NURETH-16
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Karlsruhe Institute of Technology

JOINT ADVANCED SEVERE ACCIDENTS MODELLING

Single and two-phase sodium flow analysis for two TUCOP CABRI tests using the ASTEC-Na code

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Introduction

- High safety standards need a robust validated computational simulation software able to predict fuel pin behaviour, thermodynamics, thermo-hydraulics and the whole accident sequence.

- Code system development and validation require adequate sets of experimental data covering the various interfering physical processes and phenomena taking place during an accidental event.

- The **JASMIN** project of the 7th European Framework Programme aims at developing a new computer code system, **ASTEC-Na**, capable to evaluate the consequences of protected and unprotected accidents in Sodium-cooled Fast Reactors.

- **JASMIN** project is devoted to modelling and validation of:
  - Sodium thermal-hydraulics
  - Fuel pin thermo-mechanical behaviour
  - Source term
  - Neutron physics

Introduction

- In **JASMIN** project **ASTEC-Na** sodium thermal-hydraulics models will be validated using:
  - SCARABEE and CABRI in-pile experiments
  - Natural circulation test conducted in PHENIX reactor
  - Tests to be carried out in the KASOLA sodium loop (ULOF, ULOHS, natural circulation)

- **ASTEC-Na** models will also be benchmarked against other safety codes: SAS-SFR, CATHARE, SIMMER, RELAP5-3D, RELAP5-Na.

- Participants of the present TUCOP tests simulations (E8 and EFM1) are:
  - ENEA (Italy) using **ASTEC-Na**, CATHARE and RELAP5-3D
  - KIT (Germany) using SAS-SFR

Outline

- **ASTEC-Na** Thermal-Hydraulics Models
- Transient Under Cooling Over Power (TUCOP=LOF+TOP) Tests Analyses
  - E8 Test: 22.1 s LOF phase (21.6 s 1phase + 0.5 s 2phase)
  - EFM1 Test: 30.7 s LOF phase (22.7 s 1phase + 8.0 s 2phase)
- Conclusions and Future Work

ASTEC-Na thermal-hydraulics models

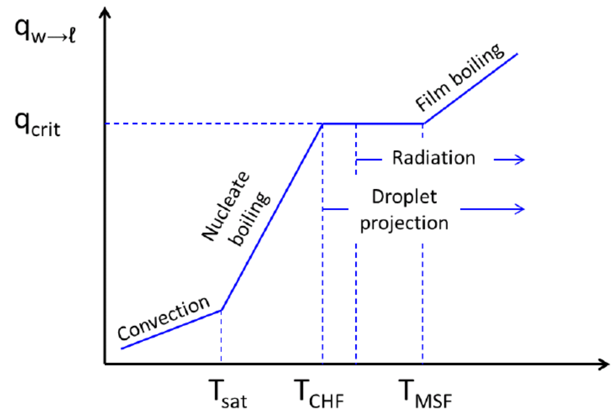
- **ASTEC-Na** based on ASTEC code system (IRSN and GRS), extensively validated in European projects for LWR.

- **CESAR** is the thermal-hydraulics module of **ASTEC-Na**.
  - Adapted to SFR by implementing Na physical properties and updating heat and mass fluxes for liquid-vapor phases and Na-wall interphases.
  - Provides boundary conditions to: fuel pin behavior, neutronics, fission products and aerosol transport , ...

- Description of the CESAR models:
  - Liquid-vapor heat and mass exchanges based on the kinetic theory of gases.
  - Flows at the interphase: vapor condensation when hitting the wall and spontaneous liquid evaporation.
  - Sodium flashing: pressure is lower than the liquid saturation one, liquid bulk boiling.
  - Bulk condensation: vapor pressure is greater than the vapor saturation pressure.
  - Convective mechanism between Na phases, significant in volumes with massively non-condensable gases.
  - Newton's law of cooling where HTC is estimated as the inverse of individual thermal resistances of phases:
    - pure conduction for the dispersed phase (bubbles or droplets)
    - convection for the continuous phase

ASTEC-Na thermal-hydraulics models

- Wall-fluid heat exchange phenomena modeled: convection, nucleate boiling, film boiling, thermal radiation and droplet projection (heat flux from the droplets in the quench front).



- For convection several options are given to estimate Nusselt number.
- Nucleate boiling is modeled with the Forster & Zuber correlation.
- Critical Heat Flux Temperature T_{CHF} obtained from Thom's correlation.
- Critical Heat Flux q_{CHF} from Zuber's correlation corrected for liquid subcooling.
- Minimum Stable Film Temperature T_{MSF} calculated using Berenson's correlation.
- Radiation exchange based on the grey-body approximation (300 W/m^{2}K HTC value in the projection region).
- Convection and radiation for wall-vapor heat transfer (max. convection assumed between nat. and forc. reg.)
- Dittus-Boelter correlation used for forced regimes.

RELAP5-3D, CATHARE, SAS-SFR codes

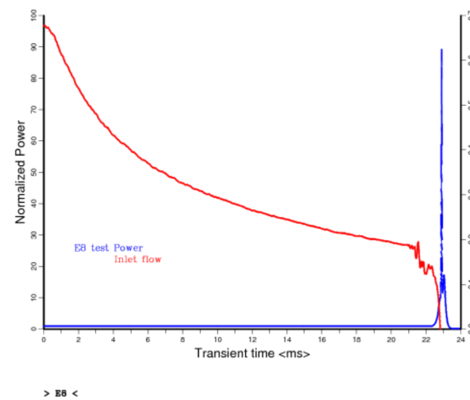
- RELAP5-3D (Idaho National Laboratory)
  - Thermal-hydraulic transient analysis of LWR but extended to SFR.
  - Wide variety of hydraulic and thermal transients (nuclear and non-nuclear systems).
  - Mixtures of vapor, liquid, non-condensable gases and non-volatile solutes.
  - Seban-Shimazaki correlation is used for convective heat transfer with sodium coolant.

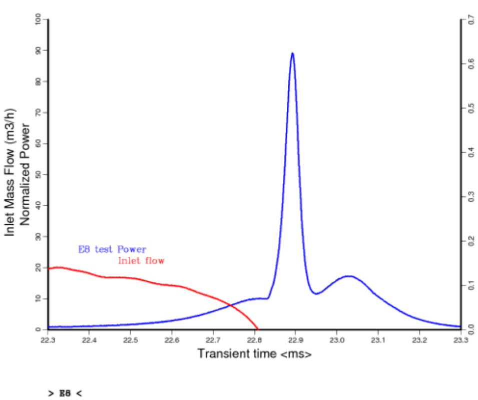
- CATHARE (CEA, EDF, AREVA and IRSN)
  - Originally conceived for PWR safety studies and recently extended to other nuclear reactors, such as SFR.
  - Flexible modular structure for thermal-hydraulics: simple experimental test facilities to nuclear power plants.
  - The Spukinski heat transfer correlation is used for sodium.

- SAS-SFR (KIT/INR, CEA, IRSN and JAEA)
  - Deterministic analysis for steady state power operation and accident conditions in SFR during the initiation phase.
  - Extensive qualification of steady-state fuel irradiation, transient fuel deformation, primary coolant system heat transport, sodium boiling model, cladding melting and motion, fuel failure behavior in voided and un-voided regions.
  - Multi-channel model with fuel assemblies represented by a single pin. Multiple-bubble slug ejection model.

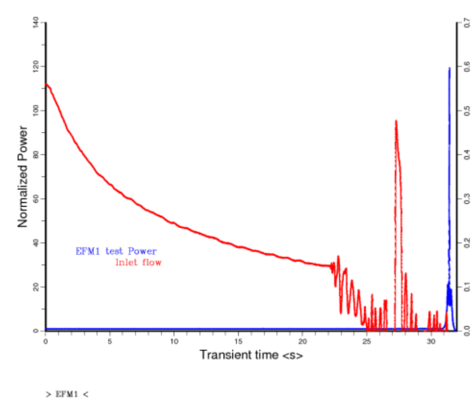
Transient Under Cooling Over Power (TUCOP)

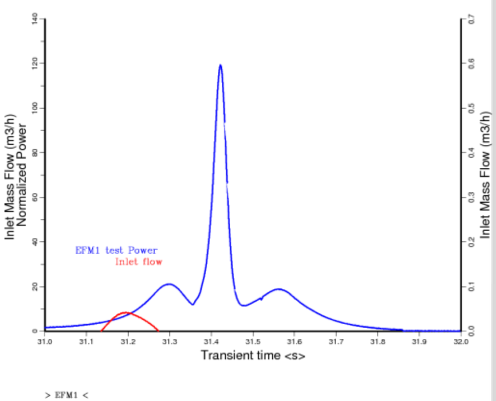
- **E8 test**
  - Annular MOX fuel with 4.6 at. % burn-up
  - 316 SS cladding
  - PPN linear power 593 W/cm
  - TOP triggered in a partially voided channel
  - 22.1 s LOF phase (21.6 s 1ph + 0.5 s 2ph)





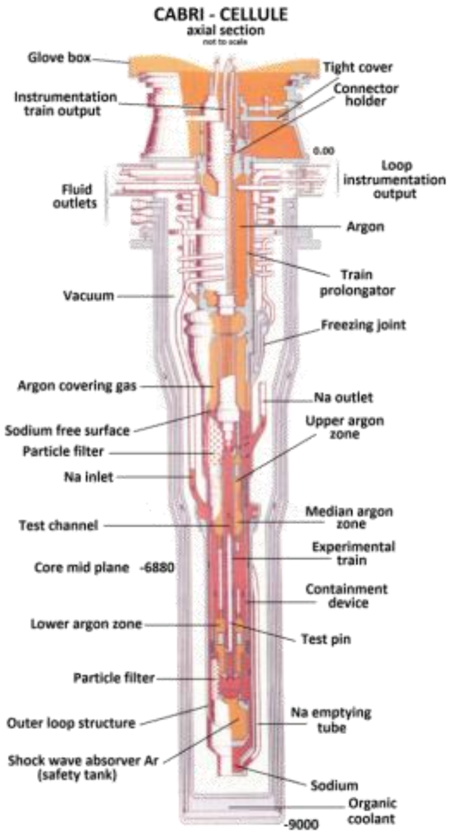
- **EFM1 test**
  - Annular MOX fuel with 6.4 at. % burn-up
  - 15-15 Ti cladding
  - PPN linear power 487 W/cm
  - TOP triggered in a voided channel
  - 30.7 s LOF phase (22.7 s 1ph + 8 s 2ph)

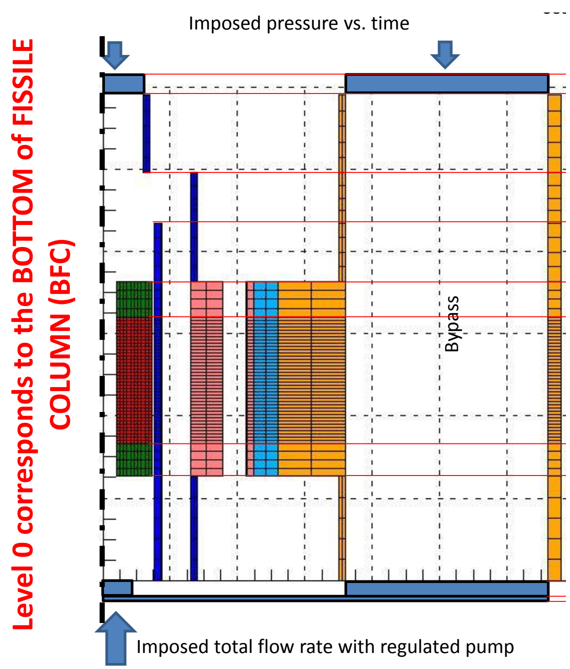




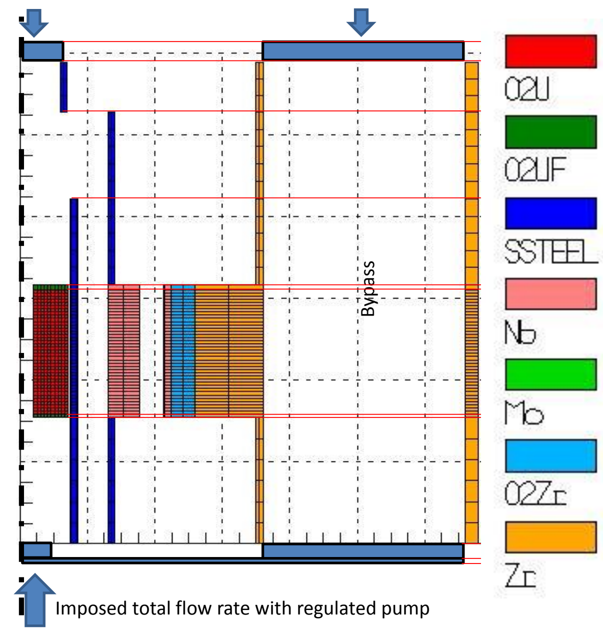
E8 and EFM1 Code Modelling

- **ASTEC-Na** modelling: fissile column, lower and upper fertile blankets, fission gas plena and lower and upper structures of the test section.
- CATHARE, RELAP5-3D and SAS-SFR consider the same regions as **ASTEC-Na**.
- RELAP5-3D only simulates the structure of the test section above the fuel pin.





E8 test

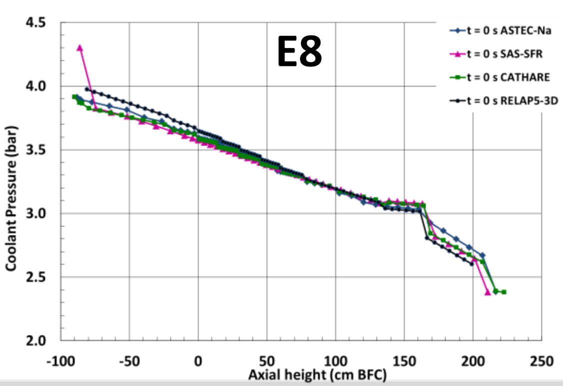


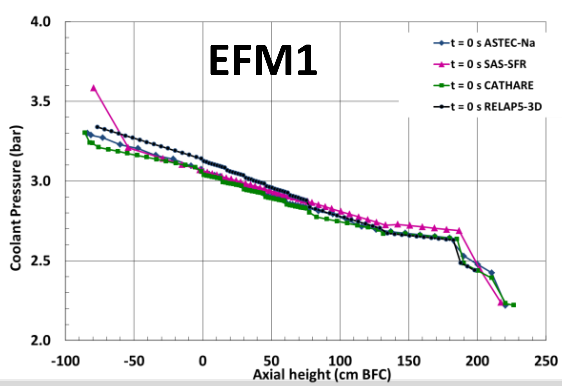
EFM1 test

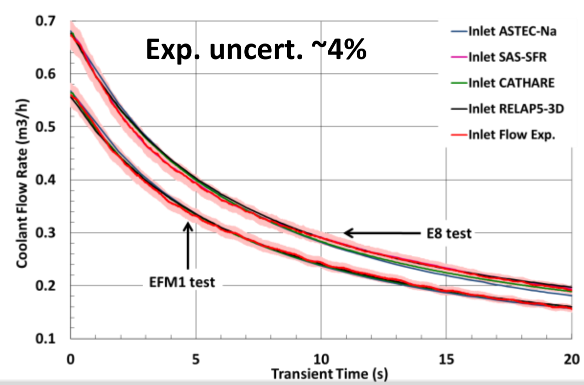
E8 and EFM1 Initial Conditions

- GERMINAL code provides pin characteristics to **ASTEC-Na**, CATHARE and RELAP5-3D after the PHENIX power operation simulation.
- SAS-SFR is able to simulate the power operation providing pin geometry, fission gas retention, fuel and clad thermal and mechanical characteristics.

	Experiment	ASTEC-Na	CATHARE	RELAP5-3D	SAS-SFR
E8					
Fuel Clad Gap Width at PPN (µm)	-	66.12	66.10	66.12	0.00
Inner Fuel Radius at PPN (mm)	-	1.023	1.023	1.023	1.144
Total channel power (W)	36,709	36,600	36,600	36,600	38,830
Total power produced in the fuel (%)	96.6	100	100	100	98.0
Peak Linear Rating (W/cm)	593	611	612	626	608
EFM1					
Fuel Clad Gap Width (µm)	-	80.30	80.30	80.30	0.00
Inner Fuel Radius (mm)	-	1.036	1.036	1.036	1.234
Total channel power (W)	31,050	31,050	31,050	31,050	31,530
Total power produced in the fuel (%)	98.3	100	100	100	97.3
Peak Linear Rating (W/cm)	491	504	504	513	491

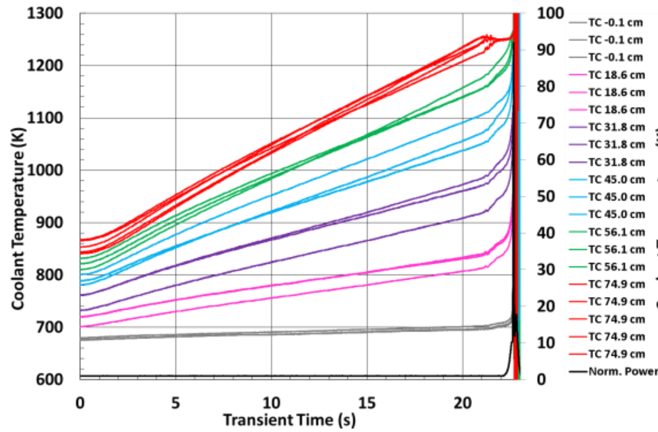


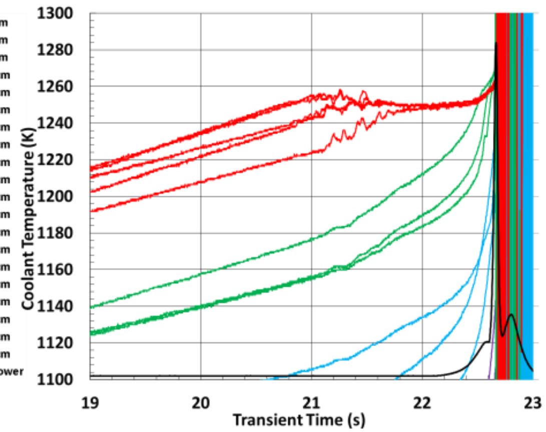


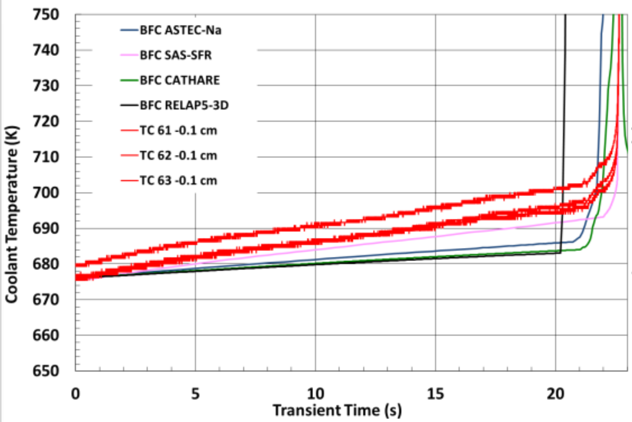


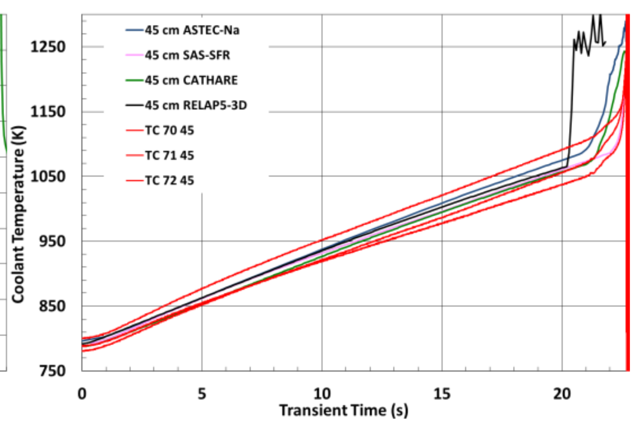
E8 test results

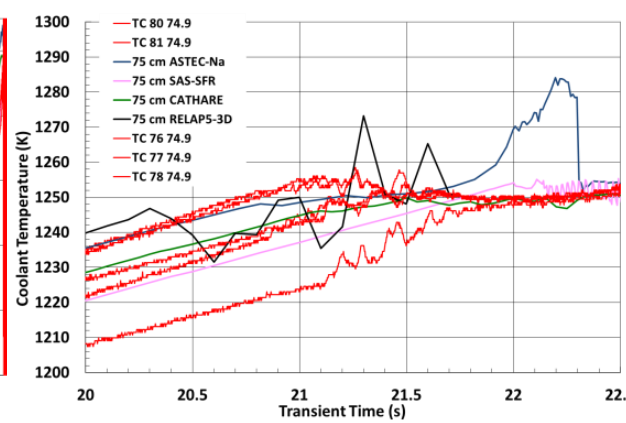
- Linear temperature increase
- Pin distortion: diff. cross sections
- Estimated TC uncertainty 1.6 %
- LBO at TFC 20.7 s
- BBO total at 21.6 s (0.9 s)





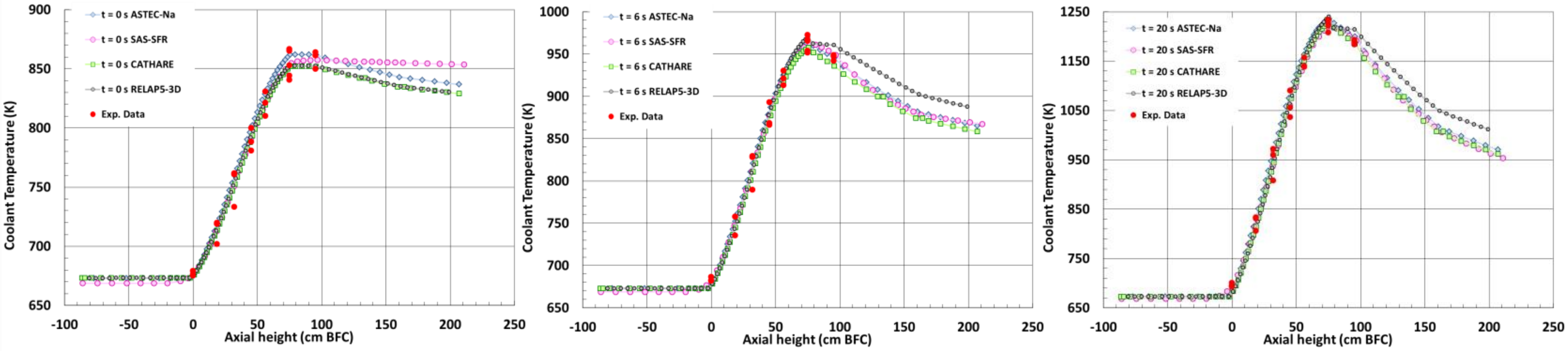






E8 test results

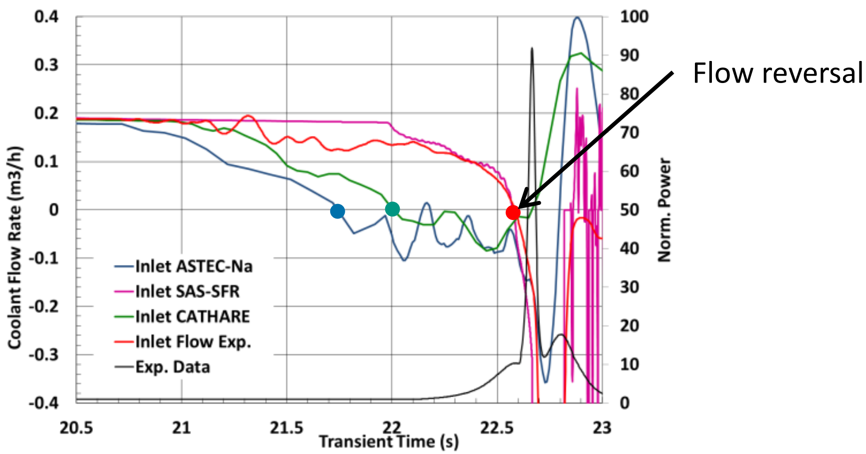
- Axial profiles of coolant temperature

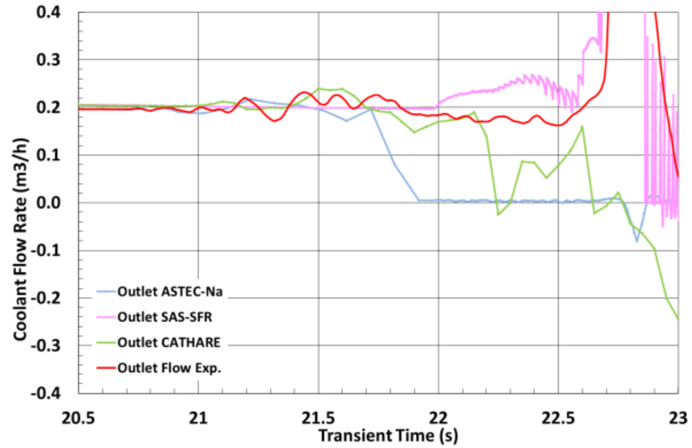


	Experiment	ASTEC-Na	CATHARE	RELAP5-3D	SAS-SFR
Local boiling onset time (s)	20.7	-	-	-	-
Bulk boiling onset (s)	21.6	20.92	21.10	20.40	21.98
Boiling onset height (cm BFC)	75*	75.62	75.62	69.72	79.17
Saturation temperature (K)	1250	1247.2	1246.3	1248.0	1251.0
Pressure at TFC (bar)	-	2.17	2.19	2.19	2.18
Clad dry-out time (s)	Not observ.	22.0	22.5	-	-
Clad melting onset time (s)	Not observ.	22.7	22.8	-	-

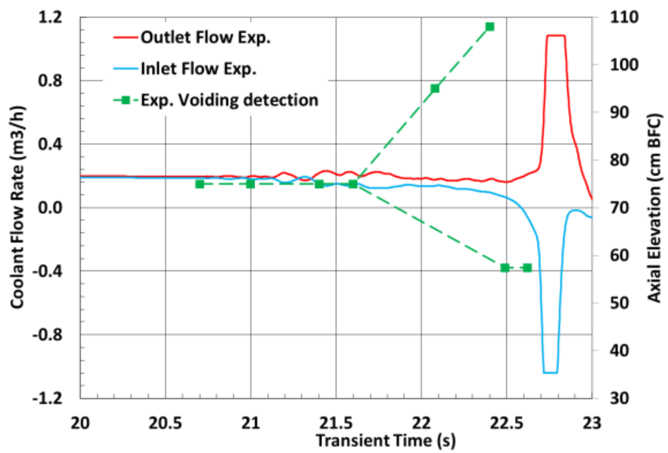
E8 test results

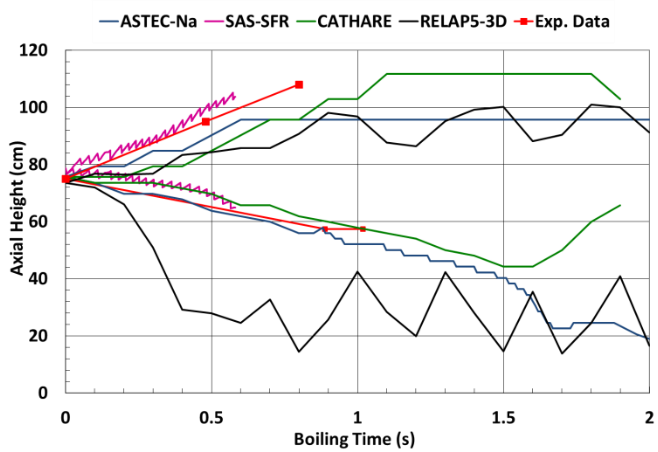
- Inlet and Outlet Flow rate after boiling onset





- Voiding front





EFM1 test results

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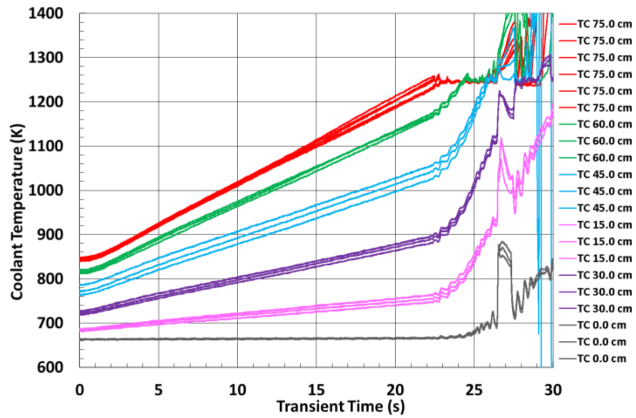
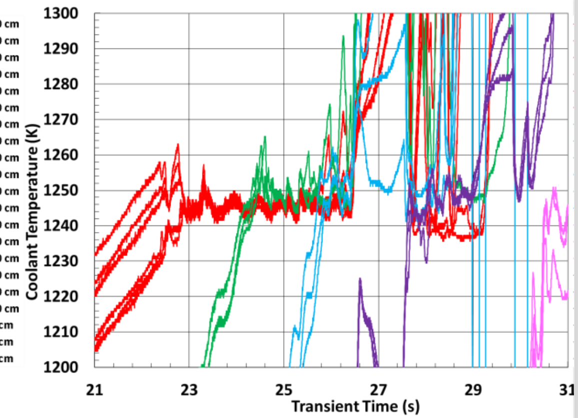
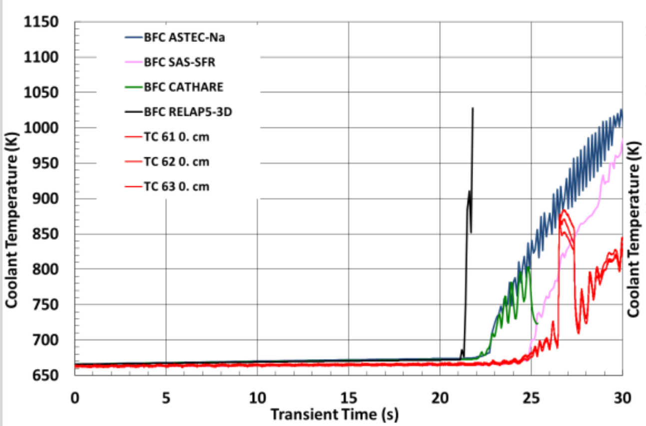
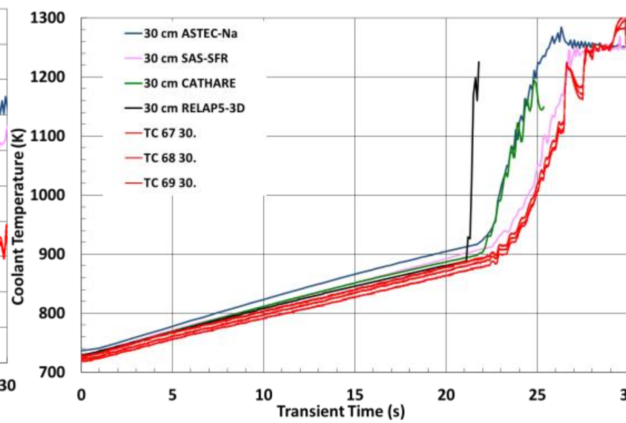
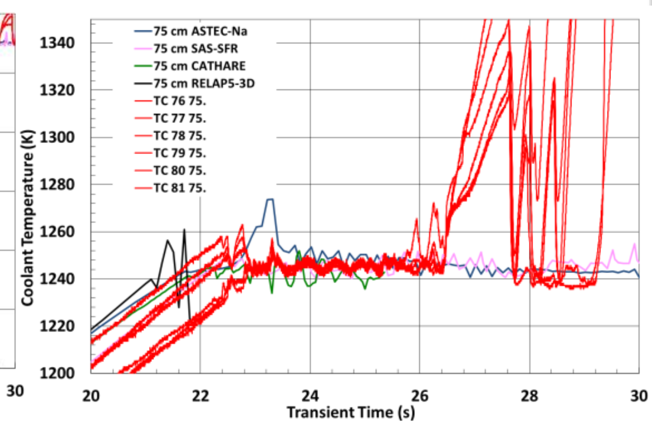
- Pin distortion
- Estimated TC uncertainty 1.6 %
- LBO at TFC 21.9 s
- BBO total at 22.7 s (0.8 s)
- Clad dry-out observed (3-3.7 s)
- Fission gas blow-out
- Clad melting





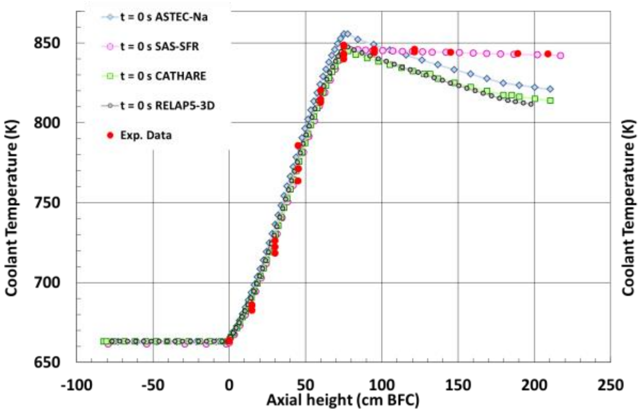


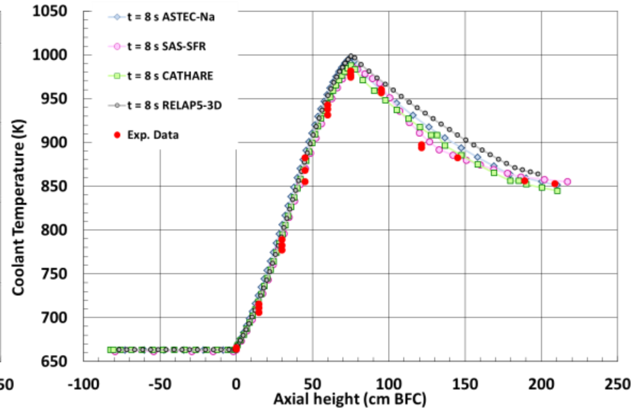


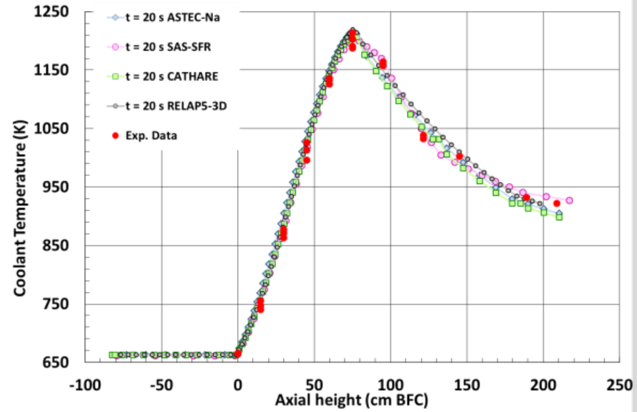


EFM1 test results

- Axial profiles of coolant temperature



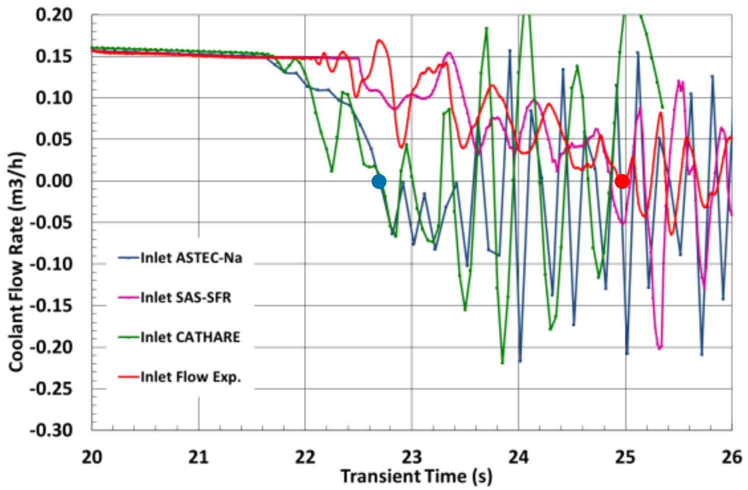


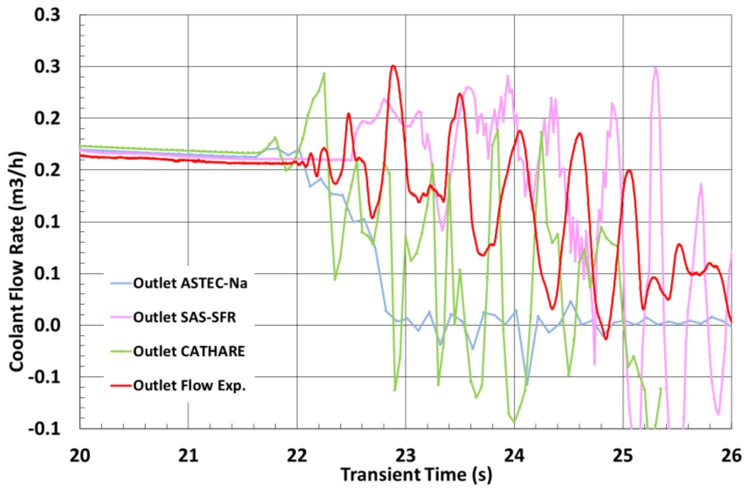


	Experiment	ASTEC-Na	CATHARE	RELAP5-3D	SAS-SFR
Local boiling onset time (s)	21.9	-	-	-	-
Bulk boiling onset time (s)	22.7	21.80	21.70	21.60	22.51
Boiling onset height (cm BFC)	75*	75.22	75.22	75.45	77.00
Saturation temperature (K)	1246	1242	1241	1252	1248
Pressure at TFC (bar)	-	2.09	2.10	2.27	2.12
Clad dry-out time (s)	25.7-26.4	23.00	23.5	-	25.1
Clad melting onset time (s)	-	24.70	24.7	-	26.3

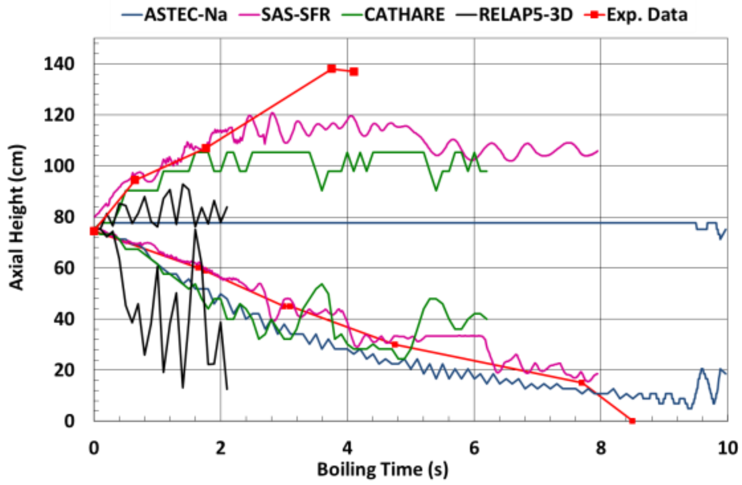
EFM1 test results

- Inlet and Outlet Flow rate after boiling onset





- Voiding front



ASTEC-Na sensitivity analysis in EFM1 test

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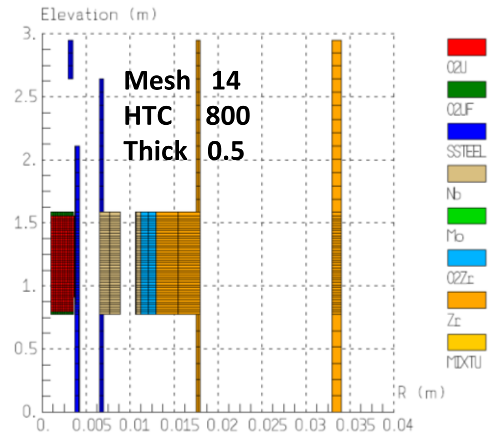
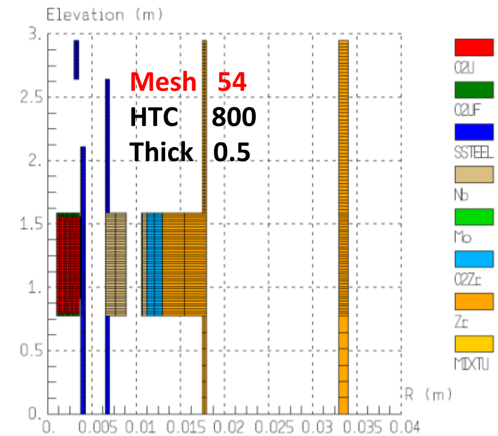
- Modification in the number of axial meshes above fissile zone

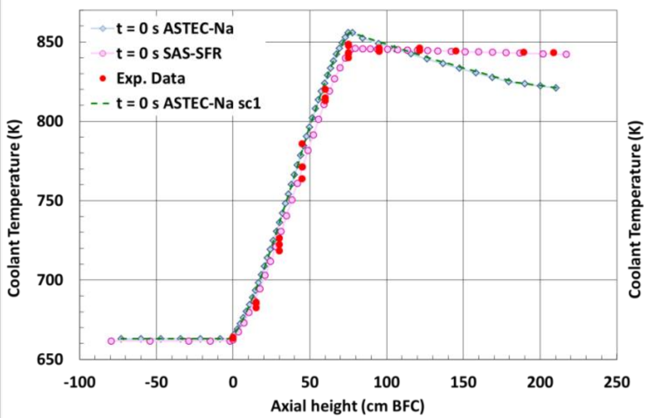


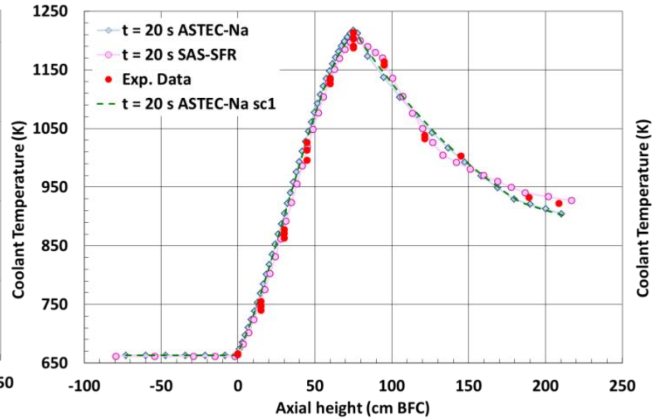
Base Case (BC)

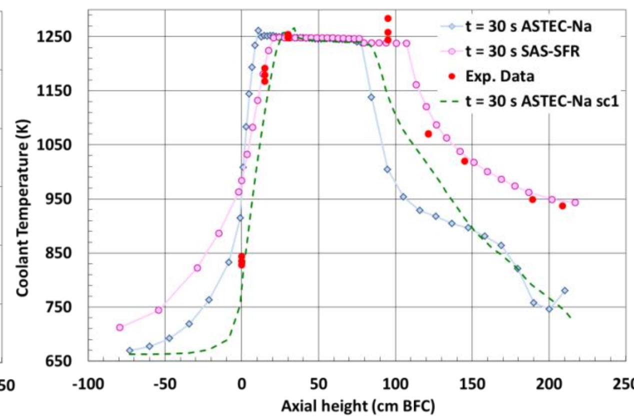


Sensitivity Case 1 (SC1)

- Coolant Temperature







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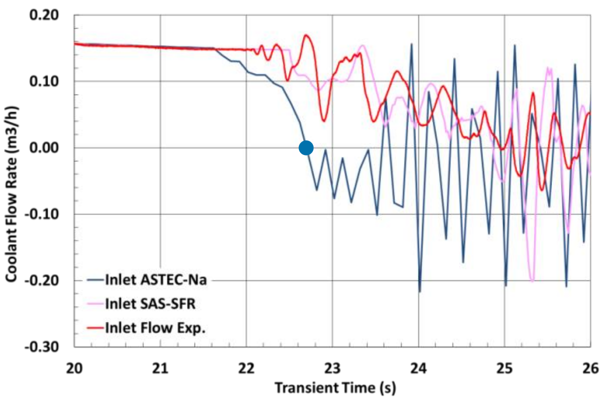
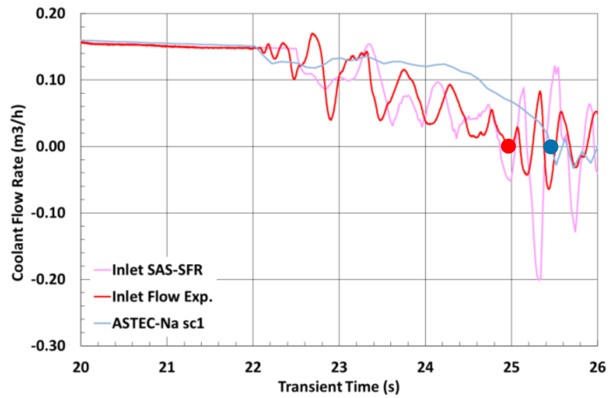
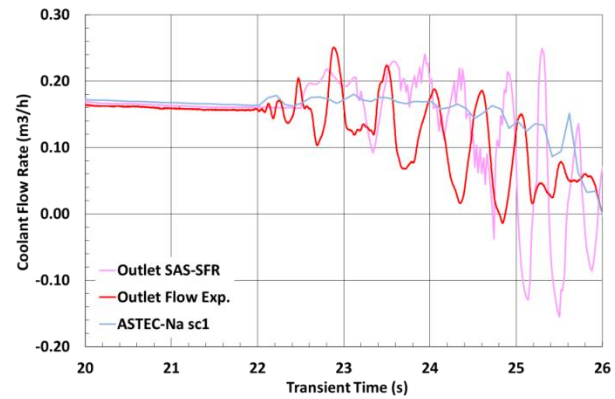
ASTEC-Na sensitivity analysis in EFM1 test

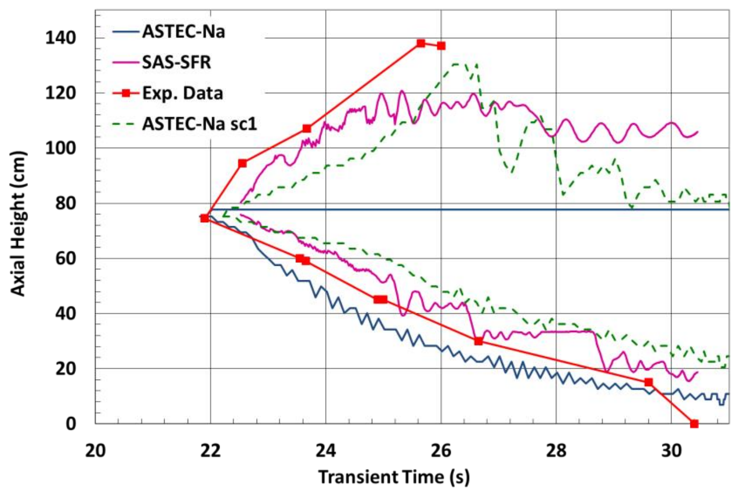
- Inlet and Outlet Flow Rate







- Voiding Front



Conclusions

- **ASTEC-Na** satisfactory results during the single phase of both E8 and EFM1 tests.
- Small differences in boiling onset between **ASTEC-Na** and the experimental data.
- Difficulty to consider the fuel pin distortion, local boiling inception.
- **ASTEC-Na** inlet and outlet flow after boiling onset is inaccurate. Lower voiding interface is in agreement with the experimental data for E8 and slightly underestimated for EFM1. The upper front is underestimated for both tests.
- Parametric studies on the consideration of radial heat losses and on the meshing above the fissile zone:
  - The finest meshing slows down the flow rate reduction after the onset of boiling according to experimental measurements
  - The heat loss and thermal inertia of heated section above the fissile zone has impact on the axial coolant temperature distribution in the upper part of the test section affecting after boiling onset also the voiding behaviour and coolant temperatures in the lower part.
- Code-code comparison: differences attributed to models, radial heat losses and pressure calculation.

Future work

- Implementation of Lockhart-Martinelli model for two phase pressure drop based on the single phase:

$$\Delta P_{2\varphi} = \Phi^2 \Delta P_{1\varphi}$$

(Currently **ASTEC-Na** wall friction model is based on Blasius correlation with an averaged parameter, homogeneous mixture)

- Implementation of various correlations for Φ^2 estimation

- Validation foreseen:
  - Steady-state boiling experiments: ISPRA Tubular Experiment and KNS-37 S-33
  - LOF transients: KNS-37 L-22

Acknowledgement

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Joint Advanced Severe Accidents Modelling

www.jasmin-fp7.eu