Main outcomes from the JASMIN project: development and validation of ASTEC-Na for severe accident simulation in Na-cooled fast reactors

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Abstract. The 4 ½ year JASMIN Project, ended in May 2016, was launched in the frame of the 7th Research Framework Programme of the European Commission. It was inspired by the renewed interest in SFR technology and the Gen-IV challenging targets developed through the GIF initiative [1] of designing innovative reactors with higher safety standards than for the original Gen-III reactors that intrinsically prevent severe accidents or mitigate their consequences. The generic aim was the enhancement of the current capability of the ASTEC-Na code in analyzing severe accidents in Na-cooled reactors and, more especially, to develop its capacities to evaluate the consequences of unprotected accidents with fuel pin failure on material relocation and primary system loads and to predict fission product and aerosol behavior once released. To that extent, the JASMIN project represented a unique opportunity to bring together different actors of nuclear safety in Europe around the ASTEC-Na code. This enabled the achievement of a lot of work to validate and qualify the developed models in ASTEC-Na by comparison with other severe accidents codes (SAS-SFR, SIMMER…) and with dedicated past experimental results (i.e. CABRI tests..). The development and validation work performed during the project explored different technical areas (Na thermal-hydraulics, fuel pin thermos-mechanics, source term and neutronics). In that sense, ASTEC-Na has a big opportunity to cover many aspects of the SFR safety analysis. As a consequence, JASMIN perfectly fits what stated in the ESNII (European Sustainable Nuclear Industrial Initiative) roadmap by developing and benchmarking European computer codes for ESNII Fast Neutron Reactors, in particular relating to safety performance, leading to establish a common platform for modelling and simulation.

Key Words: JASMIN project, Na-cooled fast reactors, ASTEC-Na simulation tool, Severe Accidents.

1. Introduction

The JASMIN project was launched in the frame of the 7th FrameWork Programme of the European Commission (EC). It was inspired by the Gen. IV target of designing innovative reactors that intrinsically prevent severe accidents from occurring or drastically reduce their consequences, The generic aim was set to be the enhancement of the current capability of analysis of severe accidents in Na-cooled reactors and, particularly, to develop a new simulation tool able to evaluate the consequences of unprotected accidents with fuel pin failure on materials relocation, primary system loads, fission product and aerosols releases. To do so, the ASTEC platform, conjointly developed by IRSN and GRS for LWRs, has been chosen to be adapted and extended to the environment of Na-cooled fast reactors, the result being what has been called ASTEC-Na. The integrated features of the ASTEC software platform represents a good opportunity for simulating in a single code what is generally today simulated in separate codes (i.e., SAS-SFR and CONTAIN-LMR…). The high modularity of ASTEC-Na allows to work separately with each of its module and to integrate it back in the code at a later stage, so that any development, implementation and assessment is more straightforward than once implemented in the integral code. In addition, the flexibility in defining the core geometry, materials composition and reactor components makes ASTEC-Na ready to study new SFR designs with fertile layers in outer radial or inner axial core regions (such as in ASTRID design), subassemblies with an inner duct channel to induce fast fuel axial relocation (such as FAIDUS design), or new safety systems to shut-down the core power. The project addressed four main areas: Thermal-hydraulics, pin behavior, source term and neutronics. In each area, model development and validation have been performed. In addition to the test matrices built within the frame of the project and used as references for the model validation, the adequacy of ASTEC-Na models have been evaluated through the application of other suitable codes for benchmarking purposes. The main takeaways from the validation work were withdrawn in the form of a SWOT analysis (Strengths Weaknesses Opportunities Threats) that allows clearly identifying the main needs for future model developments and outlining what is considered as the pathway to fully make ASTEC-Na a reliable analytical tool for severe accident in Na-cooled reactors.

Key features of ASTEC-Na development and assessment performed within the JASMIN project are successively highlighted in this paper.

2. ASTEC-Na code development

During the JASMIN project, great efforts have been made for modelling the early stages of severe accidents in the primary vessel while some models relative to the in-containment phenomenology were also implemented. Scale of the phenomena considered for modelling can be varying from a single rod surrounded by a coolant channel up to a whole reactor core or plant. The ASTEC-Na code development was mainly based on the ASTEC platform conjointly developed by IRSN and GRS for LWRs [2], whose modules were progressively adapted and extended to the environment of Na-cooled fast reactors during the project. The different modules communicate with each other through a database managed by the ODESSA library and use a Material Data Bank which was updated for ASTEC-Na by including new material physical properties and notably sodium compound properties. ASTEC-Na takes also advantage of the mechanical and fission gas models issued from SCANAIR simulation tool developed in IRSN for reactivity-initiated accident (RIA) in LWRs. Those models were adapted during the project to take into account the SFR fuel pin specificities (fuel re-structuration, MOx properties….) before being included in the dedicated module.

More especially, during the project, four modules were essentially made available and operational in ASTEC-Na: CPA*, CESAR-Na, ICARE-SFR and SYSINT. Except for the last one (SYSINT dealing with the management of systems and events involved in an accident scenario), those modules were continuously developed and qualified during the whole duration of the project. In-containment thermal hydraulic is computed by CPA*. Models for sodium pool fires and associated sodium aerosol production and ageing were added for simulation of ex-vessel phenomena in case of primary sodium ejection in the containment. CESAR-Na computes the sodium single and double-phase thermal hydraulic in circuits and in the vessel. Models and correlations for sodium liquid/vapour heat and mass exchanges as well as wall/liquid sodium heat exchanges have been added. ICARE-SFR deals with the in-vessel thermal and mechanical behaviour of the fuel pins. During the project, this module was greatly developed by implementing models for fission gas behaviour (release, fuel swelling, cavity pressurisation…), for in-pin molten fuel motion, for fuel pin mechanics (clad and fuel circumferential deformation including an elastic, plastic and viscoplastic contribution, crack modelling…) and for clad failure prediction (several models can be activated). In addition, a 0D kinetics model was added for computing the neutronics in which five different reactivity feedbacks are taken into account sodium density effect, Doppler effect, clad axial and radial expansion effect, hexagonal wrapper tube axial and radial expansion effect, fuel axial expansion and in-pin relocation effect).

3. ASTEC-Na code assessment

The ASTEC-Na code verification and validation were performed by comparing its calculation results with experimental data, whenever available, and through code-to-code comparisons during several different benchmarks.

3.1 Assessment matrix

Regarding in-vessel phenomena, the assessment has been mainly done on experimental tests carried out in the CABRI and SCARABEE experimental reactors in the 1980 and 1990s operated by CEA (France) and investigating respectively single fuel rod and fuel rod bundle behavior during thermal-hydraulics and reactivity transients. For ex-vessel phenomena, the validation was more restricted as only sodium pool fires (thermal-hydraulics and sodium aerosol production) were investigated during the project**.** The assessment was based on the available experimental results found in the literature and more especially on ABCOVE and FAUNA tests performed in the 1980s respectively in the US CSTF (HEDL) and German FAUNA (FZK) facility. Two different validation test matrices were thus established at the beginning of the project for in-vessel phenomena (including Na thermal-hydraulics and fuel pin behavior) and ex-vessel phenomena focusing on Na pool fires. Both were then extended in accordance to the model developments. For neutronics, due to the lack of experimental data, the validity of the ASTEC-Na neutronics model was only assessed through a benchmark with TRACE and SAS-SFR codes under a selected accident scenario which was a ULOF transient in a pool-type sodium cooled fast reactor. As a benchmark reactor model serves the “Reference Oxide Core Design” concept of the CP-ESFR project under Beginning Of Life (BOL) core load conditions. For the in-vessel phenomena, the validation matrix includes different fuel characteristics (solid or annular fuel pellets, various burn-up…), multiple scales (from single pin in-pile tests to whole reactors) and explore several transients with different kinetics (ULOF, UTOP, CRWA…). For ex-vessel phenomena, five tests were included in the validation matrix taking into account the representativeness of their boundary conditions, multiples scales, the accuracy of the available data and the key variables reported.

3.2 Main Results in Thermalhydraulics validation

Several experiments from the general validation matrix for ASTEC-Na have been calculated in order assess the capability of the code and the reliability of the results (Table I).

Table I: Test matrix for thermal-hydraulic model validation and benchmarking codes.

Facility / Pant	**Test analysed with ASTEC-Na**	**Benchmarking codes**	**TH investigation**
CABRI	BI1, E8, EFM1	SAS-SFR, CATHARE, SIMMER-III, RELAP5-3D, RELAP5-Na	1D, Single-Phase Two-Phase
SCARABEE	BE+3, APL1, APL3	SAS-SFR, SIMMER-III	2D,Single-Phase Two-Phase
KASOLA	ULOF & ULOHS transients (Pre-test analysis)	CATHARE, RELAP5-3D, RELAP5-Na	Single-Phase Natural Convection
KNS	N02 natural circulation test	RELAP5-Na	Single-Phase Two-Phase
Phenix	Natural circulation test	RELAP5-Na, CATHARE	Plant Application
SuperPhenix	Stabiliz. & natural conv. Tests	DYN2B	Single-Phase Natural Convection

Among the in-pile single fuel pin CABRI tests performed at CEA/Cadarache, the pure LOF (loss of flow) BI1 test and the LOF + TOP (overpower transient) E8 and EFM1 tests have been selected as representative for both single and two-phase flow regimes. For the E8 and EFM1 tests the focus was only on the first LOF transient phase before TOP triggering. Satisfactory ASTEC-Na results have been found for sodium single phase flow behavior until the onset of boiling *(see FIG. 1)*, in substantial agreement with the test data and other codes response [1, 2]. Main deviations from the experimental trend were found in the evolution of the upper boiling front. Sensitivity analyses on BI1 and EFM1 tests showed that axial meshing refinement and the adjustment of heat losses in the upper part of the test section might help to improve the ASTEC-Na results in two-phase flow conditions.

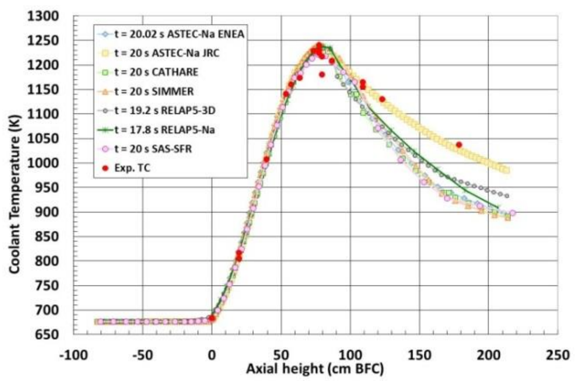


FIG. 1: CABRI BI1 test: Sodium temperature just before boiling onset around t = 20 s

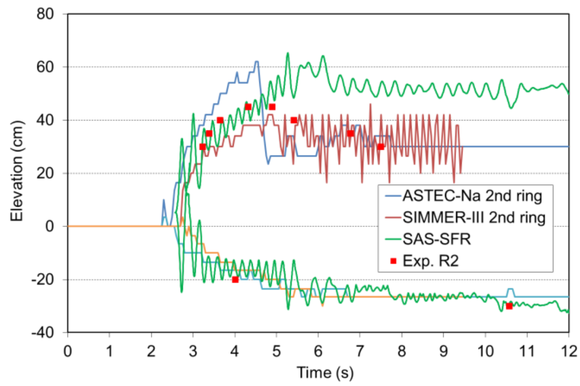


FIG: 2. SCARABEE BE+3 test: two-phase front evolution in channel 2

Among the in-pile SCARABEE experiments performed at CEA/Cadarache, the flow blockage BE+3 test and the LOF APL1 and APL3 tests have been selected as representative for both single and two-phase flow regimes in fuel pin bundle geometry. The analysis of SCARABEE tests [3] has confirmed the good performances of ASTEC-Na code for the calculation of single phase flow conditions. Some 2D effects observed in the pin bundle can be reasonably well captured by ASTEC-Na during the single-phase flow, and the results obtained with this code are of the same quality of the ones provided by SIMMER-III. However, the more complex 2D phenomena which occur after the onset of boiling cannot be reproduced with adequate accuracy. As for CABRI tests, the largest deviation from the experimental trend was found in the evolution of the upper boiling front *(see FIG. 2)*.

Pre-test analyses of KASOLA sodium loop under contraction at KIT/Karlsruhe have been performed with the ASTEC-Na code in various operating conditions and the ASTEC-Na results have been benchmarked with other codes (CATHARE, RELAP5-3D and RELAP5-Na). Several transient conditions were investigated during benchmarks which confirmed the suitability of the new thermal-hydraulic models implemented in ASTEC-Na.

Results from natural circulation tests performed in the pool-type Phenix French reactor have been used to verify the capabilities of ASTEC-Na code to simulate the overall plant behavior under both steady-state and transient conditions. The Phenix end-of-life test, aimed at studying the establishment of natural circulation in the primary circuit by tripping of primary pumps, was calculated with ASTEC-Na, CATHARE and RELAP5. These simulations pointed out that ASTEC-Na can calculate the whole plant including a multi-channel core, the pool-type primary system and the secondary circuits. The predictions of steady-state and transition to natural circulation conditions are good enough and in line with other thermal-hydraulic code results.

3.3 Main results in fuel pin thermo-mechanics validation

Four CABRI tests were used in ASTEC-Na fuel pin thermo-mechanical models assessment (Table II).

Table II: Test matrix for fuel pin thermal-mechanical model validation

Test analysed	AGS0	E7	E9	LT2
Burn-up (at.%)	2.9	4.6	4.6	12.4
Fuel pellet design	Solid	Annular	Annular	Annular
Clad material	316 CW	316 CW	316 CW	15-15 Ti stabilized
Transient type	TOP	TOP	Power Ramp	TOP
Duration (s)	0.6	0.8	118	0.9
Max. power (P/P_N)	12	150	2.2	25
Experimental observations	No failure Partial melting	Pin failure Cavity pressure built-up	No failure Extensive melting	No failure Fuel squirting

As these CABRI tests were conducted using irradiated fuel pins, the fuel pin power operation stages prior to the experimental tests have been evaluated by means of the GERMINAL code. GERMINAL results concerning fission gas behaviour, fuel and cladding structure evolution are used as input for ASTEC-Na calculations. By comparing Post Irradiation Examinations with GERMINAL results, it has been concluded that calculation results were suitable to be used as entry data for ASTEC-Na.

Considering the difference in power level reached during the power operation and the CABRI steady-state just prior to the transient, it is likely that the fuel pin undergoes changes in fission gas behaviour and internal structure (e.g. if CABRI steady-state power is much larger than the one during power operation, additional fission gas release and inner central hole variation are expected). For that reason, the steady-state phase was also simulated so to be as close as possible to the real state of the fuel pin just prior to the onset of the experimental test.

As for the overall fission gas behaviour during both fast and slow transients, ASTEC-Na results are quite satisfactory except for the AGS0 test where the overestimation of the fission gas release mainly comes from the GERMINAL irradiation calculations at low power heating rates. In all other tests more than 80 % of the fission gases are predicted to be released at the end of the transients despite the slower release calculated for the fast transients. Regarding the fuel melting limits for slow and fast power transients, results are in reasonable good agreement with Post Test Examinations *(see FIG. 3)*. Only the axial fuel melting limits were found slightly underestimated while the radial limits are in agreement with experimental results. Concerning the gap heat transfer model, the calculated coolant temperature was analysed by comparing against experimental data and results are in good agreement with the experimental measurements obtained from the thermocouples *(see FIG. 4)*.

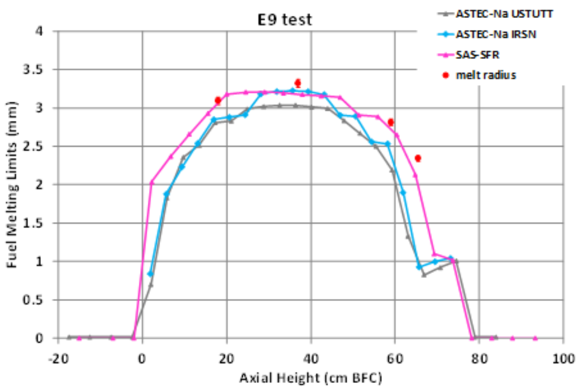


FIG. 3: CABRI E9 test: fuel melting limits

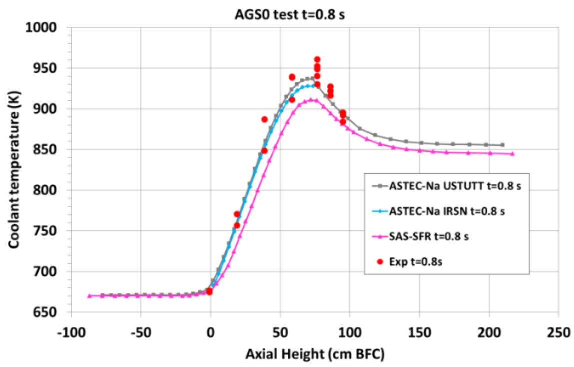


FIG: 4. CABRI AGS0 test: coolant temperature

Being ASTEC-Na able to simulate the fuel pin failure, E7 test was used to assess the failure models (Table III:). ASTEC-Na predicts an earlier clad failure than measured while the failure location is predicted below the real site and close to the maximum of the power profile.

Table III: Time and location of failure in E7 test

	Exp. Measurements	ASTEC-Na[1]	SAS-SFR
Failure time (ms)	467	444-448	468
Failure location (cm BFC)	53	39.4-41.3	47.7

Considering the ASTEC-Na modelling improvement of the latest version (V2.1) compared to the previous one (V1.1), one can generally conclude that the fuel pin thermal-mechanical

1 Results obtained by two different calculations

models predict acceptable results, although further work is still needed to improve some specific results as well as to enlarge the experimental database simulated so that an extended validation process is achieved.

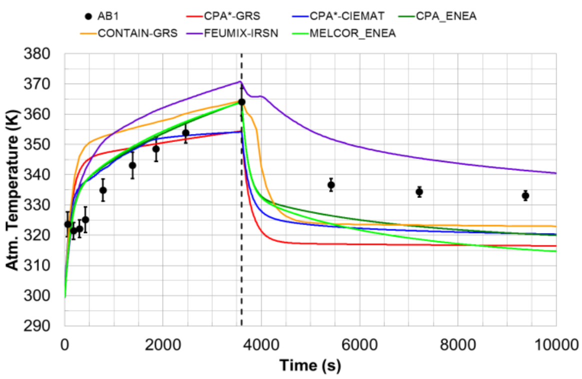
3.4 Main results in containment source term validation

In order to assess the enhanced ASTEC-Na CPA version (CPA*), a literature survey was carried out and more than 20 experiments related to in-containment source term have been reviewed and properly stored in a data base. Five Na-pool fire experiments have been chosen (Table IV). Other codes also participated in the benchmarking against this data set.

Table IV: Experimental test matrix.

	AB1	AB2	F2	F3	EMIS 10b
Geometry					
Type	Cylindrical	Cylindrical	Cylindrical	Cylindrical	Cylindrical
Volume (m^3)	852	852	220	220	4.4
O2 (%)	19.8	20.9	17-25	15-25	20
Temperature (K)	299.65	293.65	298.15	298.15	294.05
RH (%)	35.5	43.3	-	-	-
Steam Addition	NO	YES	NO	NO	NO
Initial Na Temp. (K)	873.15	873.15	773.15	773.15	554
Burning Area (m^2)	4.4	4.4	2	12	0.125
Fire duration (s)	3600	3600	12600	4800	6000

Consistently with experimental observations, CPA* show two main phases (heat-up and cool-down) and even though differences in magnitude are roughly moderate, qualitative deviations may be observed during some periods of both phases (*FIG. 5-. 6)*. In particular, cooling rate due to fire quenching is largely overpredicted by CPA* at the beginning of the cooling phase, which might indicate a too high energy source during the fire period (0-3600 s); in the longer cooling period there are also deviations but not as noticeable as those discussed above. The thermal behavior described is relying on two parameters (f_1 and f_2) in the CPA* model for the Na combustion energy distribution, which lack of sound basis to be set and/or estimated. .



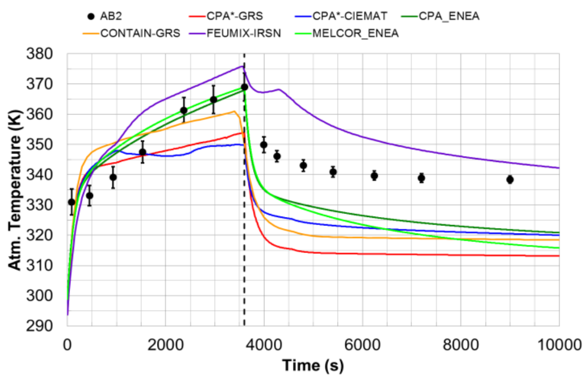


FIG. 5: AB1 test: atmosphere temperature *FIFIG.4.6CABRItest:Sodium atmosphere temperature* *FIG: 4. CA*
axial profile at 0.8 s

Three main metrics were used to benchmark CPA* performance: suspended mass concentration, aerodynamic mass median diameter (AMMD), and deposition of material on different surfaces at the end of the tests (*FIG. 7* through *9)*. Large data uncertainties in airborne concentration before 1000 s prevent from any meaningful comparison. Generally speaking, CPA* results fall within experimental uncertainties during the quasi-steady state period, but none of the cases follow the observed steady trend before the sodium pool fire ending. Besides, the experimental depletion rate during the first 1000 s of the depletion phase is about twice faster than the code. Despite these differences, CPA* reasonably follows AMMD measurements along time. Finally, CPA* final mass distribution captured data qualitatively, but significant differences are noted in magnitude due to the overestimate of thermophoresis as a deposition mechanism.

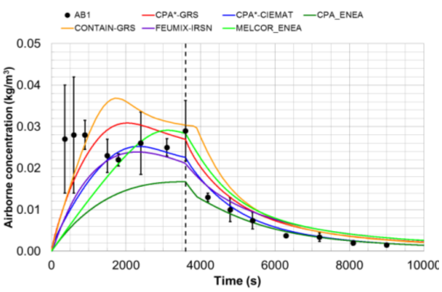


FIG. 7: Airborne concentration (AB1).

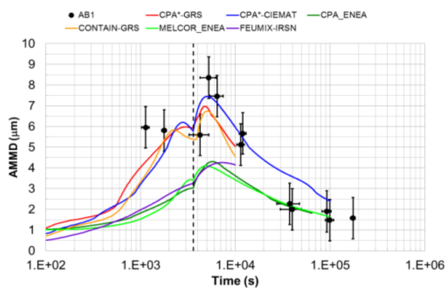


FIG. 8: AMMD (AB1).

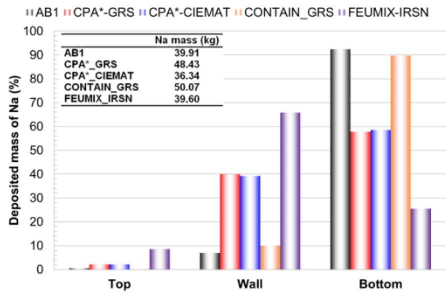


FIG. 9: Na mass distribution (AB1).

Experimental O_2 mole percent in the vessel atmosphere was also analyzed as an indicator of the performance of CPA* chemical model. Consistently with observations, CPA* captures the slight O_2 depletion during the burning period with a reasonable accuracy. As for CPA* predictions concerning deposited aerosol composition, none of the CPA* calculations showed remarkable similarities to data at the two times (16 min and 46 min) comparisons are feasible.

3.4 Main results in neutronics validation

Based on the point-kinetics theory, the ASTEC-Na neutronics model has currently two different methods to compute the reactivity feedbacks: one model (called LOCAL) uses coefficients related to temperature (pcm/K), the other model (LOCALM) is based on mass variation and allows extending the scope of the ASTEC-Na neutronics model to post boiling onset analysis. The model LOCALM takes into account five reactivity feedbacks:

- Doppler effect
- Fuel reactivity effect (fuel axial expansion and fuel in-pin relocation)
- Clad reactivity effect (clad axial expansion)
- Hexagonal wrapper tube reactivity effect (wrapper tube axial expansion)
- Sodium reactivity effect (sodium density variation, sodium voiding, as well as clad and wrapper radial expansion into the coolant channel)

The work of assessment of both models has been performed through a benchmark exercise against SAS-SFR code that involves the simulation of the initiation phase of an unprotected loss of flow accident (ULOF) in a pool-type sodium cooled fast reactor. The accident is initiated by the failure of all primary pumps without activation of the reactor shutdown systems leading to a subsequent decrease of the coolant flow in the core. The coolant flow rate decreases rapidly due to the short coolant flow halving time of 10 s, provoking undercooling of the core. The reduction in power generation develops much slower so that the power-to-flow ratio mismatch results in a subsequent rapid single phase coolant heat-up which may lead to coolant boiling, clad dryout, clad melting and relocation, and subsequent fuel pin break-up followed by core materials relocation.

The results show a good agreement among the codes both qualitatively and quantitatively in computing the progress of the main core parameters. Both models are correctly implemented and LOCALM allows ASTEC-Na to calculate the neutronics feedback (and associate power response) taking into account fuel relocation reactivity feedbacks, as well as sodium void effect. The calculations of ASTEC-Na LOCAL and SAS-SFR are in good agreement. While the ASTEC-Na LOCALM shows a lower peak due to its slightly lower total reactivity feedback (*FIG. 10* and *11*). By analyzing the individual reactivity feedbacks, it appears that ASTEC-Na LOCALM predicts similar evolutions of the coolant and Doppler reactivity feedbacks while respectively over-predicts the contribution of the fuel expansion and under-predicts that of the cladding expansion to the total reactivity feedback. Given the good agreement of ASTEC-Na results using the LOCAL reactivity feedback model (based on temperature), it appears that by using the LOCALM reactivity feedback model ASTEC-Na finally computes a different fuel and cladding expansion than SAS-SFR with the same material temperature evolution. Other discrepancies between ASTEC and SAS are associated with the different meshing system and approach used for the calculation of temperatures in the two codes.

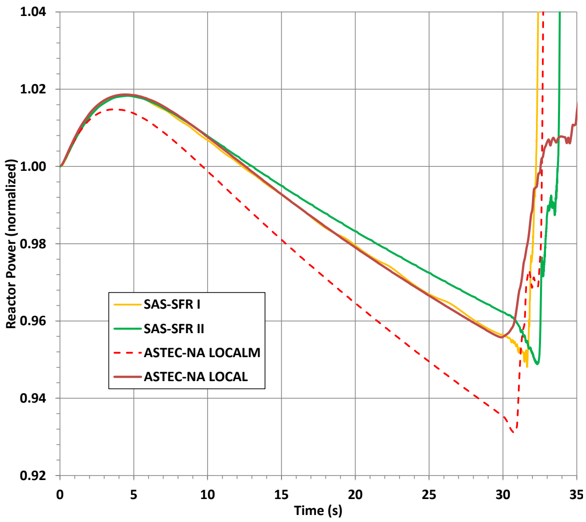


FIG. 10: Neutronics Benchmark: Power Evolution

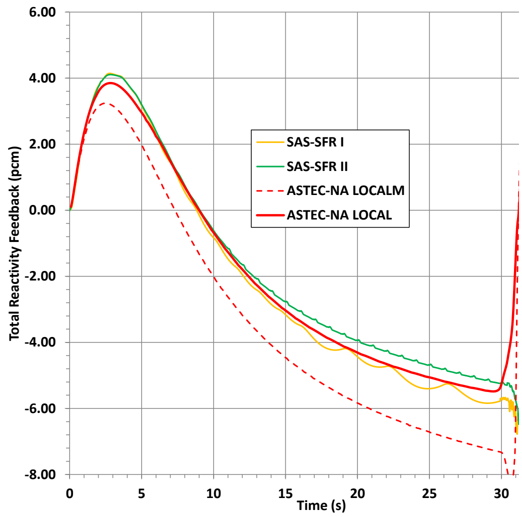


FIG. 11: Neutronics Benchmark: Total Reactivity Evolution

4. Conclusions

The main accomplishments achieved by JASMIN were to develop and validate models that were further implemented in the ASTEC-Na safety analytical tool dedicated to simulate the initiation phase of a severe accident in SFRs. Four different areas have been addressed during the project (sodium thermal hydraulics, fuel pin thermomechanics, in-containment source term and neutronics). Thanks to the work performed during the project, the predictive capability of ASTEC-Na (that goes farther than previous generation of analytical tools) was greatly enhanced as well as its strengths and shortcomings clearly identified so that a number of needs have been formulated to further enhance its robustness and to extend its scope. The sound bases of ASTEC-Na (i.e., ASTEC code system) and the existing similarities with Pb-cooled and Pb-Bi reactors, turn it to be a good option to develop an ASTEC-LM (Liquid Metal) version capable of addressing also severe accidents in this type of reactors.

Acknowledgments

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