PANCHO – Physics and Chemistry of Nuclear Fuel

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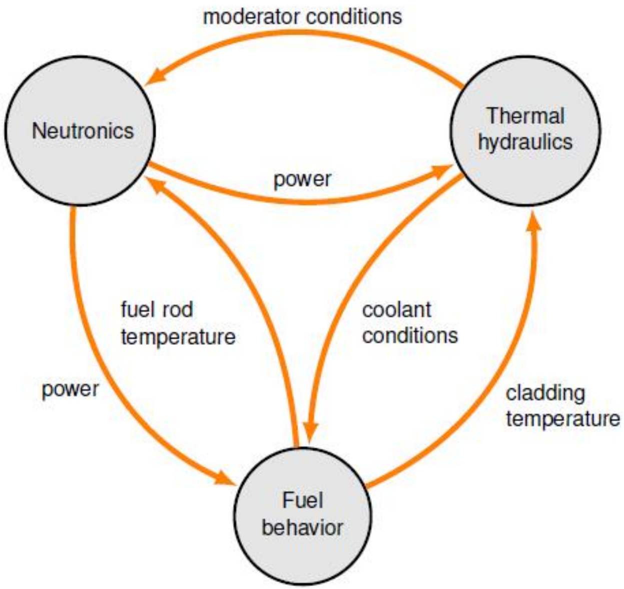
Highlights of the project

- Development of multiphysics fuel behavior module FINIX
- Participation in international benchmarks: OECD/NEA RIA Fuel Codes Benchmark and IAEA CRP Fuel Modelling in Accident Conditions
- Coupling between SCANAIR fuel performance code with GENFLO thermal hydraulics code
- Experimental studies on fuel and cladding materials

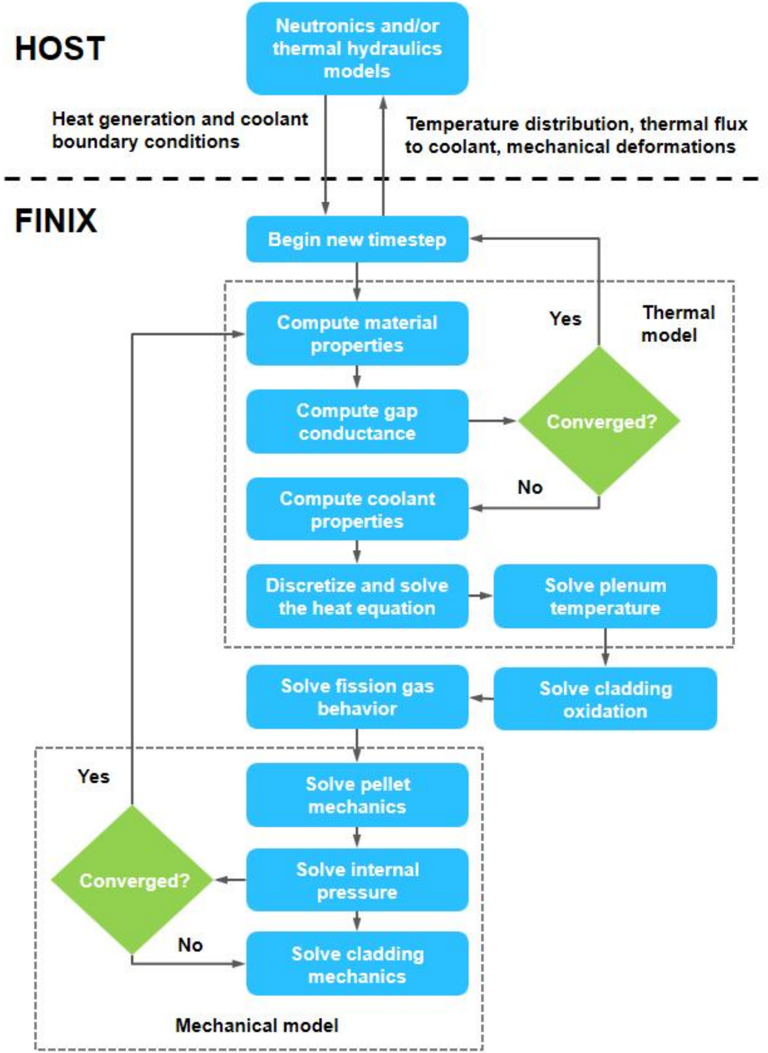
FINIX fuel behavior module

- FINIX development was begun in 2012 in SAFIR2014
- Aim to provide a reasonably accurate fuel behavior description in multiphysics couplings
- Neutron transport requires fuel temperatures for Doppler feedback
- Thermal hydraulics require the fuel temperature as boundary condition
- Typically these codes use simplified models for describing fuel temperatures

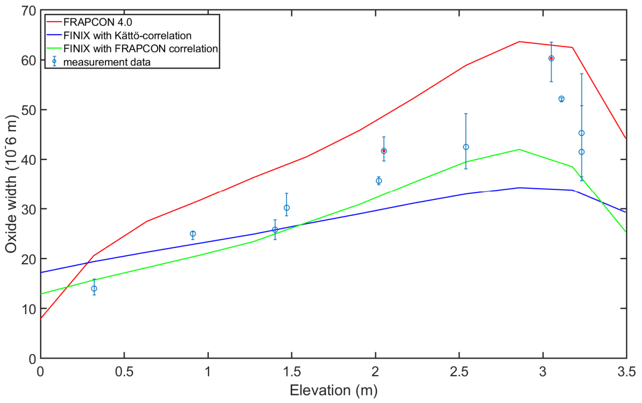


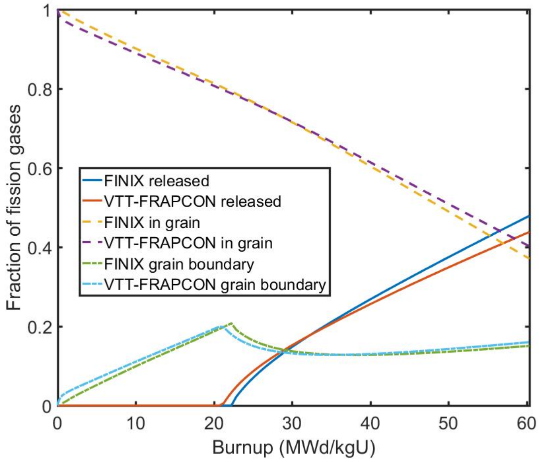
FINIX models

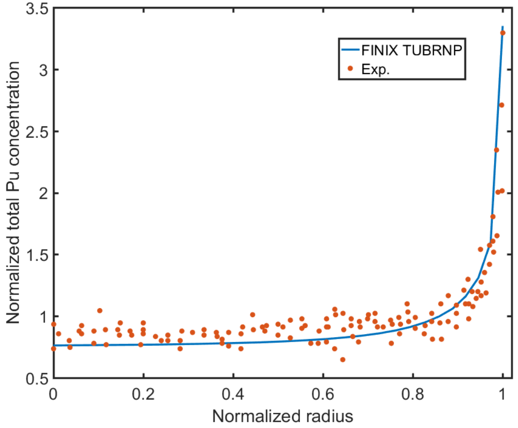
- FINIX solves the thermalmechanical behavior of a fuel rod
- In the beginning of SAFIR2018, FINIX was only capable of simulating transient scenarios, as no burnup-dependent models were present
- After the developments in this project, burnup effects in fuel can also be modeled due to several newly implemented models:
  - Pellet swelling and densification
  - Cladding plasticity, creep and oxidation
  - Radial power distribution within fuel
  - Fission gas release

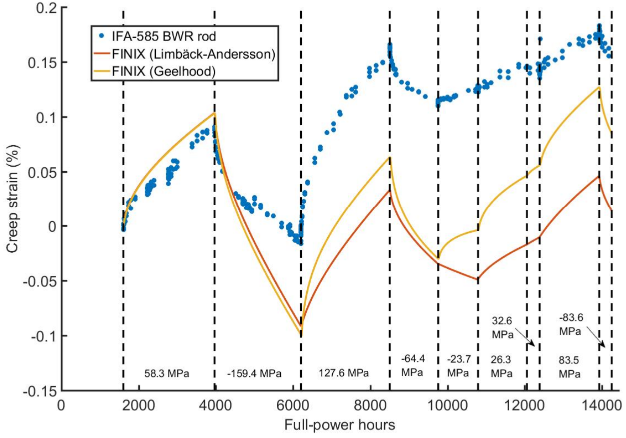


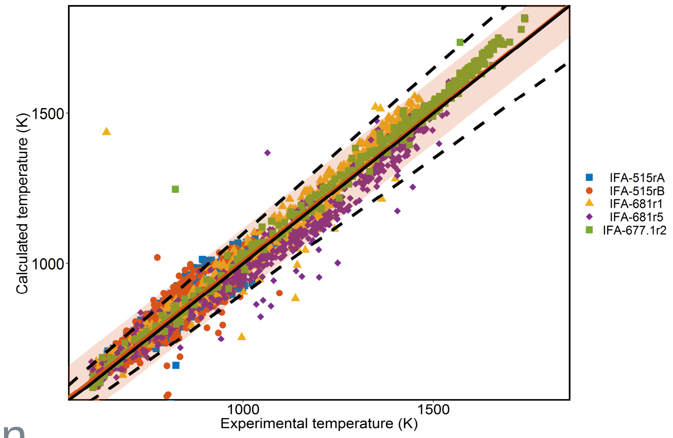
Validation and verification of implemented models





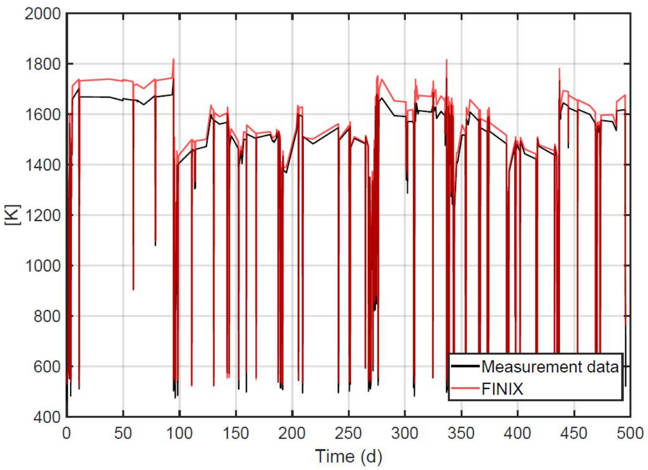




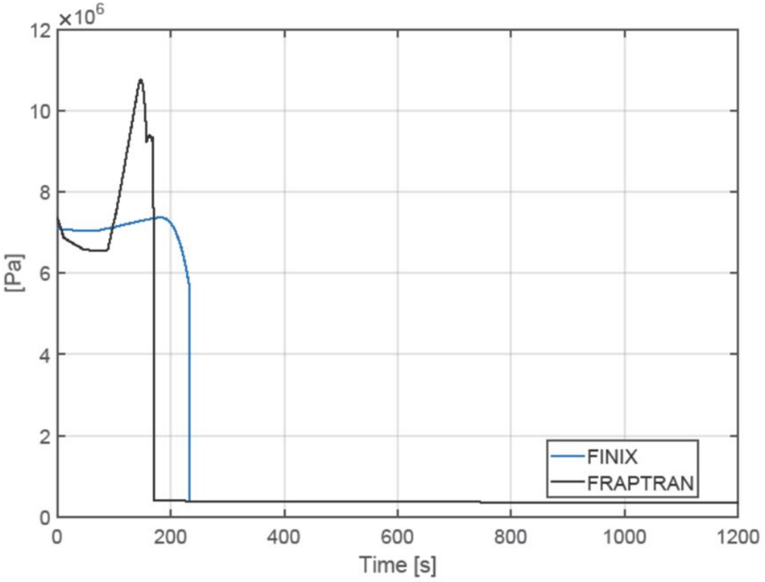


Integral validation

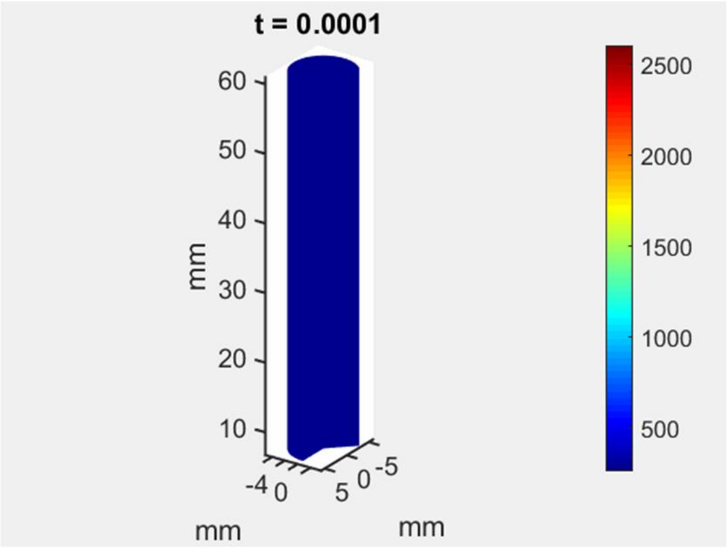
- FINIX was validated with integral fuel performance experiments
- Validation was done with the validation system SPACE developed within the project
- SPACE allows for automatic simulation of validation cases and summarizes and plots the results automatically
- FINIX temperature predictions are as good as state-of-the-art codes internationally (within 5-6% of experiment)



Application to design basis accidents



Rod internal pressure in IFA-650.5 LOCA test



Temperature in CABRI REP-Na3 RIA test

IAEA CRP FUMAC (1/3)

- VTT participated to the IAEA coordinated research program (CRP) FUMAC (Fuel Modeling in Accident Conditions) centered on the modeling of loss-of-coolant accidents
- In a loss-of-coolant accident, the fuel rod overheats and may burst due to internal pressure buildup
- Aim of the project is to support the participants from different countries in their efforts to develop reliable tools for modelling of fuel behaviour during LOCAs
- VTT participated with FRAPTRAN calculations

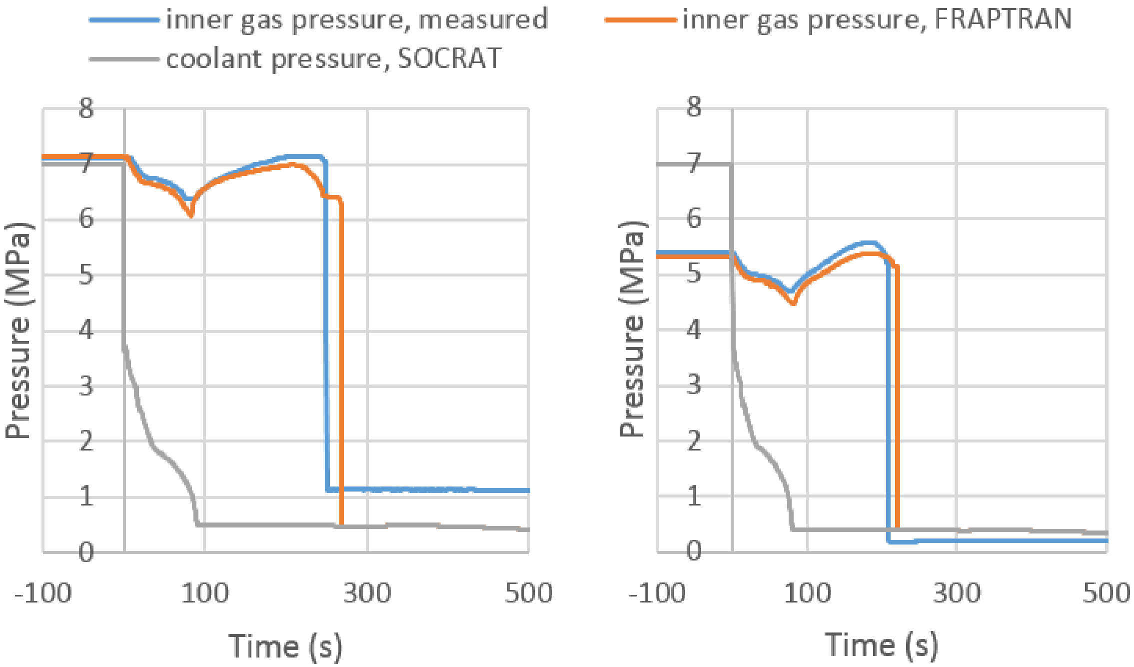
IAEA CRP FUMAC (2/3)

- The program consistend of benchmarking calculations between participants and experimental data from, for example, AEKI and Halden Reactor Project
- Ballooning tests on cladding tubes have been performed at AEKI in Hungary
  - Zircaloy-4 cladding tubes are pressurized at a specified temperature and the resulting cladding deformation measured
  - Slightly irregular behavior with VTT's FRAPTRAN-1.4 simulations
- The IFA-650 test series at Halden has consisted of integral LOCA tests performed in the Halden reactor
  - The tests IFA-650.10 and IFA-650.11 were simulated at VTT with FRAPTRAN-GENFLO and FRAPTRAN with coolant boundary conditions provided by the SOCRAT code

IAEA CRP FUMAC (3/3)

Left: IFA-650.10

Right: IFA-650.11



IAEA TECDOC to be published in the near future, some results can be found from the proceedings of the 2018 TopFuel conference in the FUMAC session (https://www.euronuclear.org/events/topfuel/topfuel2018/proceedings.htm)

OECD/NEA RIA Benchmark (1/3)

- In a RIA the fuel rod undergoes a fast (few to tens of ms) power pulse, which causes rapid fuel thermal expansion and stress to the cladding
- The first phase of the RIA Benchmark was organized based on a recommendation in a OECD/NEA RIA modeling workshop in 2009, and participated in SAFIR2014
  - RIA codes are widely used by the industry and TSOs, so a sound basis for the comparison of such codes should be obtained
- The first phase found that deeper understanding of the modeling differences in different codes should be obtained, and an uncertainty study and a sensitivity study of the results to the input parameters should be performed
- Work on these objectives were performed in phases 2 and 3 of the benchmark, during 2014-2018

See, for example, NEA/CSNI/R(2013)7 for phase 1, NEA/CSNI/R(2017)1 for phase 2, phase 3 to be published

OECD/NEA RIA Benchmark (2/3)

- First phase: Simulation of several RIA tests from test reactors with irradiated rods, but large scatter was present in the results
- Second phase: Uncertainty and sensitivity studies for unirradiated, simplified cases to obtain information on code modelling and treatment of input data
- Third phase: Uncertainty and sensitivity studies for irradiated rods, with the same irradiated state provided to each participant

OECD/NEA RIA Benchmark (3/3)

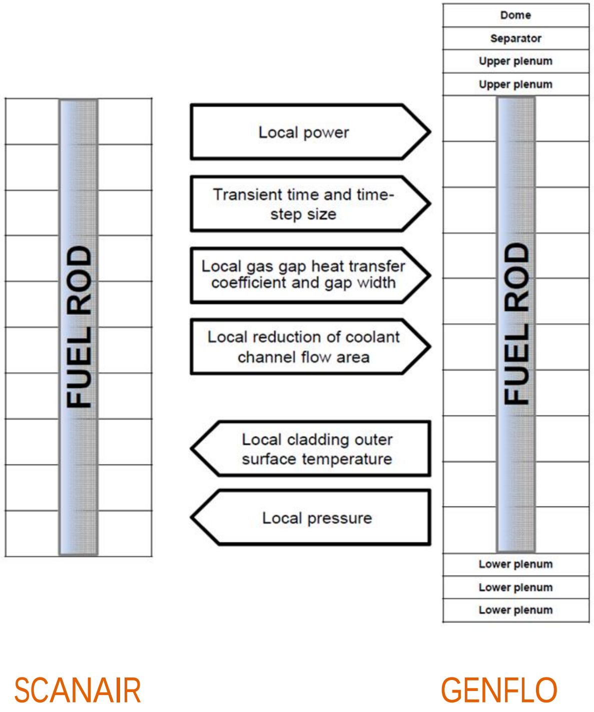
- Illustrative results from the sensitivity study results in Phase 3
- Partial rank correlation coefficients were used: Pearson correlation coefficient used on the ranks of the data
- Colour indicates different absolute values of partial rank correlation coefficients: blue is over 0.25, red is over 0.75 while the range is from 0 to 1.

	Output				
	Thermal behaviour		Mechanical behaviour		Fis-sion gas re-lease
Input parameter	Fuel	Cladding	Fuel	Cladding	
Pellet-cladding radial gap					
Cladding roughness					
Fuel roughness					
Cladding outer oxide thickness					
Injected energy in the rod					
Radial power profile					
Power pulse width					
Fuel thermal conductivity model					
Fuel thermal expansion model					
Fuel enthalpy					
Cladding thermal expansion					
Cladding yield stress					

Sensitivity indicators at the time of maximum power pulse

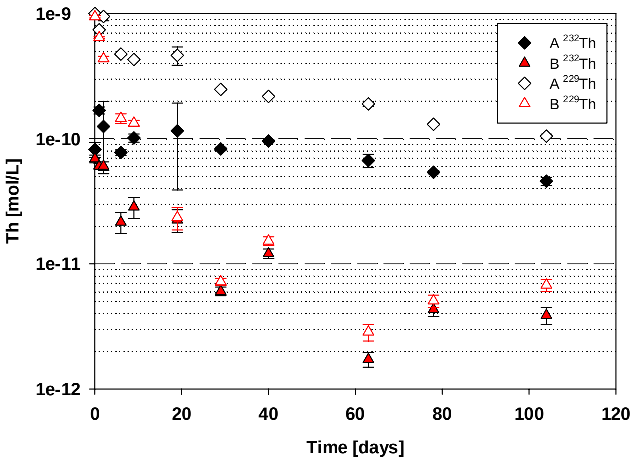
SCANAIR-GENFLO coupling

- SCANAIR is a fuel performance code designed for simulating reactivity insertion accidents (RIA), developed by IRSN in France
- SCANAIR is designed for PWRs, and the in-code thermal hydraulics models are not able to simulate BWRs
- SCANAIR was coupled with VTT’s thermal hydraulics code GENFLO to be able to simulate RIA’s in BWR conditions
- Demonstration calculations on RIA benchmark hot zero power case with and without boiling and two RIA tests performed in the NSRR reactor



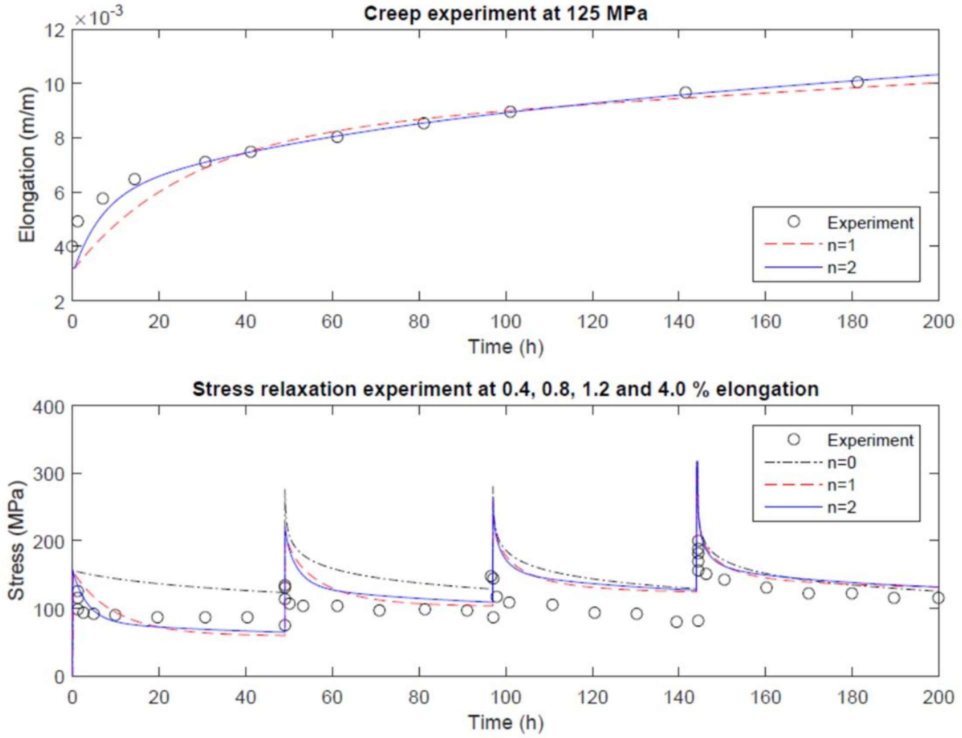
Thorium dissolution experiments

- Thorium dioxide has a similar crystal structure as uranium oxide, and has been planned to be used as a nuclear fuel by itself
- Dissolution studies on thorium dioxide were performed to investigate the initial dissolution of thorium with the help of a Th-229 tracer
- It was found that Th-232 from the oxide dissolved into the aqueous phase even though the solution was already saturated with thorium from the tracer
- The early stage of dissolution was controlled by the stability of surfaces rather than chemical equilibrium



Cladding creep modeling

- A new viscoelastic creep model was developed in the thesis by Tulkki
- The creep model is better at describing stress changes and reversals than traditional models
- Has been implemented into the BISON fuel performance code at Idaho National Laboratory



V. Tulkki, D.Sc.(Tech.) thesis, Aalto University, 2015. BISON implementation: M. Zahoor, JNM 511 (2018) 212-219.

Cladding creep experiments

- The VTT experimental capabilities for performing creep experiments under transient conditions were investigated
- Two E110 specimens were investigated with temperature transients and several stress transients (figure from the latter)