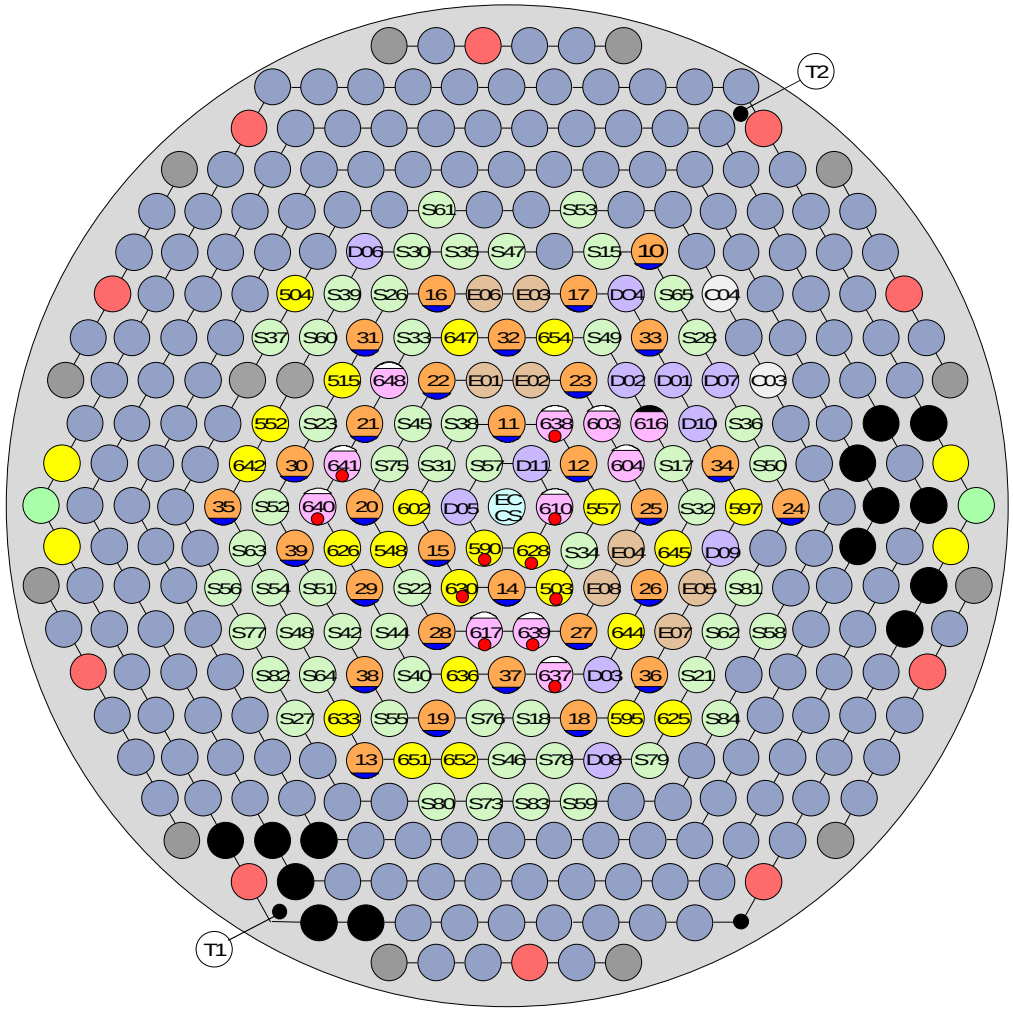
Thor Energy Th/Pu test (IFA-730): In-pile data from the first irradiation cycle

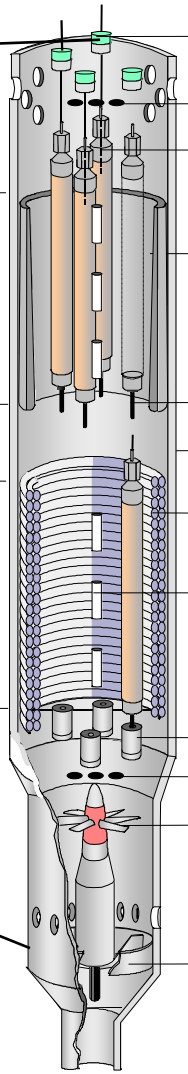
Presented by C. Vitanza, IFE/Halden

The Halden Reactor - HBWR



- 20 MW heavy water reactor at 34 bar and 235°C
- Height of active core is 80 cm
- Hexagonal array of 300 channels within pressure tank
- Rings 1-6 used for experiments (about 30/110 positions used)
- Experimental channels directly in HBWR (71 mm in diameter)
- 8-10 loops system in operation at any given time (32 to 45 mm in diameter)

Instrumented Test Rigs



In-core Connector

Outlet Coolant Thermocouples

Fuel Centre - line Thermocouple

Neutron absorber

Fuel Rod storage position

Pressure transducer (PF)

Shroud (Ø 73/71mm)

He-3 coil

Fuel Rod ramp position

Neutron Detector (V-type)

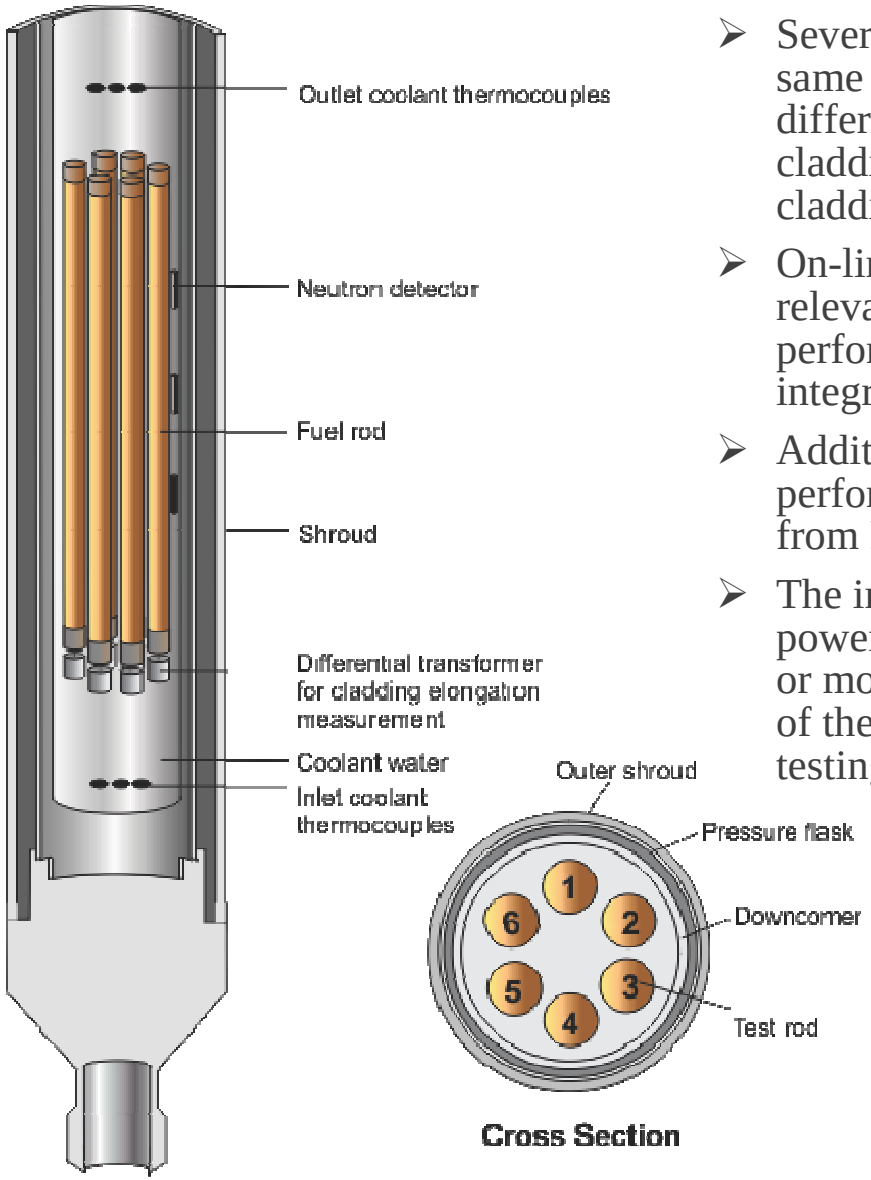
Differential Transformer (LVDT)

Inlet Coolant Thermocouples

Inlet Turbine Flowmeter

Calibration Valve

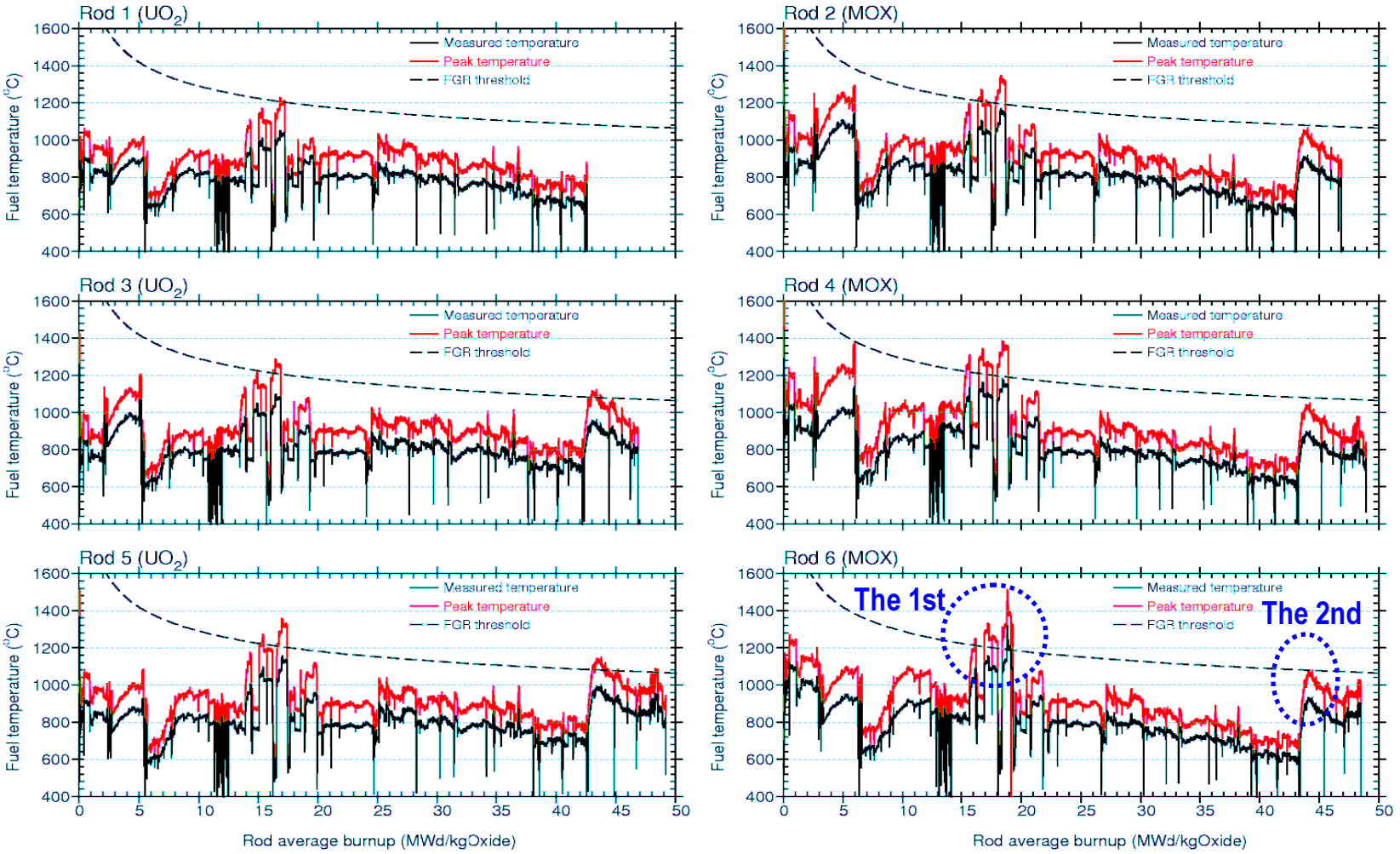
Integrity of fuel rods at base irradiation conditions



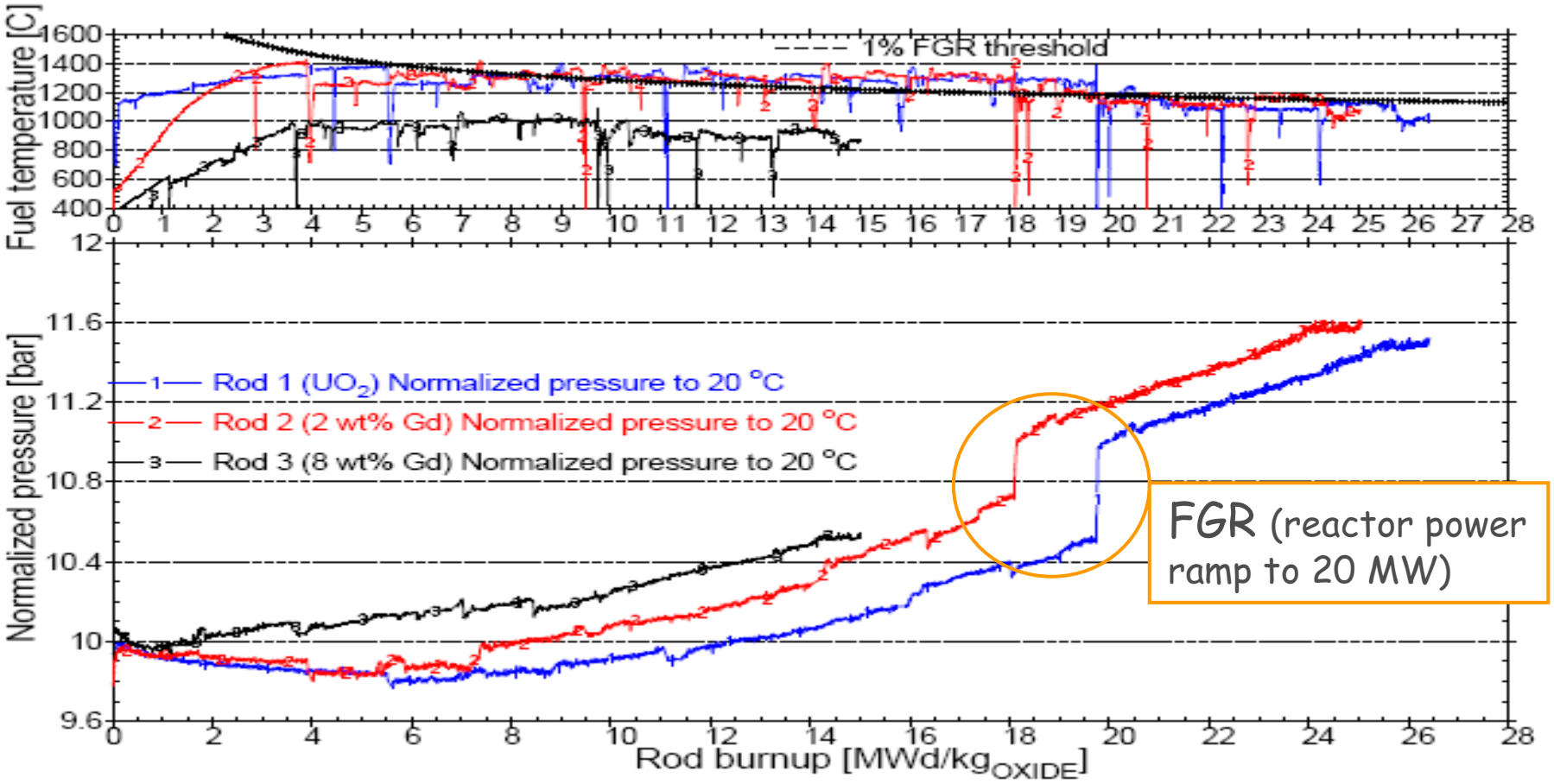
- Several fuel rods can be tested at the same time, hence enabling to compare different fuel rod variants, such as cladding type, fuel pellet type, pellet-cladding gap etc.
- On-line instrumentation provides relevant information on the fuel rod performance, e.g. on PCMI and rod integrity (and also FGR if needed)
- Additional information on other performance aspects can be obtained from PIE (dimensional & corrosion)
- The irradiation is carried out at constant power and can last for a long time (one or more years). After that, one or some of these rods can be subjected to special testing, e.g.
  power ramp or even LOCA.

Temperature histories

- Peak fuel temperatures were close to fission gas release threshold temperatures during the 1st and 2nd interlinkage tests.



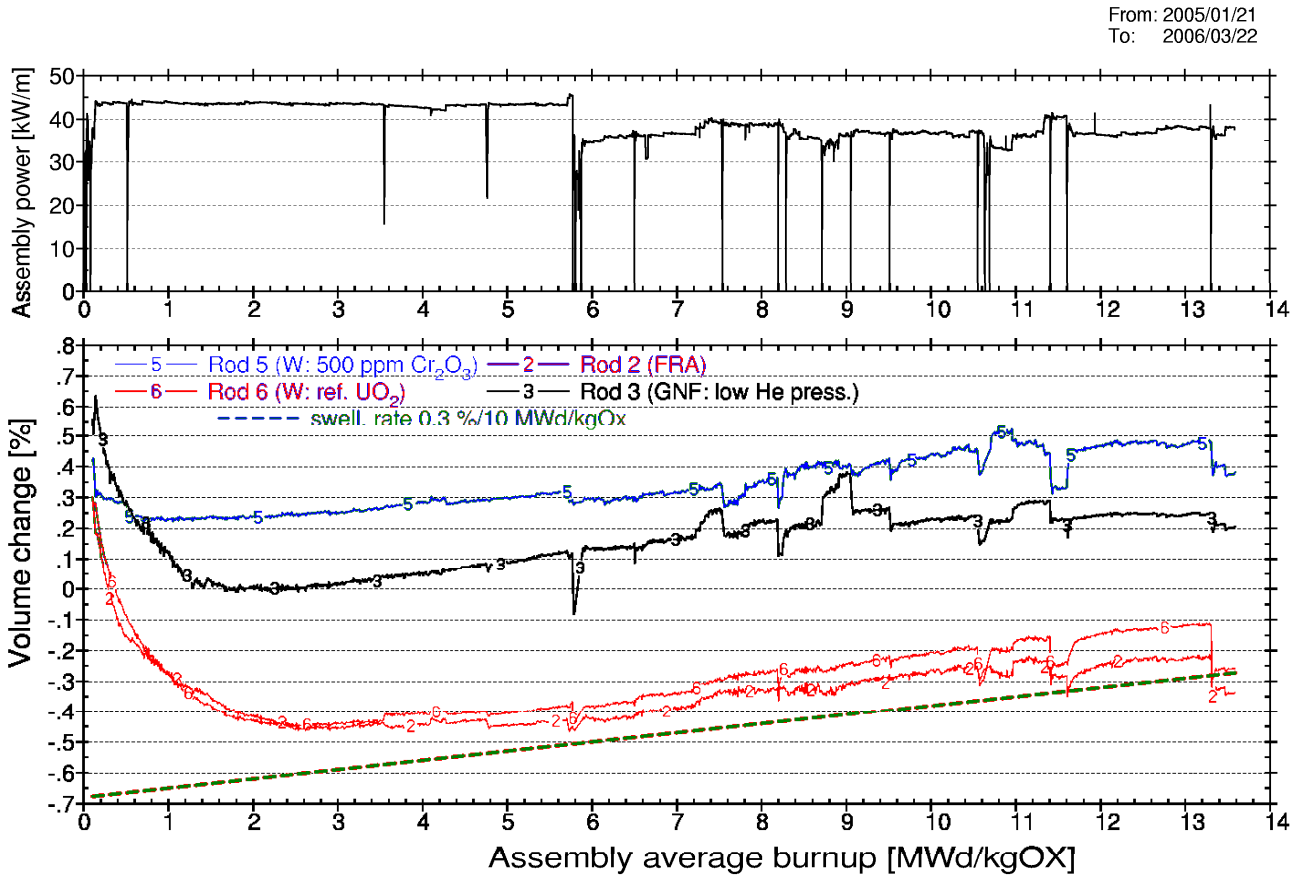
Rod pressure (fission gas release)



*Signs of densification in rods **1** and **2***

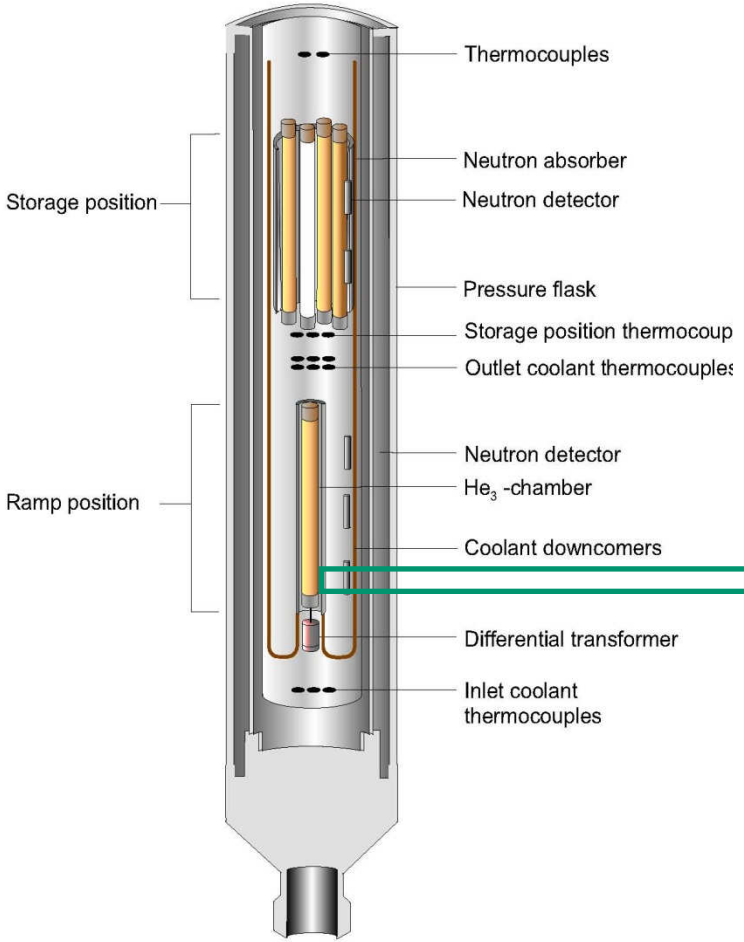
*Different behaviour for **8 %wt Gd** because $T < T_{threshold}$*

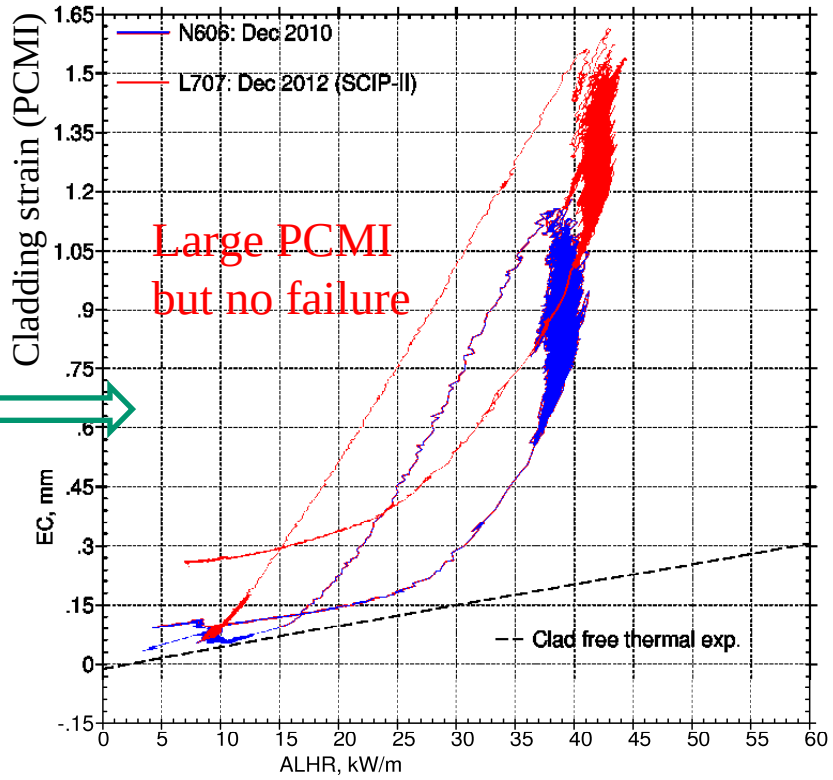
Integral test: Fuel pellet densification and swelling



Integrity of fuel rods in power ramp conditions

Assessment of PCI and PCMI margin

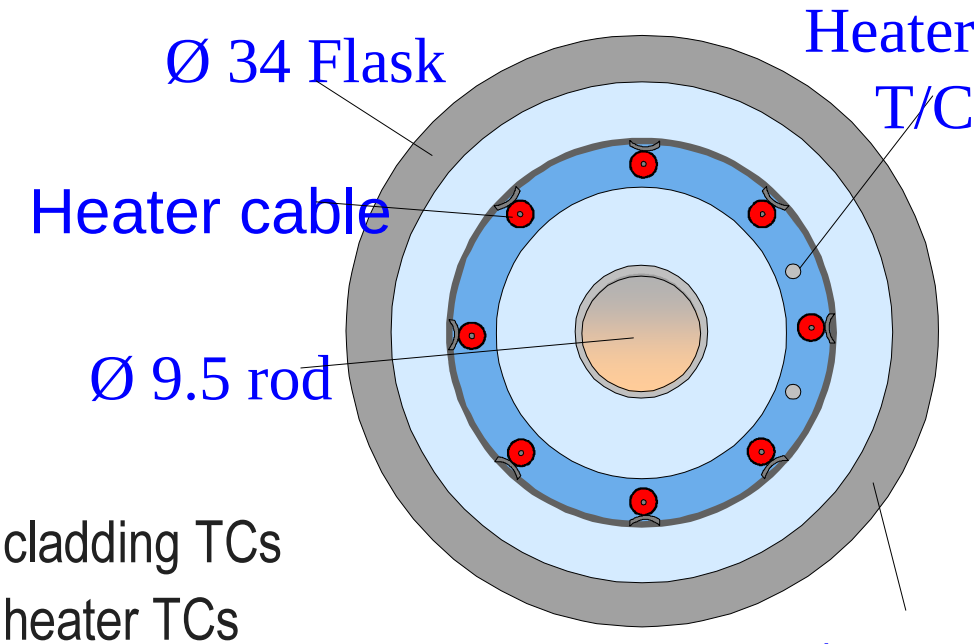




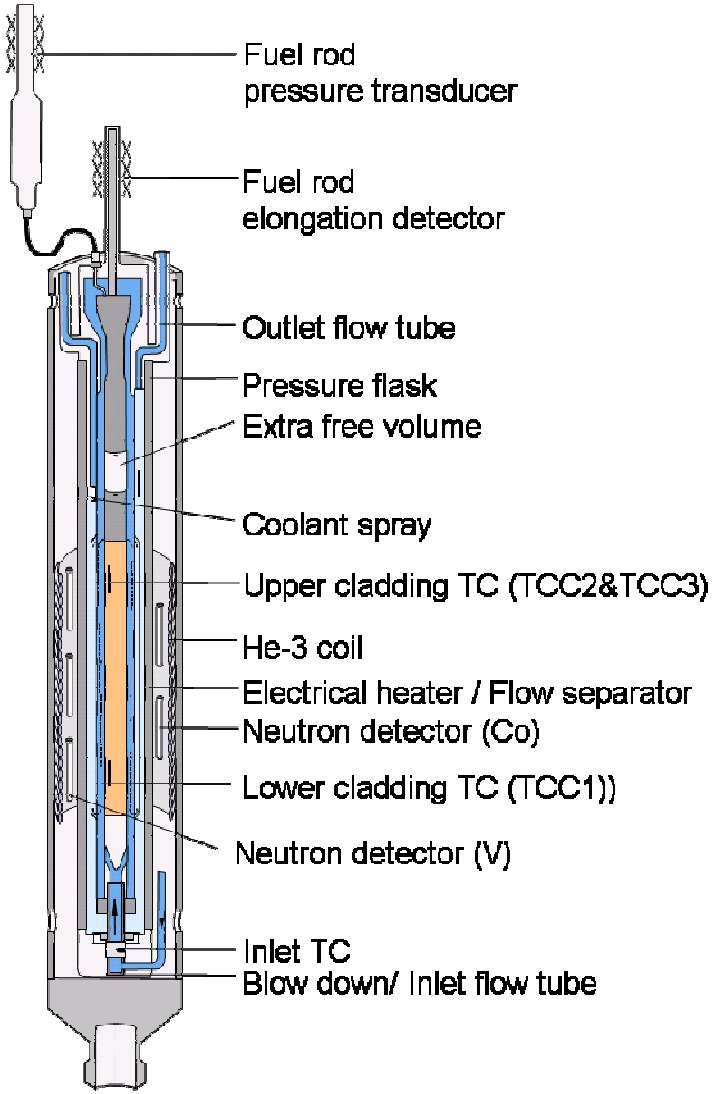
Example of failed rod

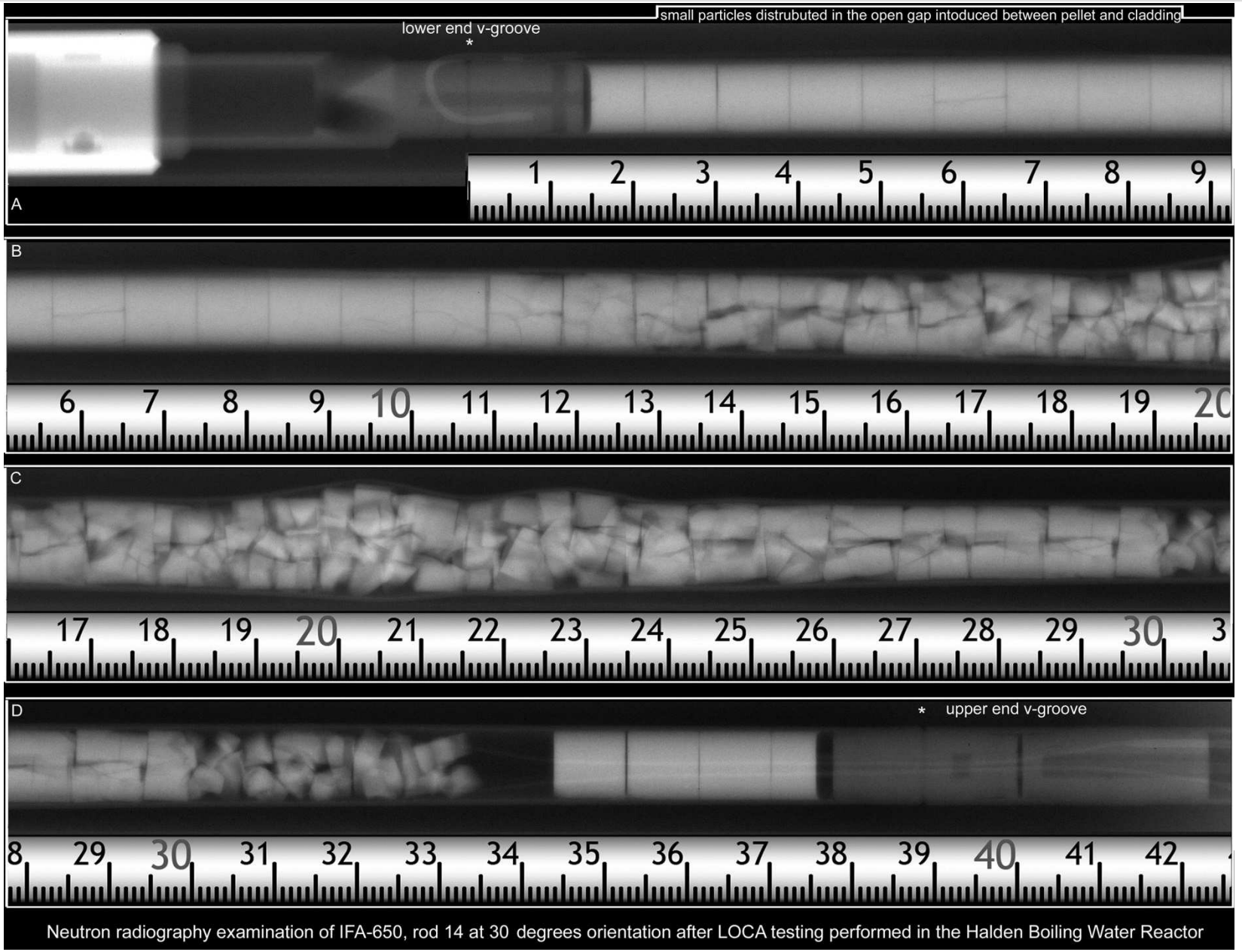
Our previous experience: IFA-650

- Single rod experiment using high burnup fuel
- Heating provided from within the rod by low level of nuclear power simulating decay heat



- 3 cladding TCs
- 3 heater TCs
- Fuel pressure transducer
- Cladding elongation detector





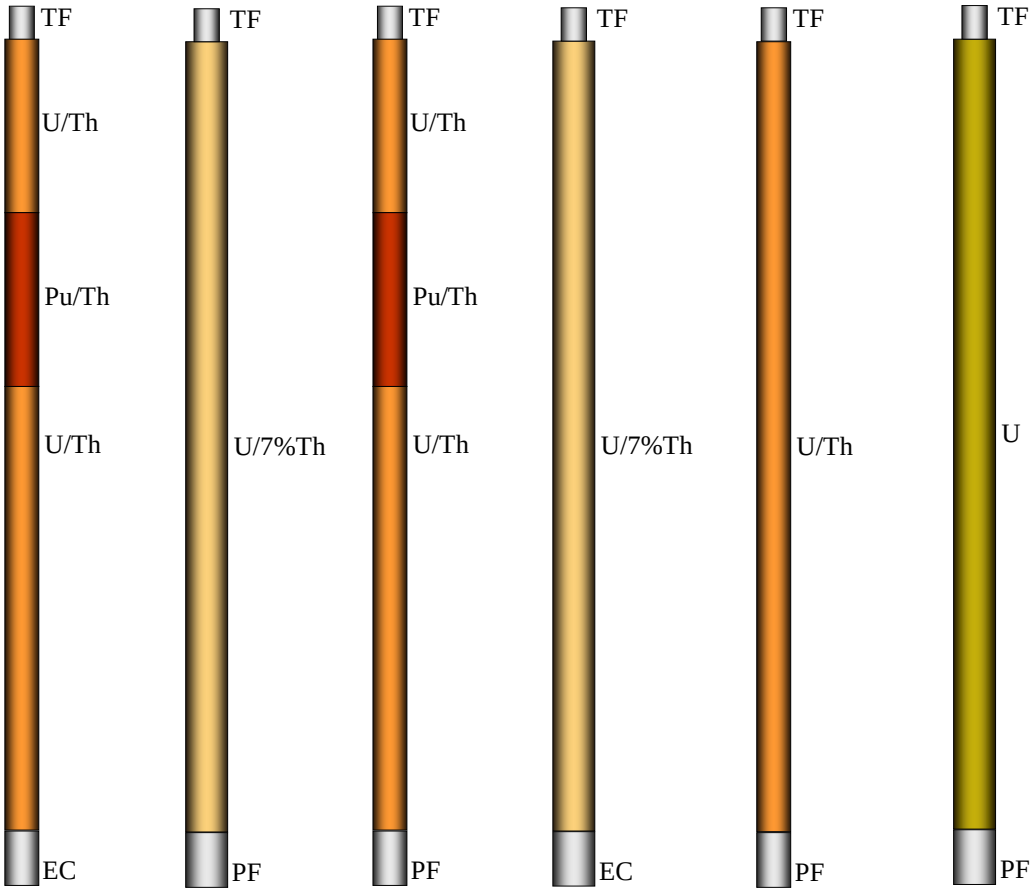
Neutron radiography examination of IFA-650, rod 14 at 30 degrees orientation after LOCA testing performed in the Halden Boiling Water Reactor

IFA-730.1 Status

- Rig operated since 28.4.2013
  - ~150 operation days so far
  - Power kept at ~30 kW/m for large diam. rods
~20 kW/m for small diam. rods
  - Measured temperature kept below ~1200° C
    - In order to keep long-term operation of the TFs
- Power calibration on 28.4.
  - Good agreement with pre-calculation (±3%)

- Behaviour of Th fuel rods as expected

IFA-730
Test Matrix – Phase 1a

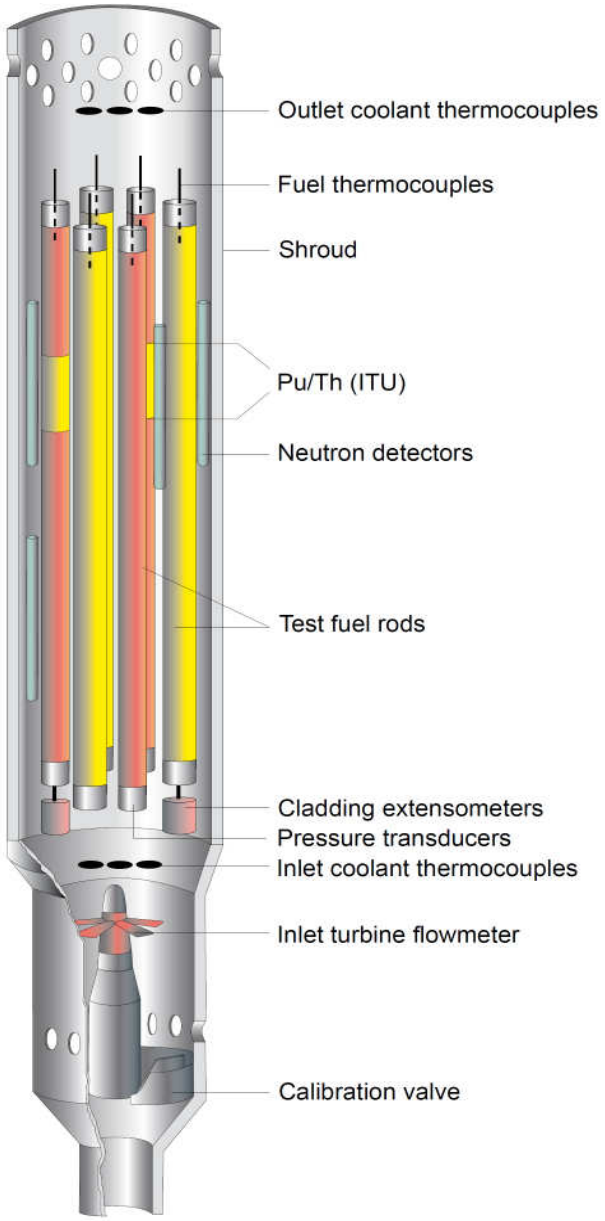


PF – Pressure transducer

TF – Fuel thermocouple

EC – Cladding elongation detector

IFA 730 test rig



IFA-730.1 Instrument status

- From 29.4. Rod pressure behaviour in UO_2 rod not as expected
  - Early on only weak or no response to power changes
    - Mechanical friction
  - Strong drop on 3.5.
  - After this, the instrument shows reasonable response to power and temperature changes and is likely to show clearly when FGR occurs
- On 3.5. a jump of ~50° C in TF2
  - Signal check showed no indication of failure
  - Subsequently following operation as expected
- Sudden drop in PF5 signal on 3.6.
  - Signal check showed water penetration into the cable
  - Instrument faulty

IFA-730 Test matrix

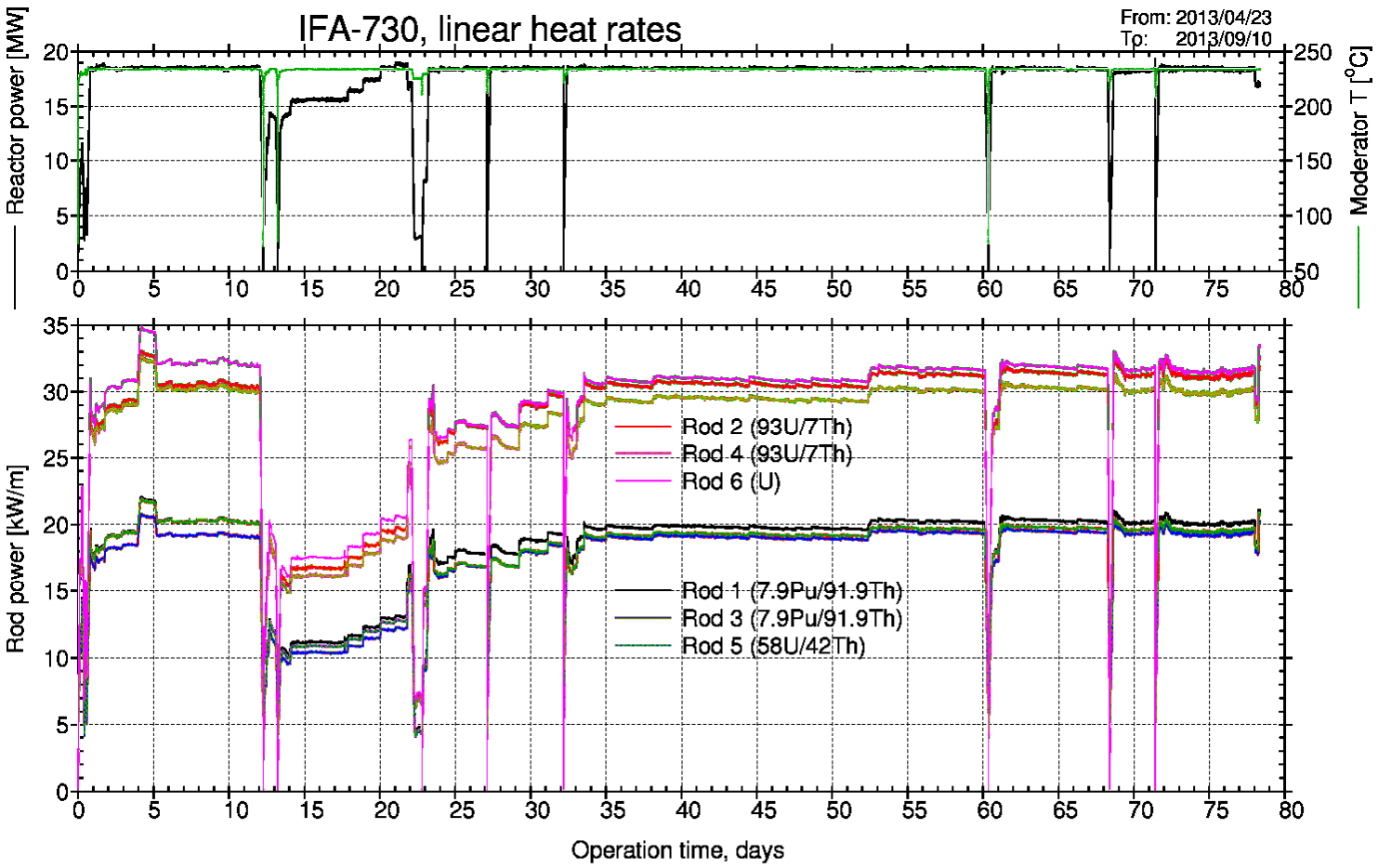
Rod ID	730-1	730-2	730-3	730-4	730-5	730-6
Fuel	58% U / 42% Th 8% Pu / 92%Th (OMICO pellets)	93% U / 7% Th	58% U / 42% Th 8% Pu / 92%Th (OMICO pellets)	93% U / 7% Th	58% U / 42% Th	UO_2
Pellet OD [mm]	5.90	8.48	5.90	8.48	5.90	8.48
Diam. gap [μm]	125	150	125	150	125	150
Instr.	TF / EC	TF[1)] / PF	TF / PF	TF / EC	TF / PF[2)]	TF / PF[3)]
Power [kW/m]	20	32	20	30	20	32
Burnup [MWd/Ox]	6.2	3.9	6.0	3.8	6.1	3.9

[1)] Thermocouple TF2 showed a jump on 3.5. – continued working normally.

[2)] Pressure transducer PF5 faulty since 3.6. – wet cable

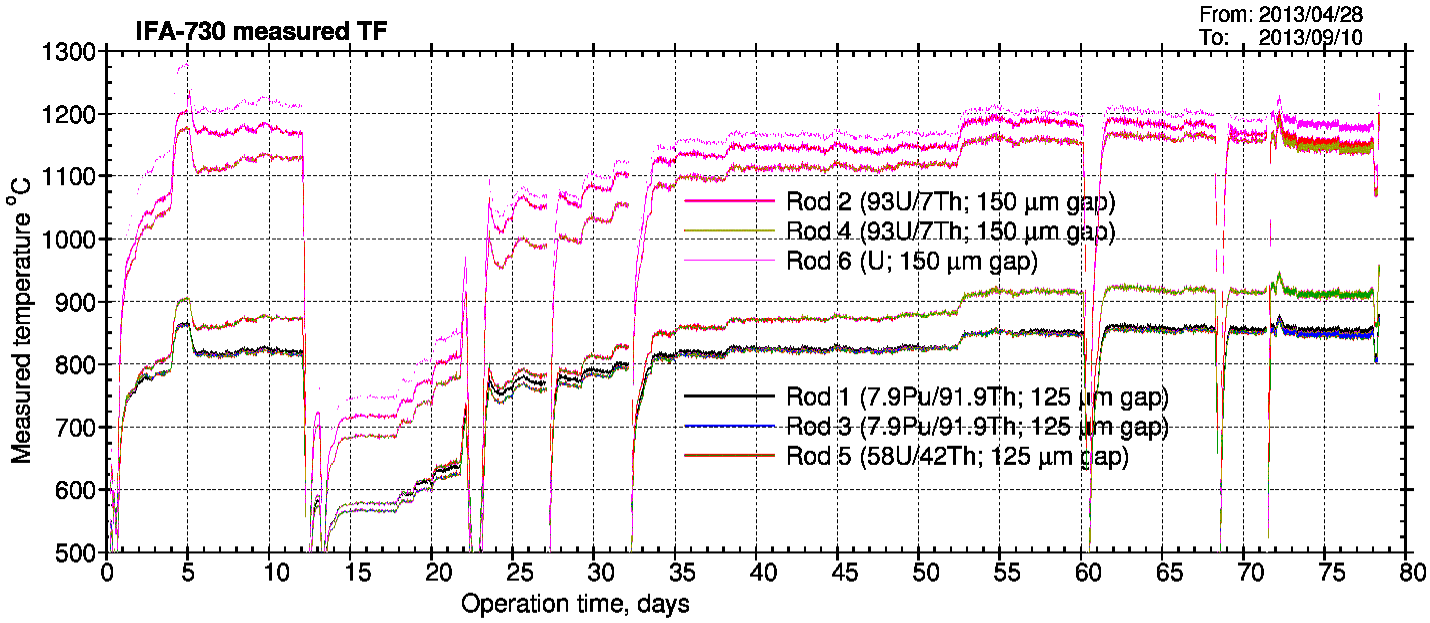
[3)] Pressure transducer PF6 unreliable at the beginning – likely to show FGR

IFA-730 Power levels



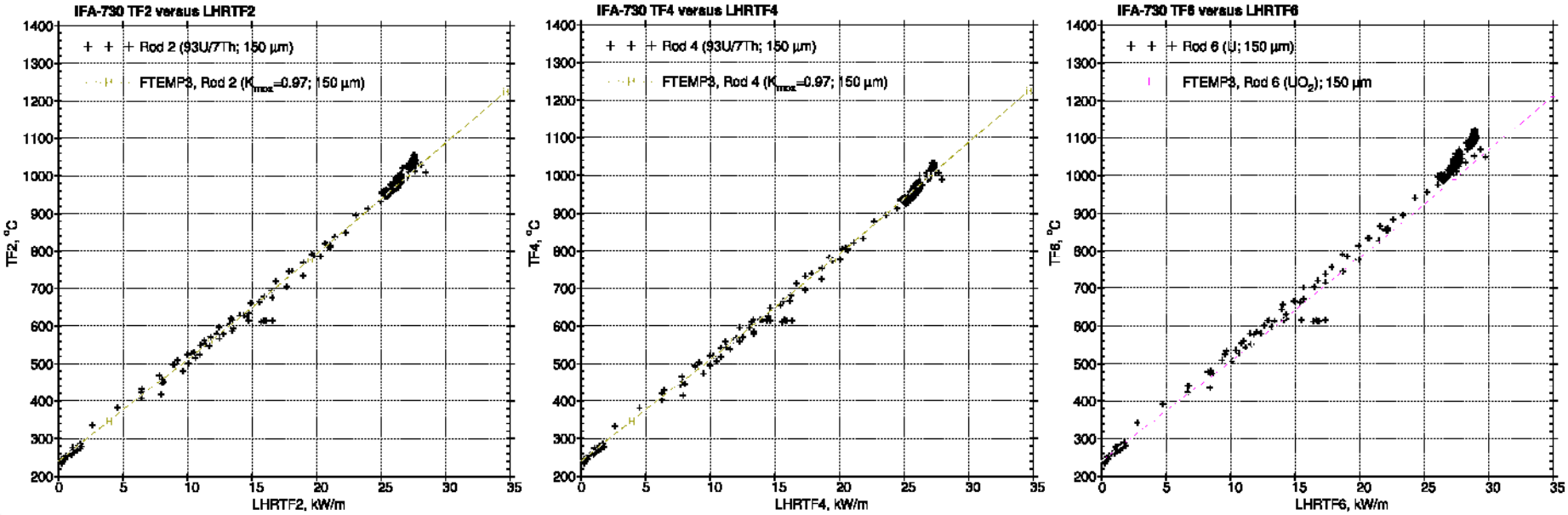
IFA-730 Fuel temperature

- Expectation:
  - Temperature in ThU rods lower or similar to reference UO_2 → confirmed
  - Temperature in PuTh pellets lower than in reference ThU → confirmed
- Further irradiation will provide information on long term behaviour



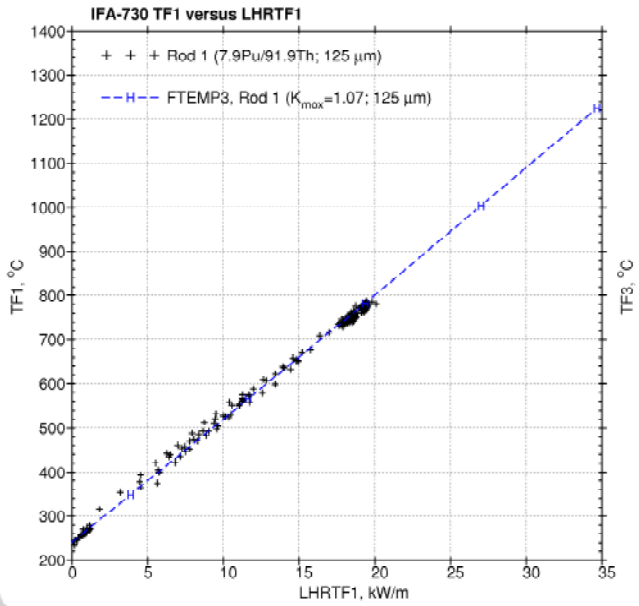
IFA-730 Results: Fuel temperature

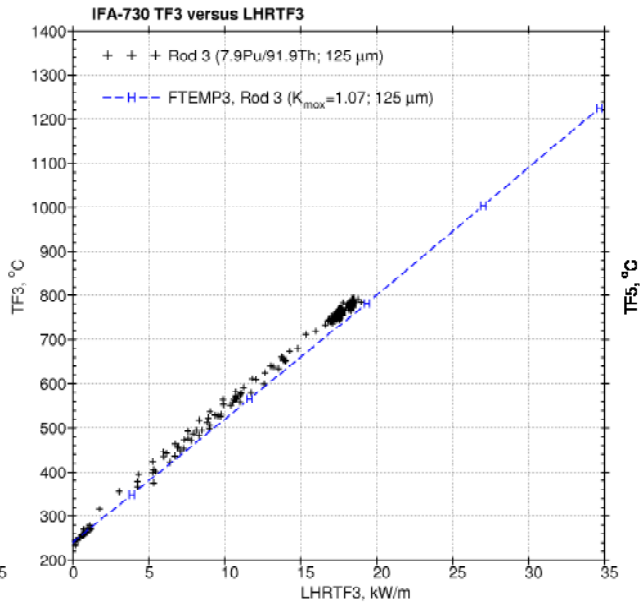
- Comparison with model predictions, start-up data: large diameter rods
  - In general good agreement between measured temperature and model predictions
  - Temperature in Th fuel slightly lower than reference UO_2 fuel

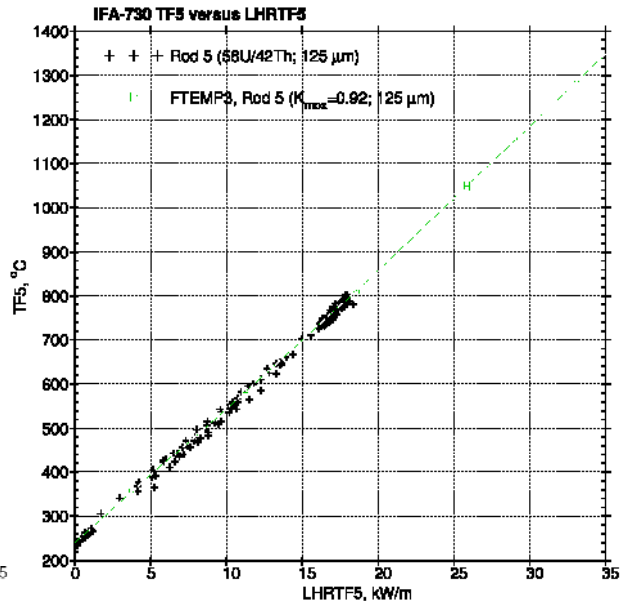


IFA-730 Results: Fuel temperature

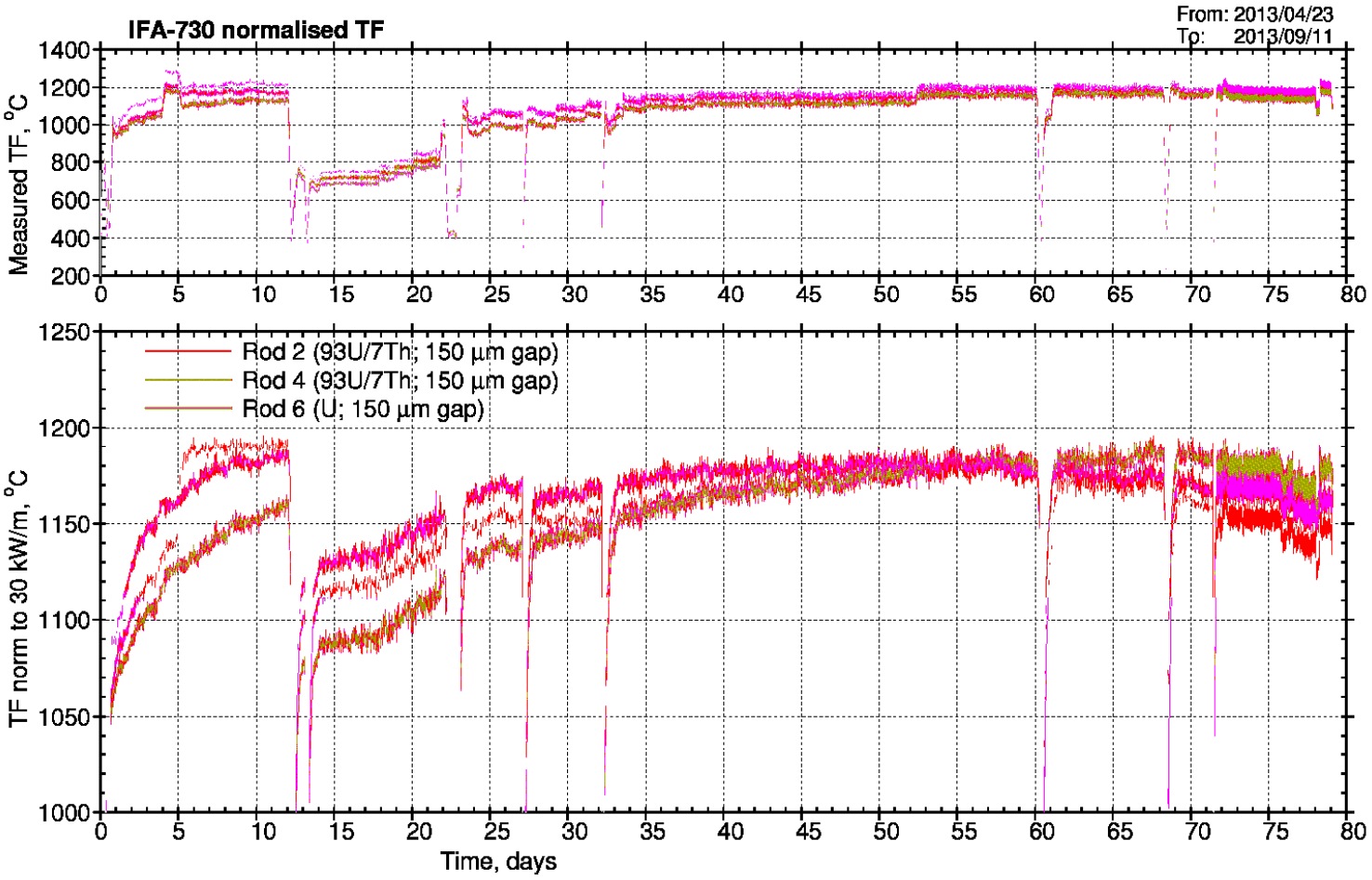
- Comparison with model predictions, start-up data: small diameter rods
  - In general good agreement between measured temperature and model predictions
  - Temperature in Th/Pu fuel slightly lower than Th/U fuel



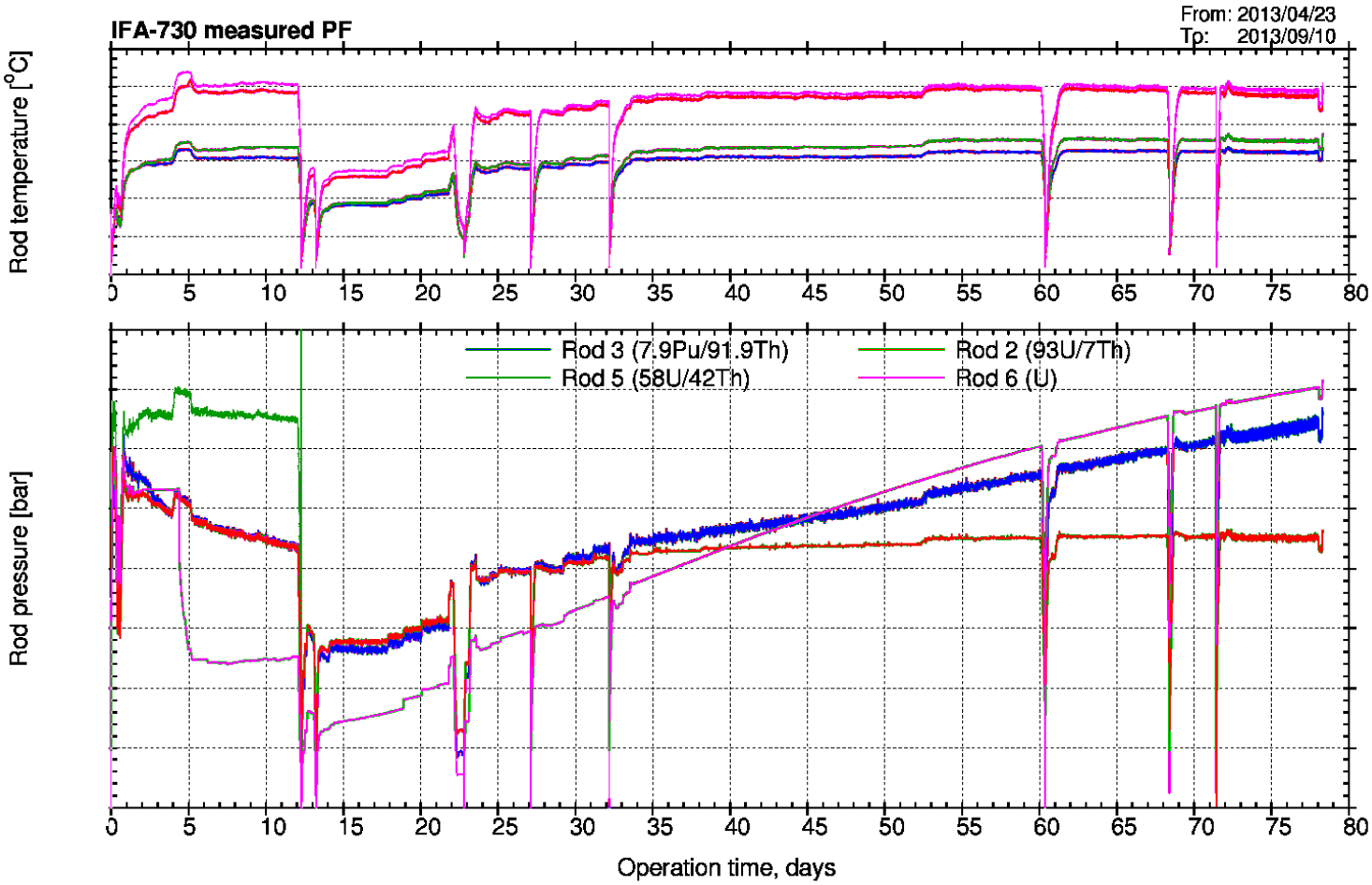




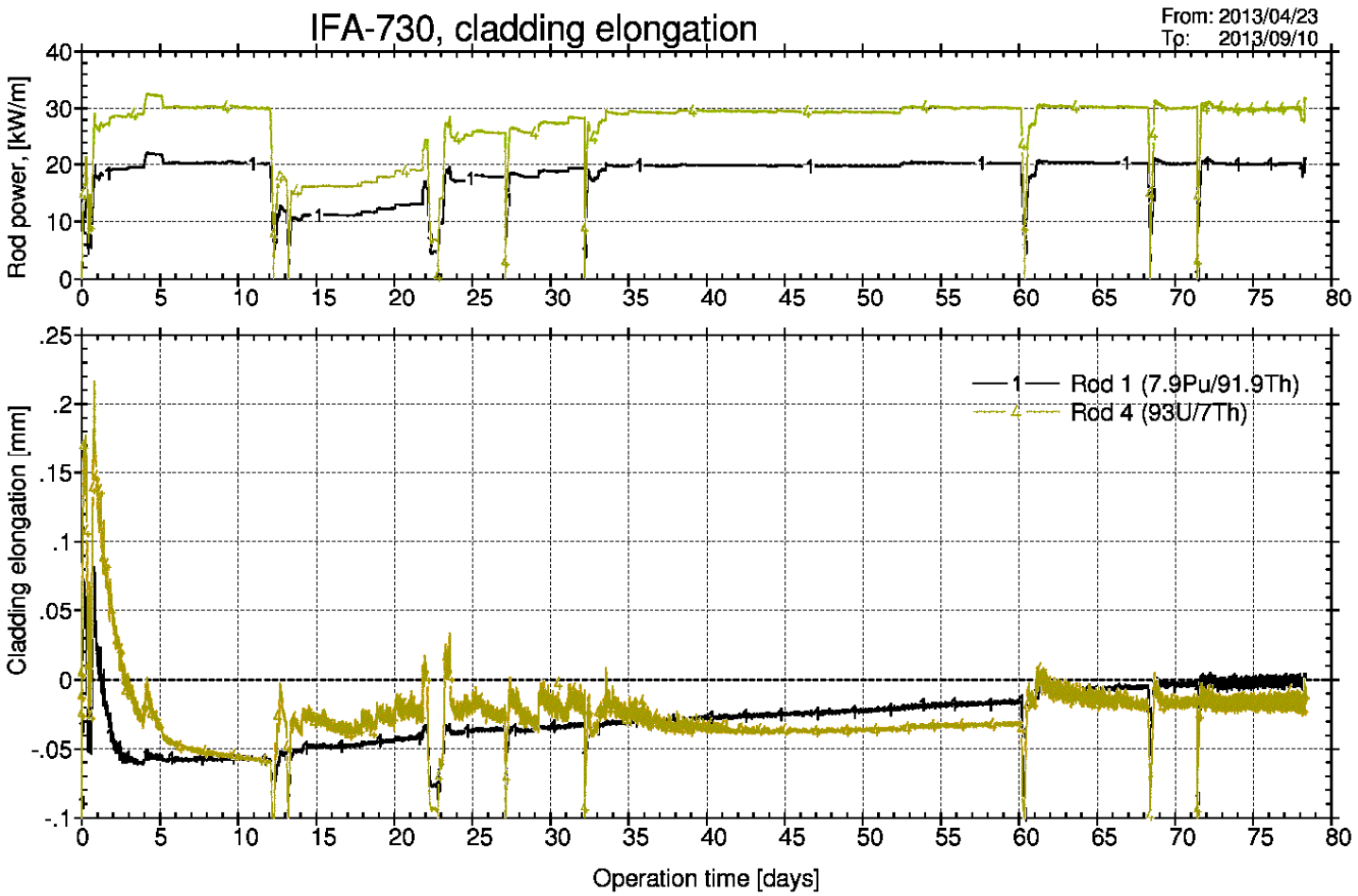
IFA-730 Normalised temperature 1/2



IFA-730 Fuel rod pressure



IFA-730 Cladding elongation



IFA-730.1 Summary 1/2

- Rig operated since 28.4.2013
  - Power kept at ~30 kW/m for large diam. rods
    ~20 kW/m for small diam. rods

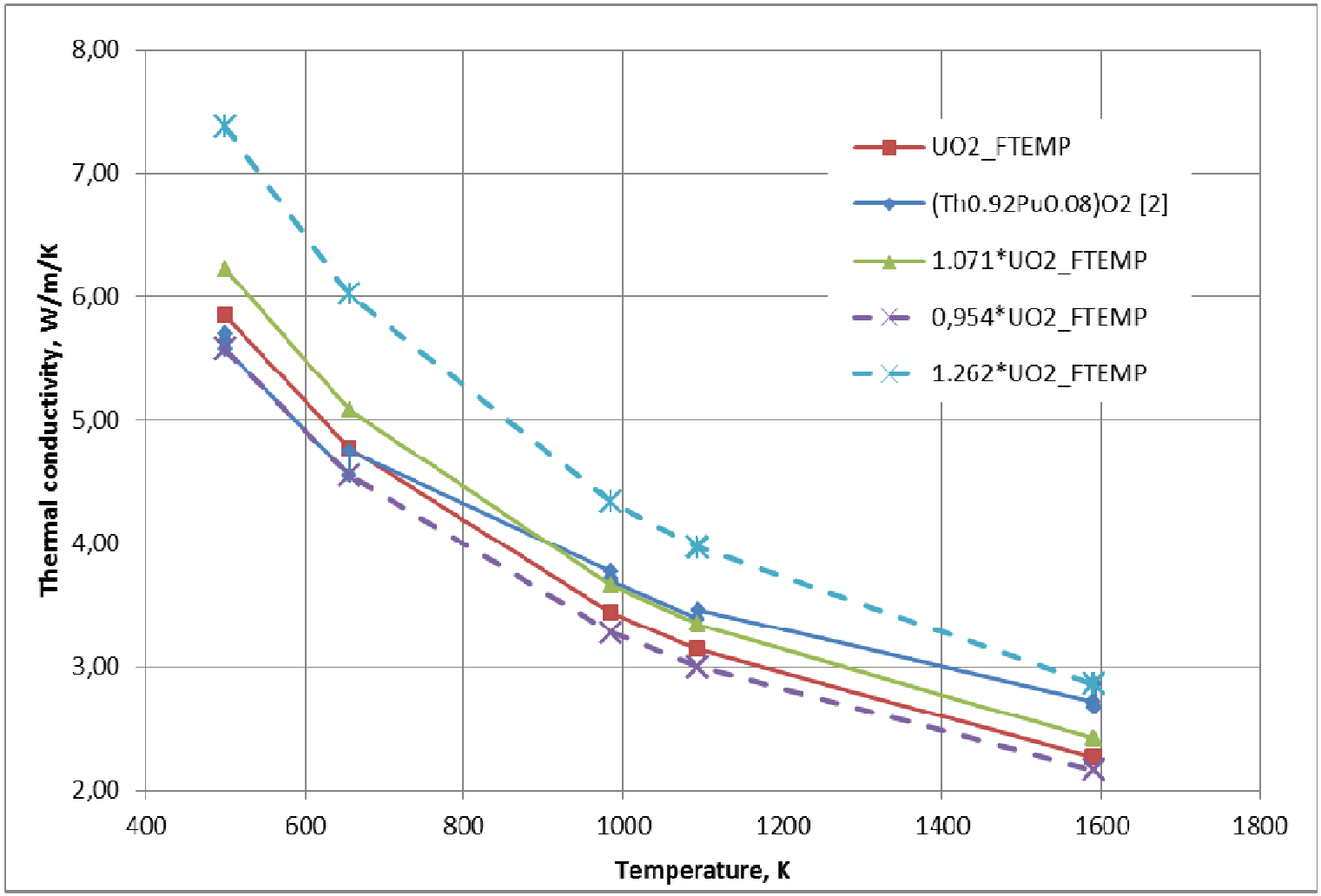
- Collection of irradiation data on
  - **Fuel centre temperature** **Fuel thermal conductivity**
  - Rod pressure Fuel dimensional stability and fission gas release
  - Cladding elongation Pellet-cladding mechanical interaction (fuel dimensional stability)

- Continued operation planned at current power levels

IFA-730.1 Summary 2/2

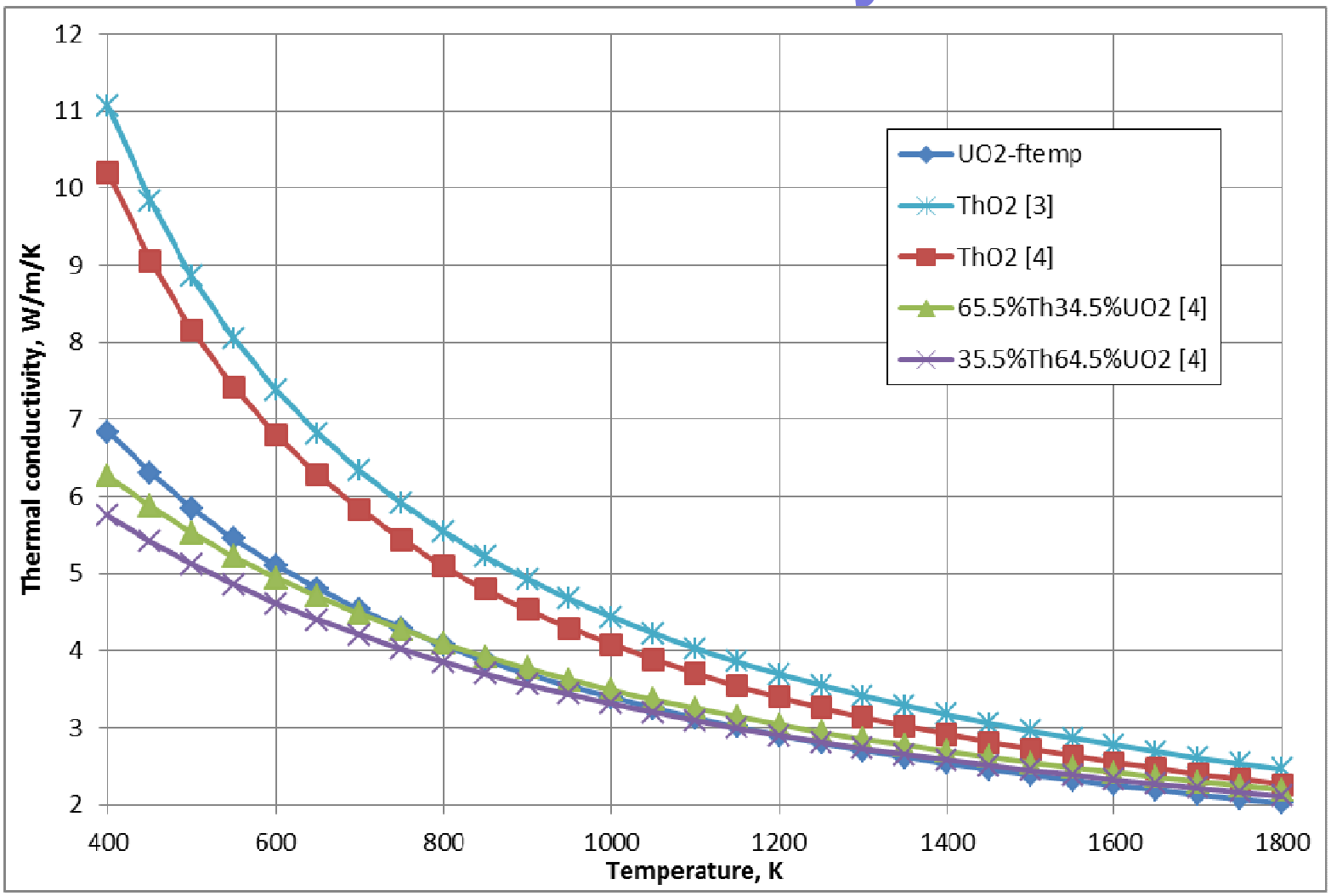
- Data collected so far indicates that Th fuel behaves according to expectation
  - Fuel centre temperatures during first start-up are in fair agreement with model predictions for fresh fuel
    - Temperatures in Th fuel lower or similar to reference UO_2
      - As expected
    - Further irradiation will provide information on long term behaviour
  - Data collected during the further irradiation may provide information about fission gas release behaviour (rod pressure)
- Current data will be analysed more carefully during the upcoming reactor outage

Thermal conductivity of Pu/Th



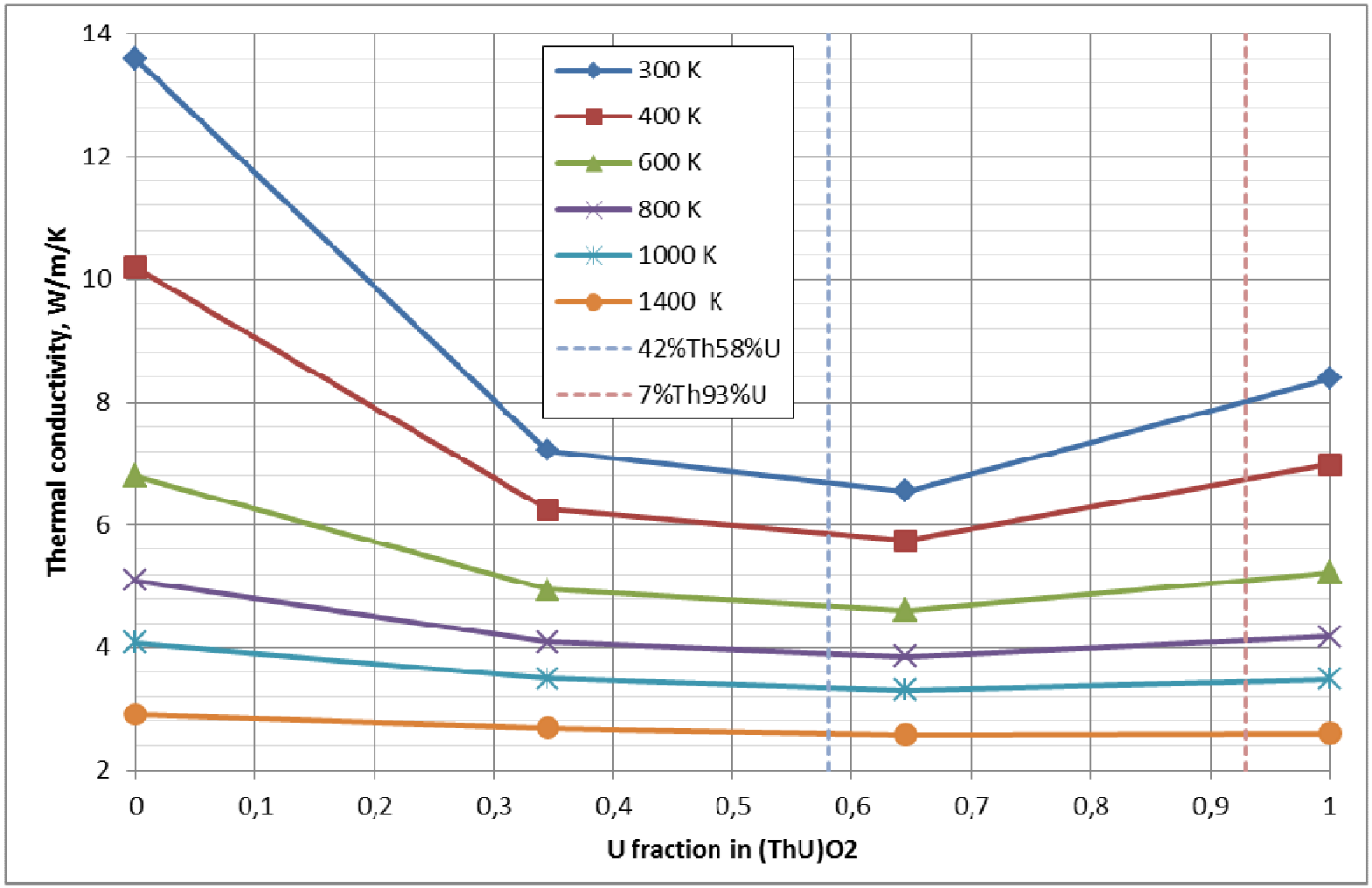
- C. Cozzo et al. , Journal of Nuclear Materials 416, 135-141, 2011

Thermal conductivity of U/Th



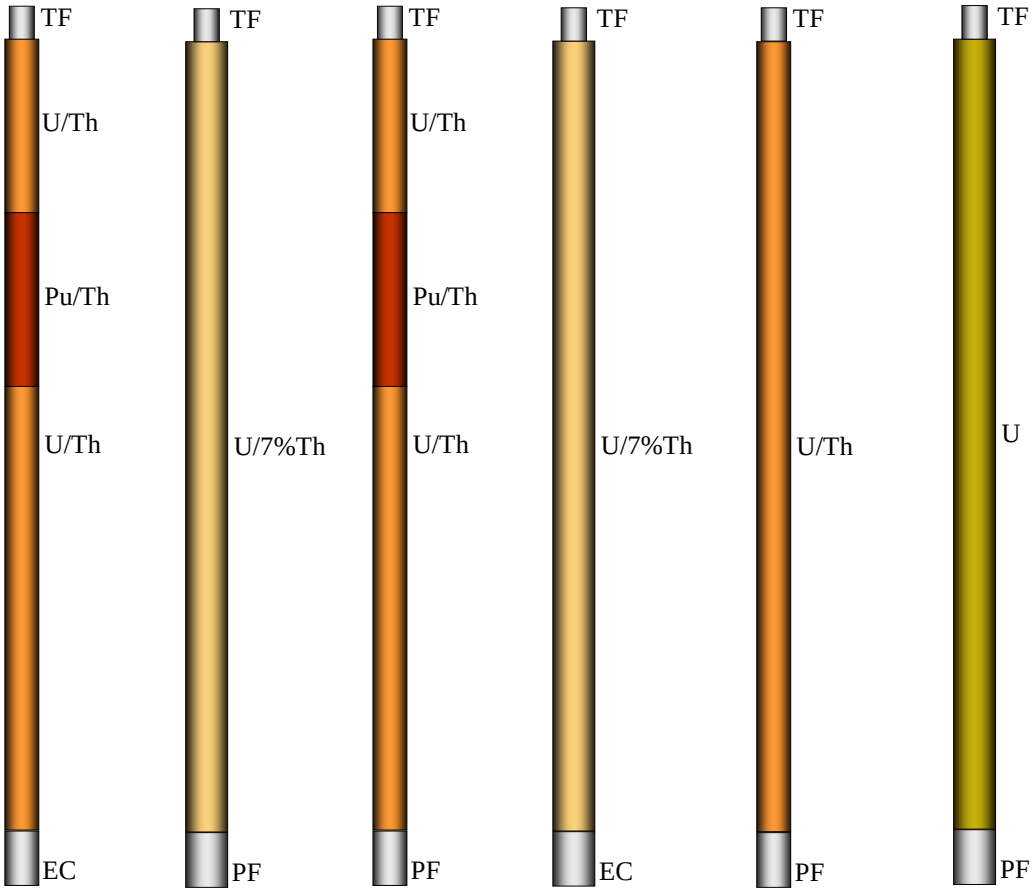
- K. Bakker, Journal of nuclear materials 250, 1-12, 1997.
- J.H. Yang, Nuclear Technology 147, 113-119, 2004.

Thermal conductivity of U/Th



- K. Bakker, Journal of nuclear materials 250, 1-12, 1997.
- J.H. Yang, Nuclear Technology 147, 113-119, 2004.

IFA-730
Test Matrix – Phase 1a

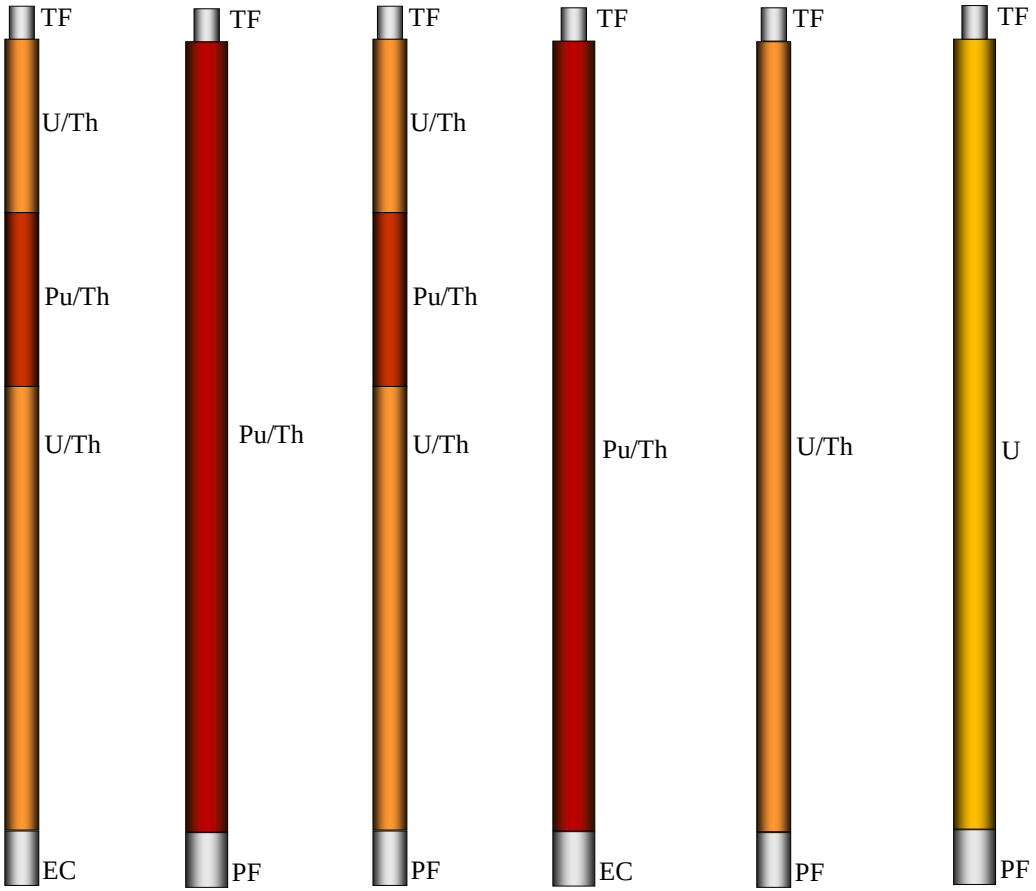


PF – Pressure transducer

TF – Fuel thermocouple

EC – Cladding elongation detector

IFA-730
Test Matrix – Phase 1b

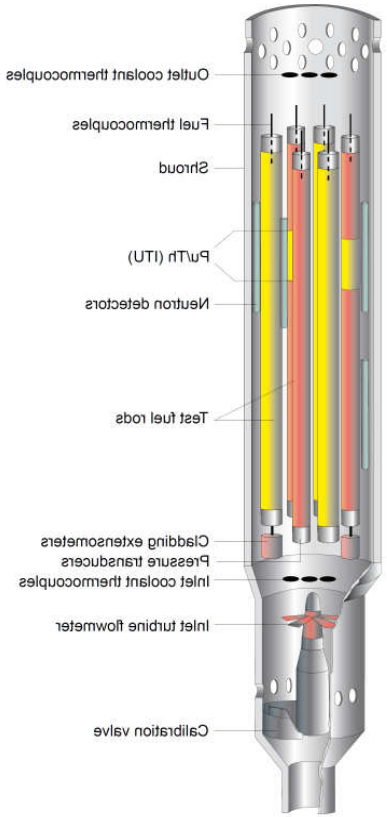


PF – Pressure transducer

TF – Fuel thermocouple

EC – Cladding elongation detector

IFA 730 program, an opportunity for Japan



- ❑ The test is performing well
- ❑ Unique data are being produced and
- ❑ Will continue to be produced on Th/Pu fuel
- ❑ An international Consortium led by TE is in place

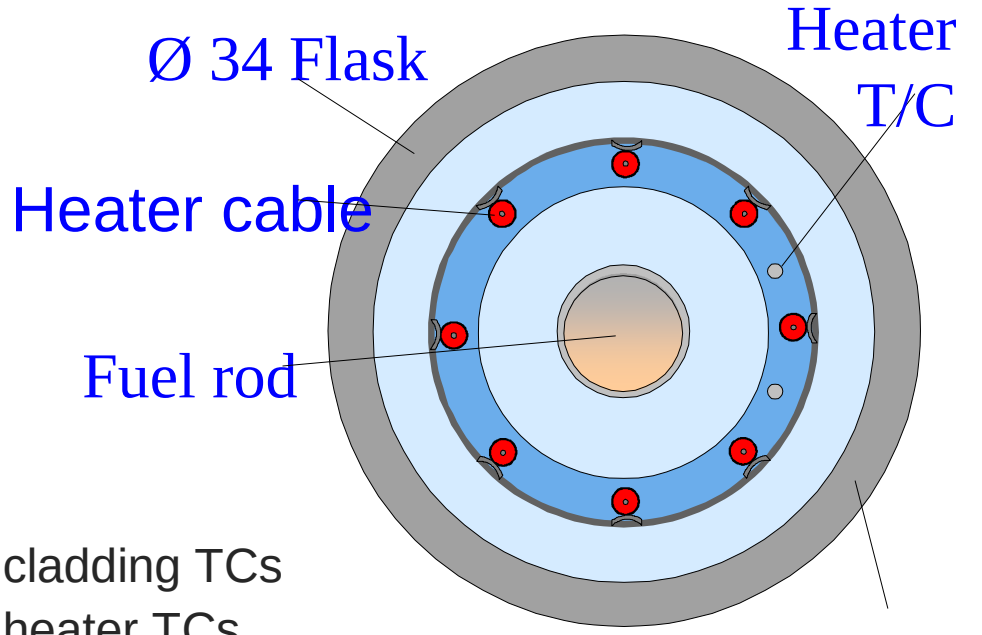
Joining this Consortium will be the most efficient and cost effective way for Japan-nuclear to get access to unique data and to be part of the Thorium future

Thank you

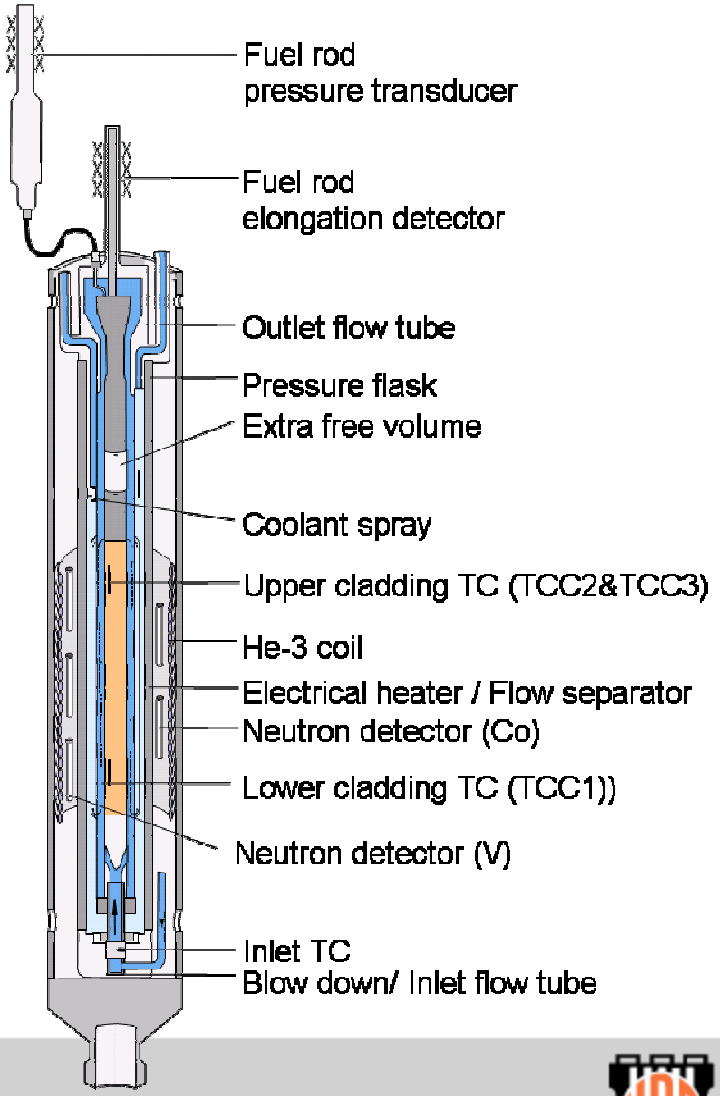
In-reactor simulation of LOCA

Features of the Halden LOCA rig

- Single rod experiment using high burnup fuel
- Heating provided from within the rod by low level of nuclear power simulating decay heat

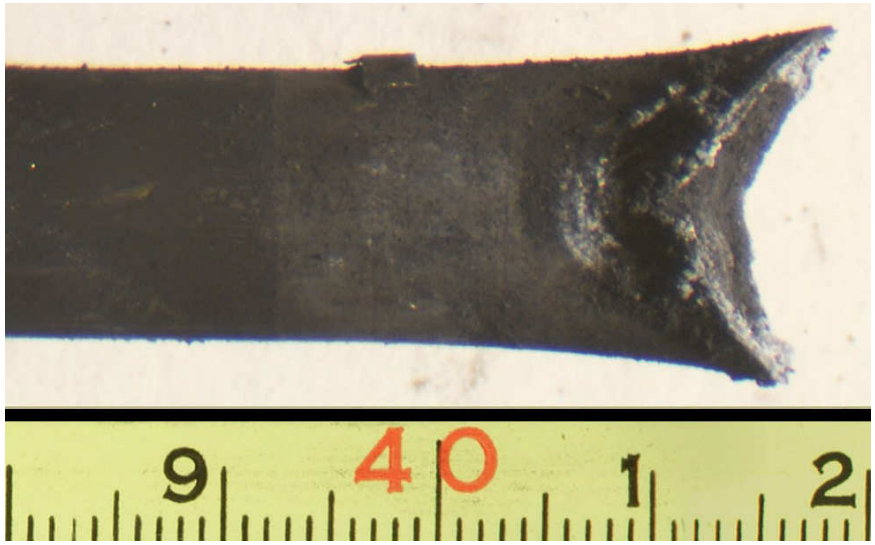


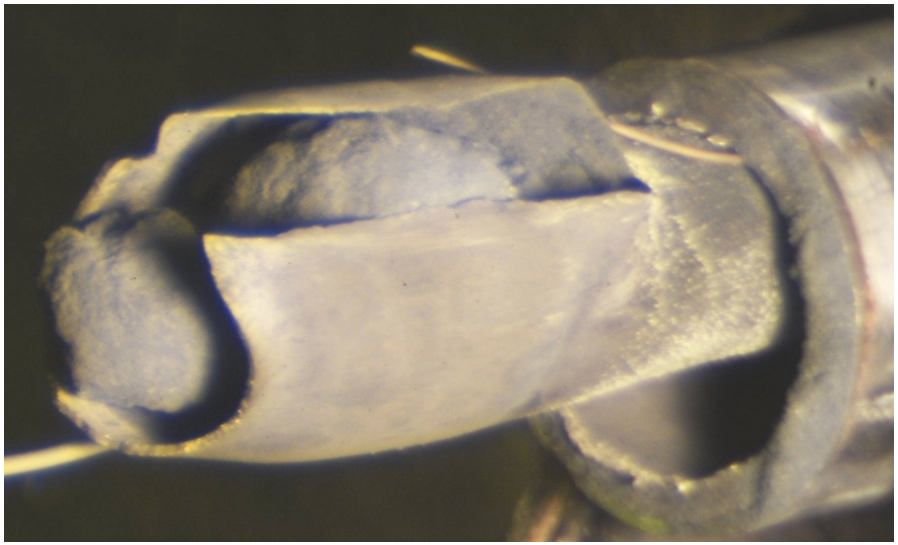
- 3 cladding TCs
- 3 heater TCs
- Fuel pressure transducer
- Cladding elongation detector



Appearance of rod cladding after LOCA (Halden test 9)

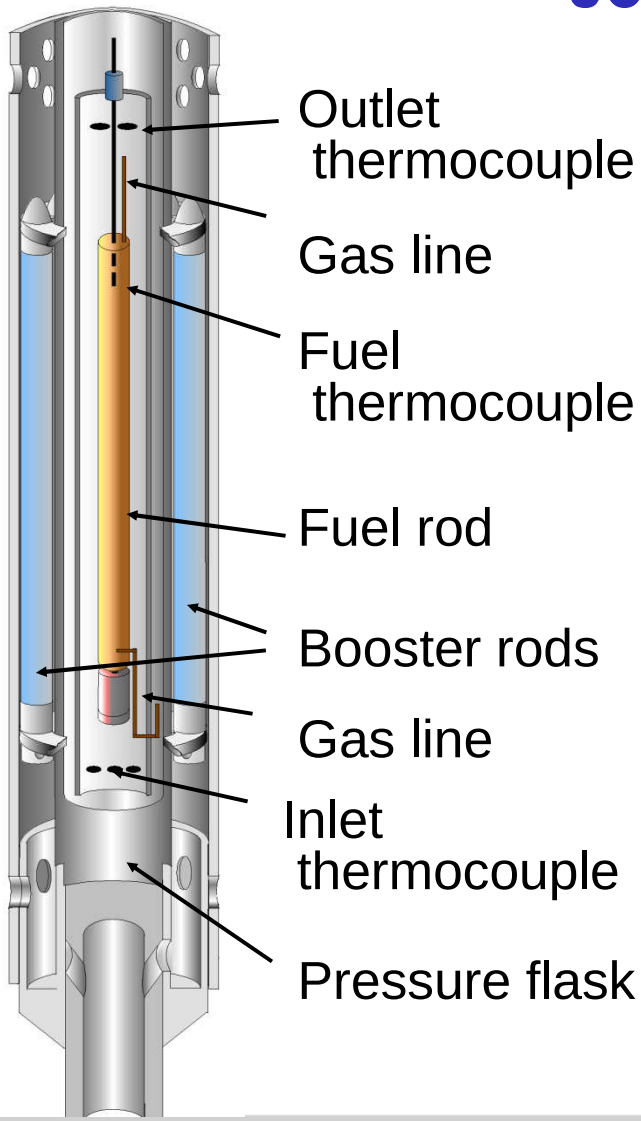
(rod probably broken during test)





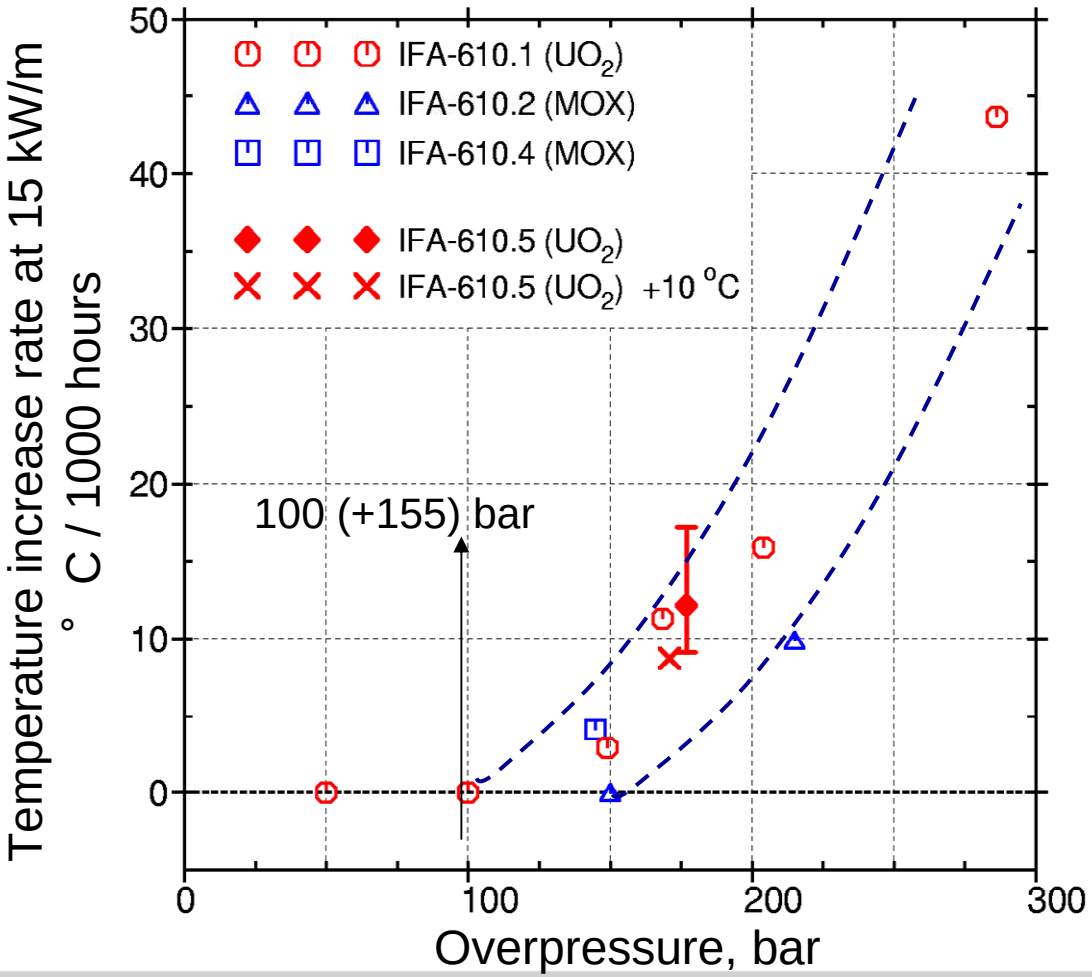
- Large balloon at bottom of rod
- Complete circumferential crack
- Hydrogen content near crack ~1600 ppm

Example 6: Tolerable rod overpressure in UO2 and MOX fuel rods



- PWR operation conditions
- Pressure flask surrounded by 12 booster rods
- Gas flow lines
  - Overpressure
  - Hydraulic diameter measurement and gamma-spectroscopy
- Thermocouple (TF) with in-core connector
- Cladding extensometry (EC)

Tolerable Rod Overpressure Tests



Observations

- Rate of temperature increase correlated with overpressure
- Thermal feedback occurs only at considerable overpressure (>100 bar)
- Below this threshold, clad creep-out is sufficiently compensated by fuel swelling, and no net thermal feedback becomes apparent

IFA-730 Normalised temperature 2/2