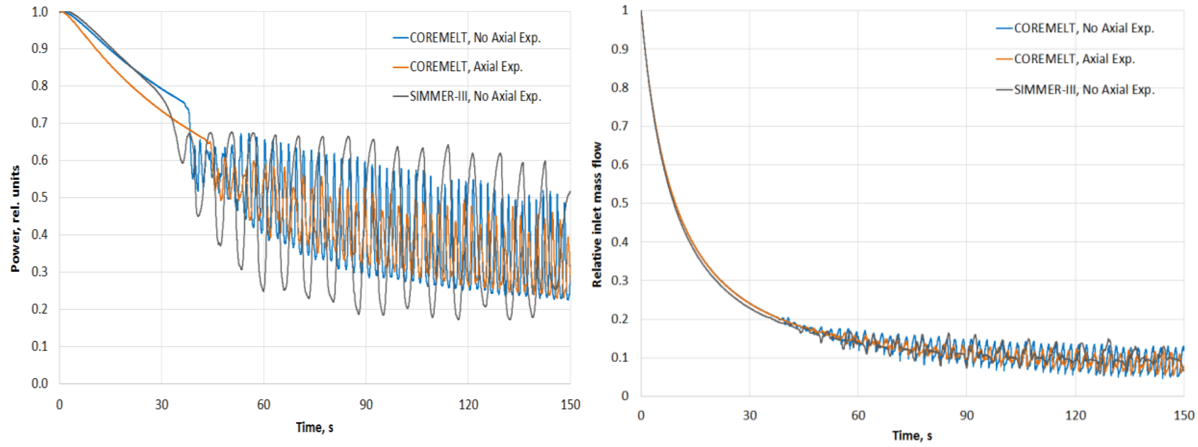

Fig. 8. Evolution of the Reactor Power and Inlet Flow Rate through the Core During the ULOF Transient

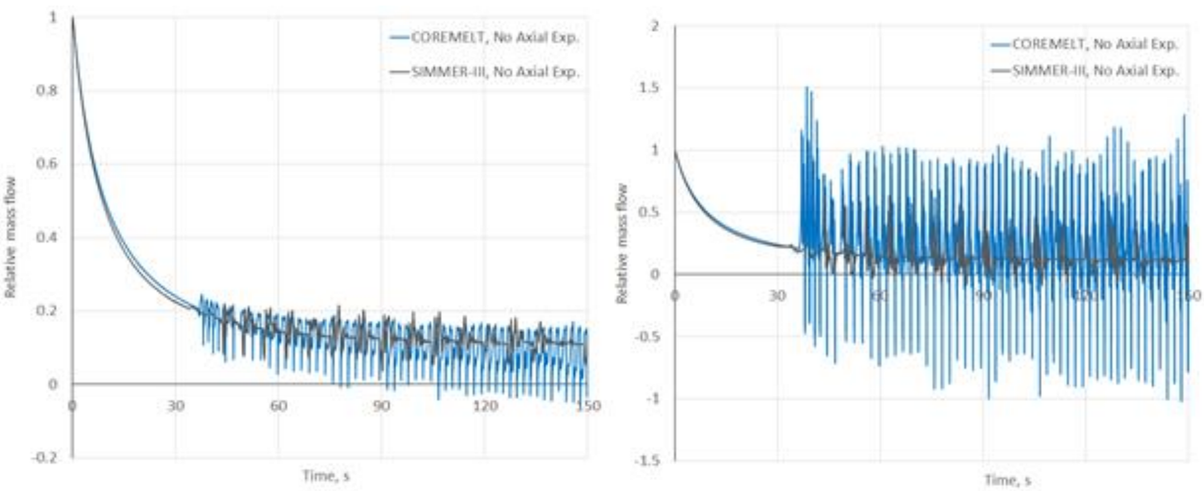

Fig. 9. Evolution of the Inlet and Outlet Plenum Flow Rate through the 2nd Channel

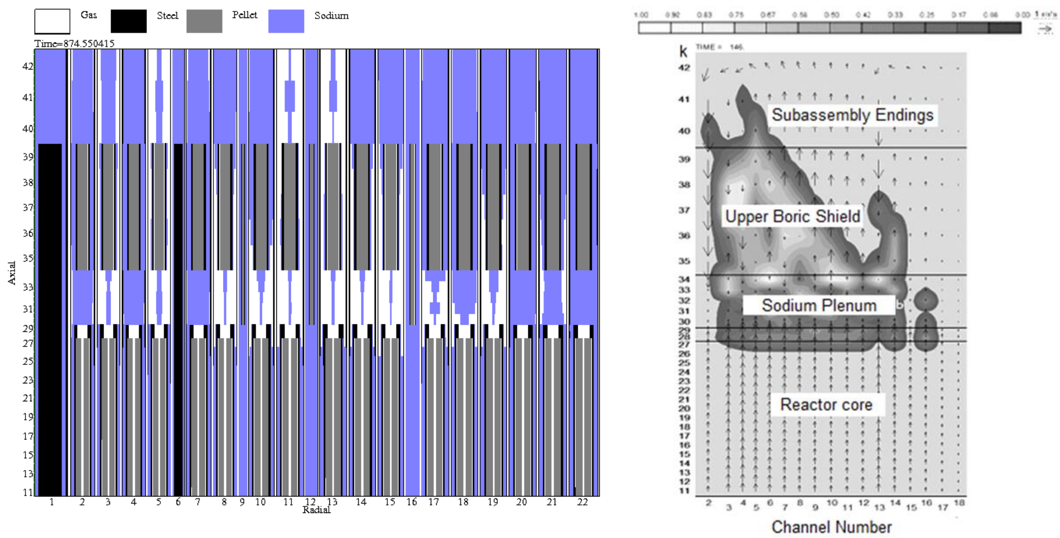

Fig. 10. Boiling at 146 s of Transient by SIMMER (left hand side) and COREMELT (right hand side)

5. Conclusion

A benchmarking exercise has been launched in the purpose of the analysis of the behavior of SVRE cores during severe accidents. The ULOF accident was simulated by SIMMER-III and COREMELT. In both cases, boiling stabilization and no core degradation are observed up to 150 s of transient. Yet, there remain certain discrepancies between the results given by the two codes. These differences are to be investigated in future work. More considerable length of transient (up to 1000 s) is to be investigated too. In addition, calculations of the benchmark with 3D versions of the codes (COREMELT3D and SIMMER-IV) are to be done.

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