In-reactor testing capabilities at Halden relevant for ATF fuel

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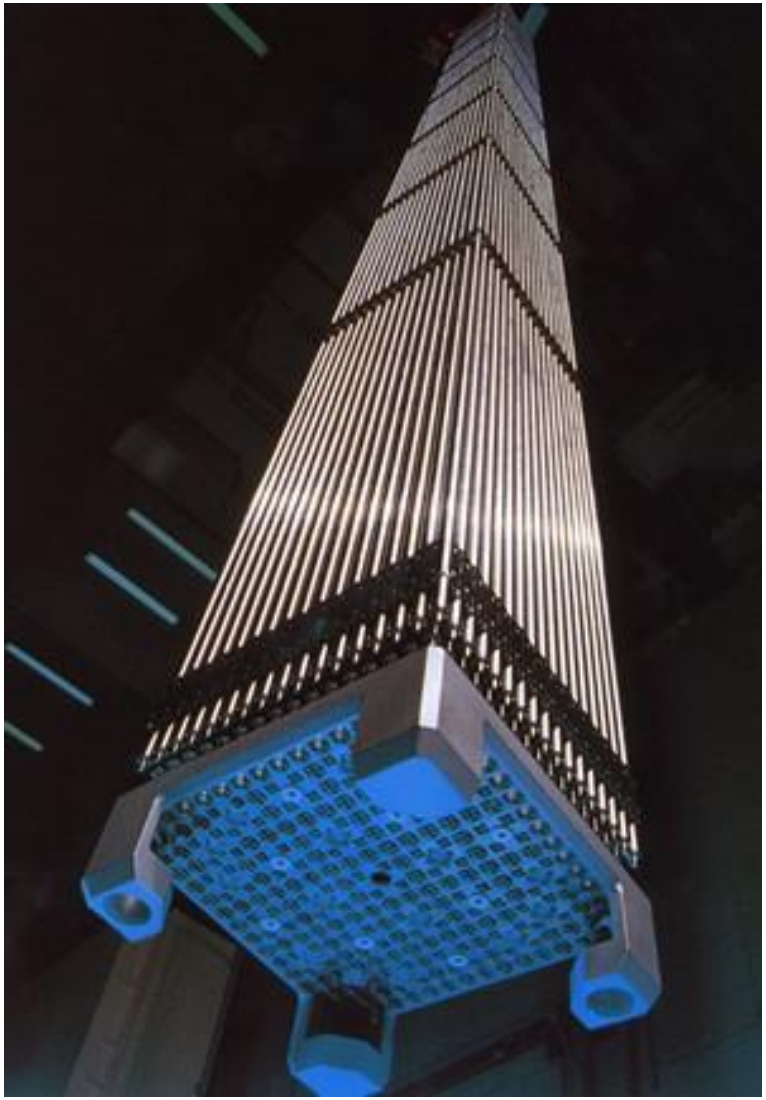

Nuclear fuel numbers

<u>Installed el-production capacity</u>

- 270 PWRs, 95 BWRs, 40 Candu
- 380 GWe el-production

<u>Western fuel production capacity</u>

- ~15,000 t U_M/year
- 6 million fuel rods /y
- 28,000 fuel assemblies /y
- 28,000 Km of fuel length /y

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Target is zero failure

Previous experience

- The development of new accident tolerant fuel for LWRs is a long-term endeavor
- It will require a very extensive selection and validation process consisting of
  - Laboratory assessments
  - Out-of reactor testing
  - In-reactor testing
- Likely, the focus will be on new cladding materials, because the cladding is the weakest part in DB and BDB accident conditions
- This presentation outlines the types of in-reactor testing that are likely to be required in the ATF development, based on previous experience at Halden and elsewhere

In reactor testing for qualifying new cladding materials

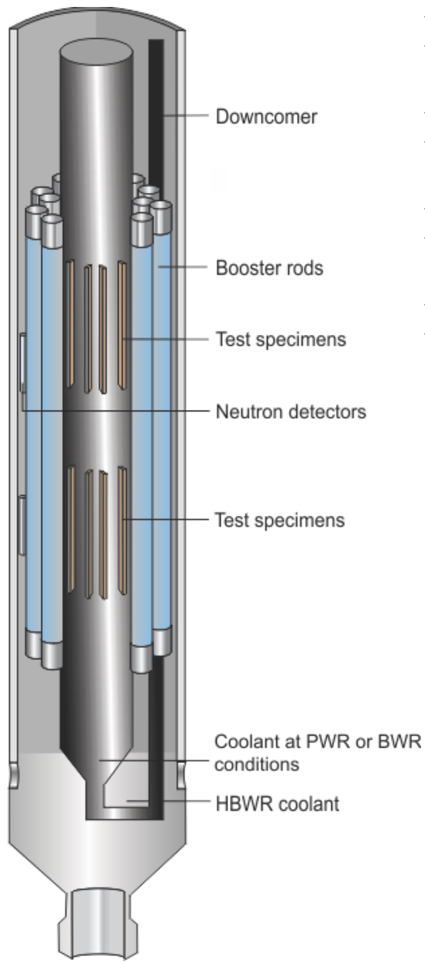
1. In-reactor material property assessments
    a. Stress corrosion cracking susceptibility
    b. Corrosion and irradiation growth
    c. Irradiation creep and stress relaxation

2. Rod behavior in normal operation
    a. Integrity of un-fuelled rods (for e.g. SiC qualification)
    b. Integrity of fuel rods, normal operation
    c. PCI/PCMI margin in power ramps

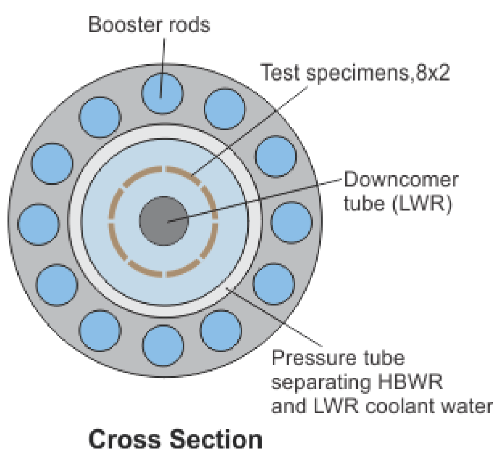
3. Rod behavior in transients
    a. RIA transients
    b. DB LOCA
    c. BDB LOCA (Severe Accident scenarios)

Corrosion and axial growth testing

Screening of different material variants



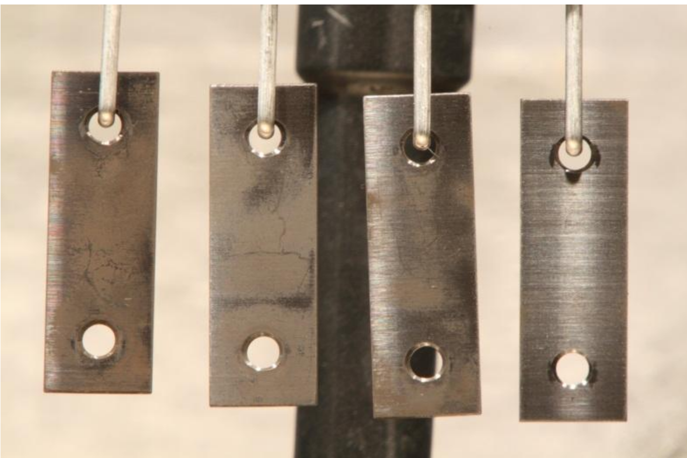
- Comparative, simultaneous testing of coupons with various material variants
- Representative LWR conditions, both for water chemistry and neutron flux
- Corrosion is assessed by weight change measurements carried out during outages
- Cladding growth is given by coupon length changes, also done during outages

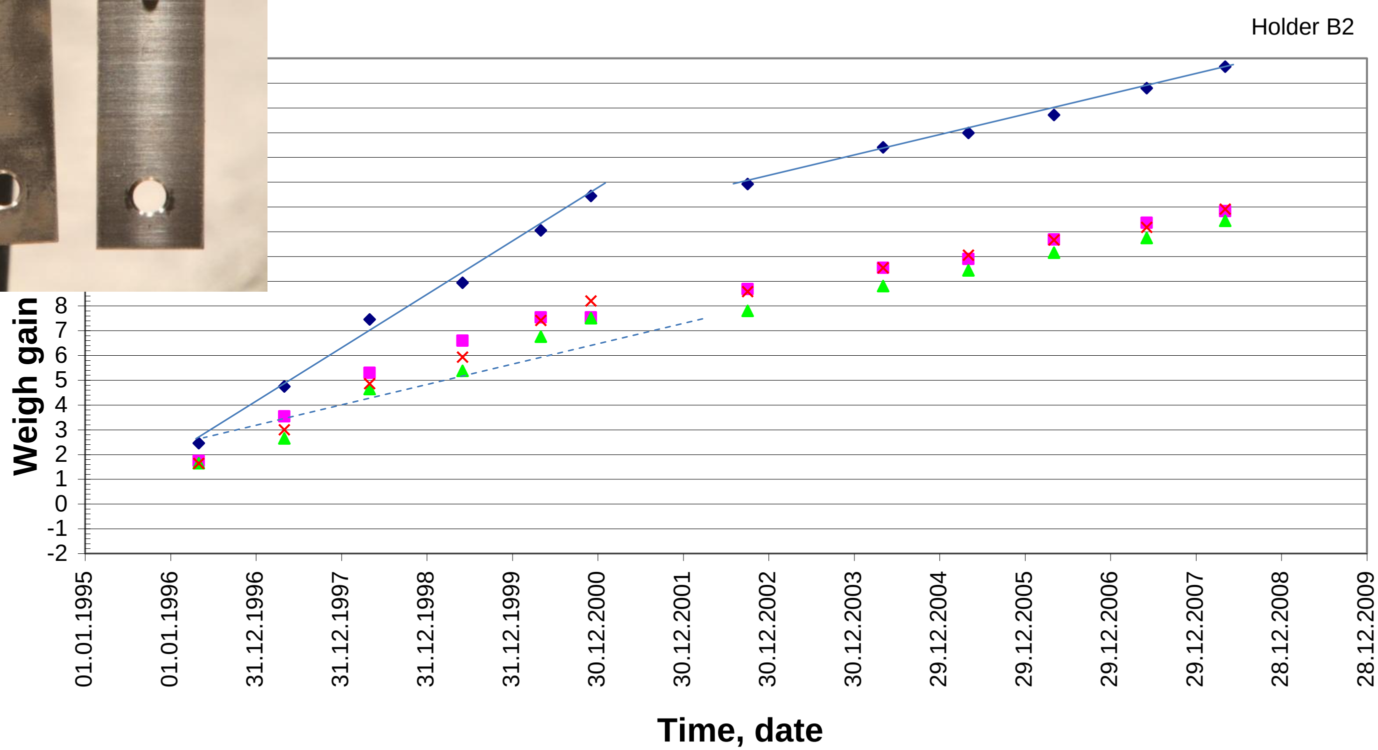


Material property assessment

Example of corrosion assessment for 4 different Zr alloys

Weight gain data vs. time

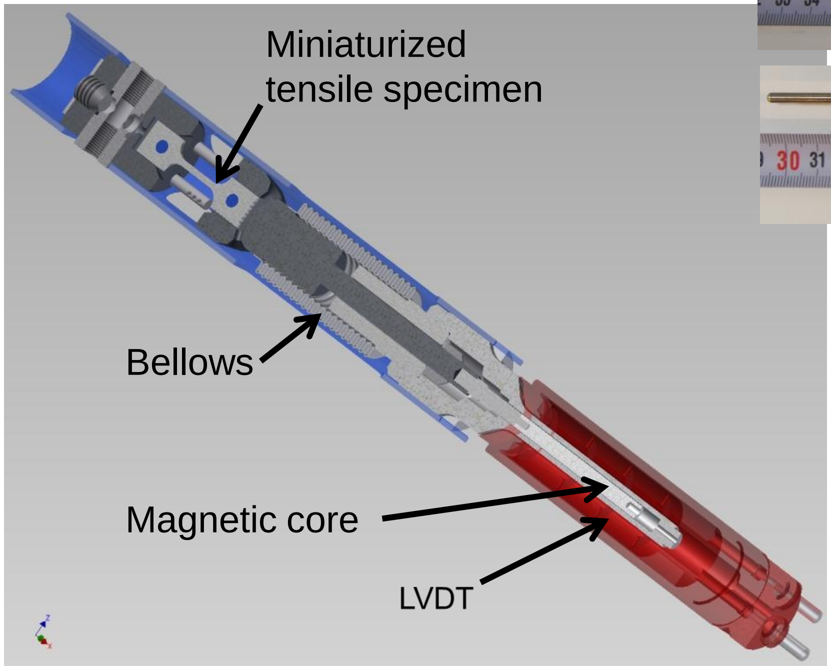




Test unit for mechanical property assessment

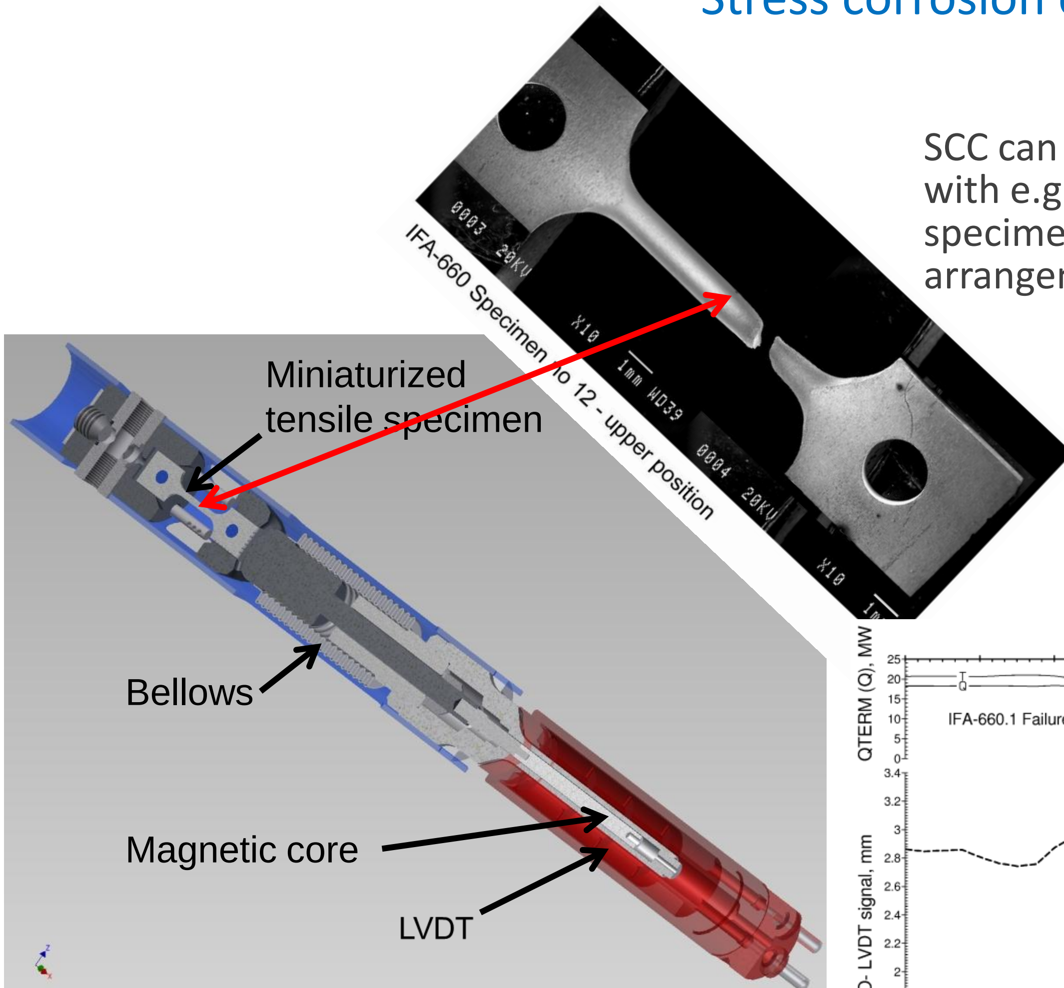
Can be used for:

- Stress corrosion cracking assessment
- Irradiation creep database
- Stress relaxation database

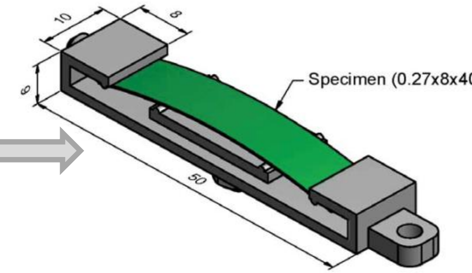


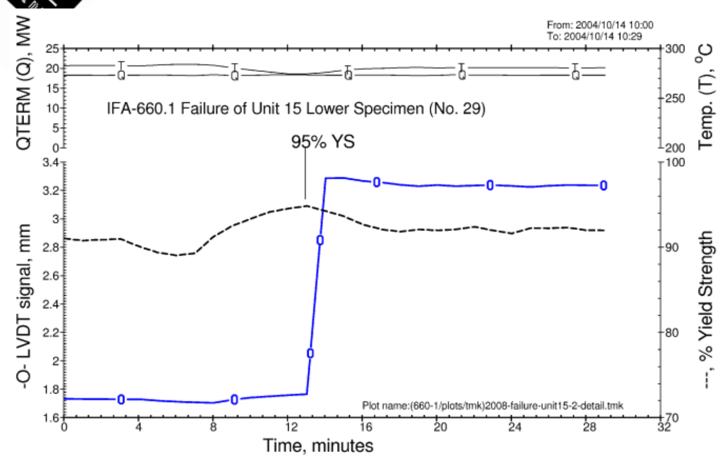
- Many of these test units can be loaded in the same rig, hence enabling simultaneous testing of various materials / conditions.
- Example of mechanical property assessments performed with this equipment are given in the following

Stress corrosion cracking



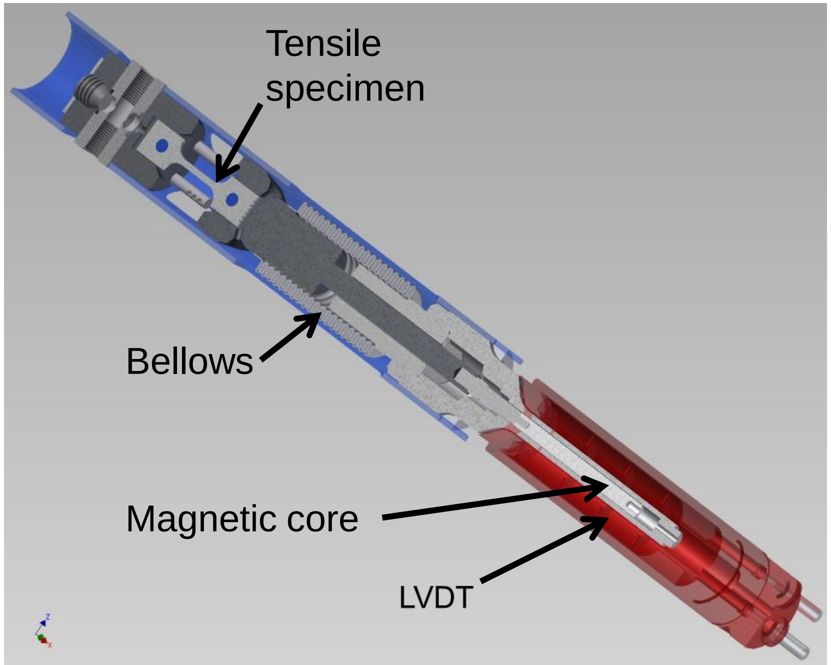
SCC can also be tested with e.g. 4-point bent specimen arrangements





Creep and stress relaxation under irradiation

Irradiation test unit



- Many miniaturized test specimens (of the type shown in the figure) are assembled in one test rig for repeatability and for comparative testing

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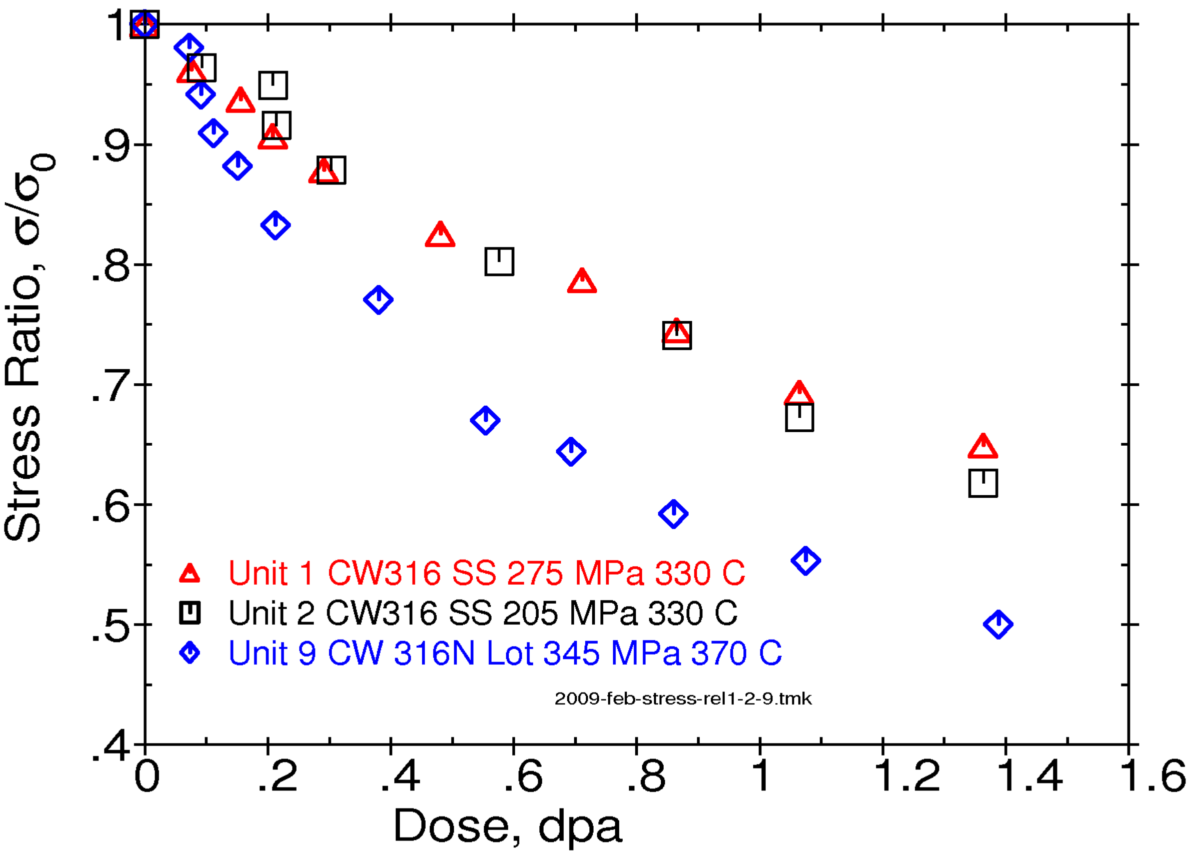
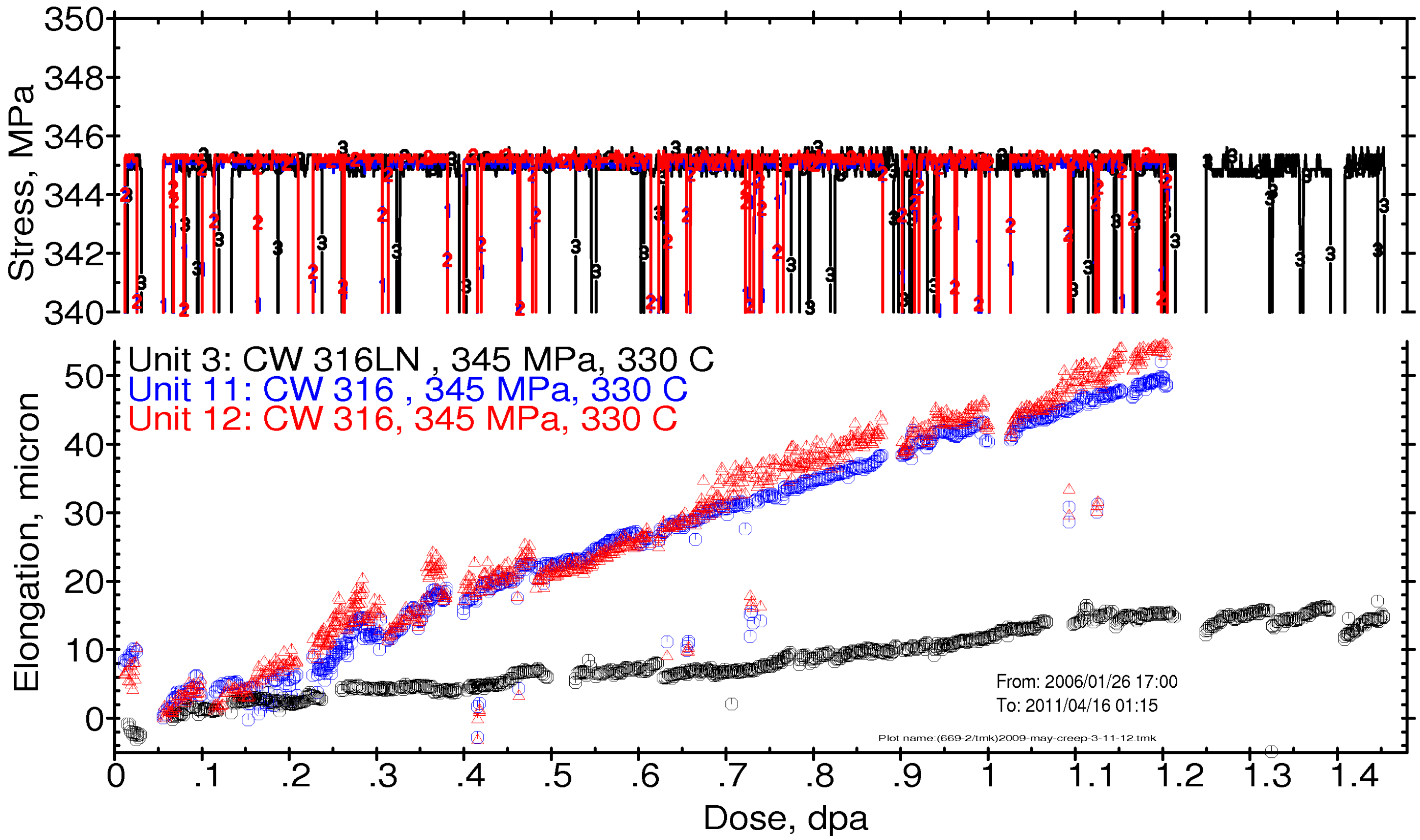
- Constant stress for determination of irradiation creep
- Constant total strain for determination of stress relaxation

Example of stress relaxation assessment for 3 different SS alloys

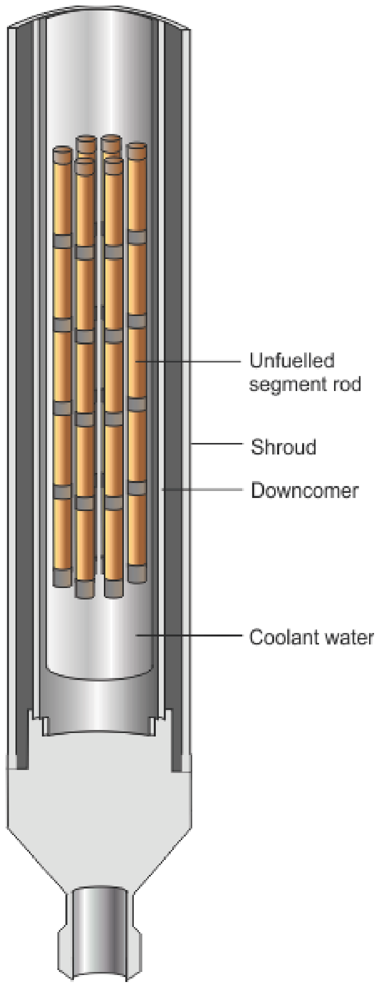
Stress ratio data vs. fast neutron dose



Example of irradiation creep assessment for 3 different SS alloys
Strain (elongation) data vs. dose

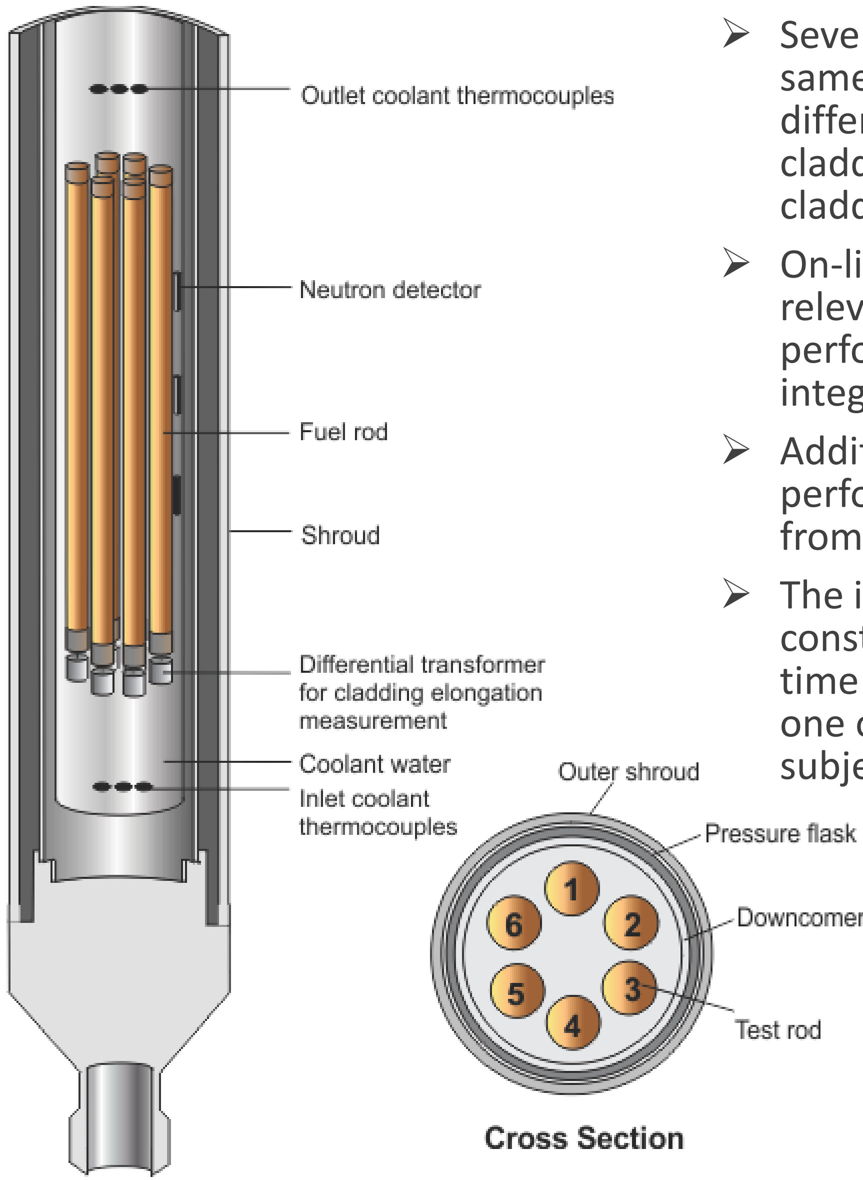


Integrity of un-fuelled rods



- Irradiation of un-fuelled test rods can be a first step for the qualifi-cation of quite new cladding types
- Many rod segments can be tested at the same time enabling to:
  - Compare different claddings
  - Assess end-plug/ weld integrity
- The irradiation is carried out at prototypical LWR temperature, pressure and (where needed) water chemistry
- Rod loss of integrity is determined during outages (or possibly on-line)
- Other properties such as creep-down or corrosion can be determined in inspections or PIE

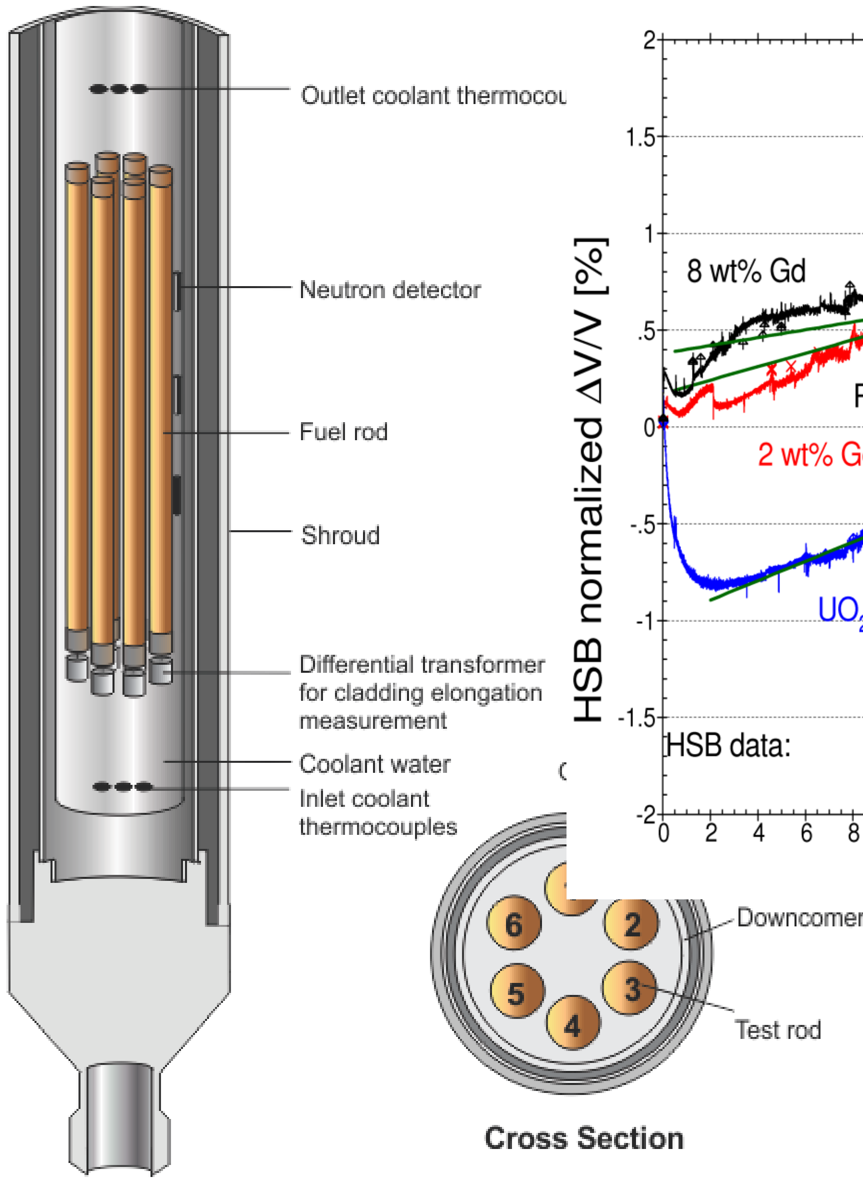
Integrity of fuel rods at base irradiation conditions

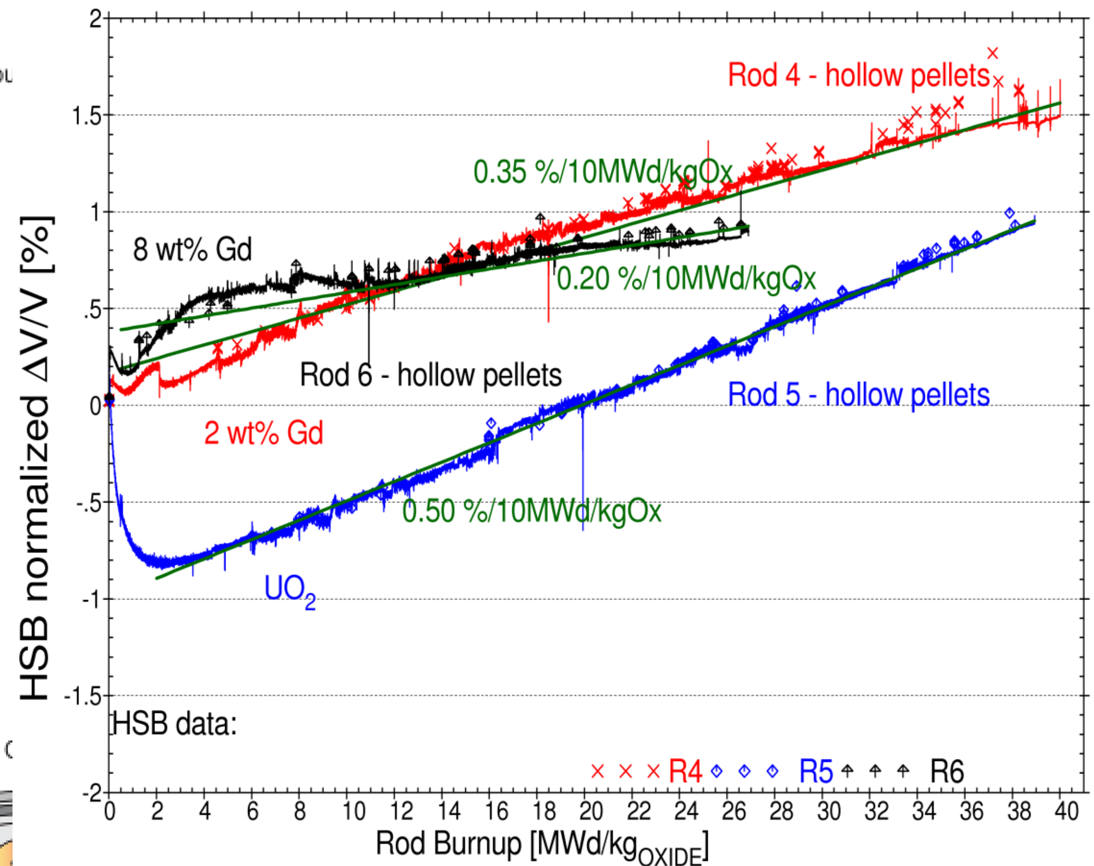


Cross Section

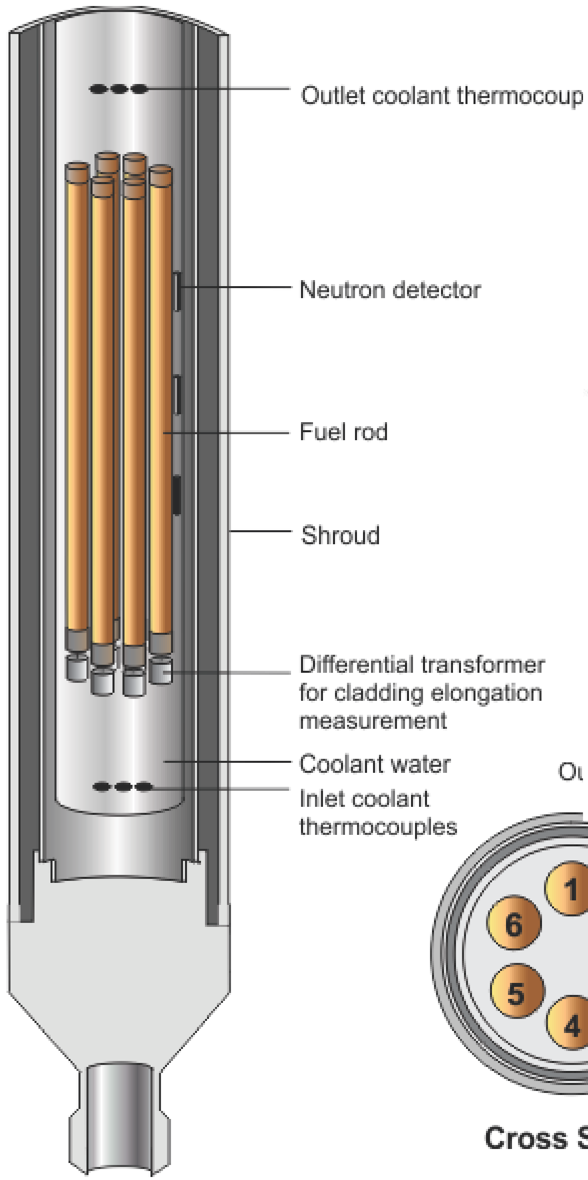
- Several fuel rods can be tested at the same time, hence enabling to compare different fuel rod variants, such as cladding type, fuel pellet type, pellet-cladding gap etc.
- On-line instrumentation provides relevant information on the fuel rod performance, e.g. on PCMI and rod integrity (and also FGR 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.

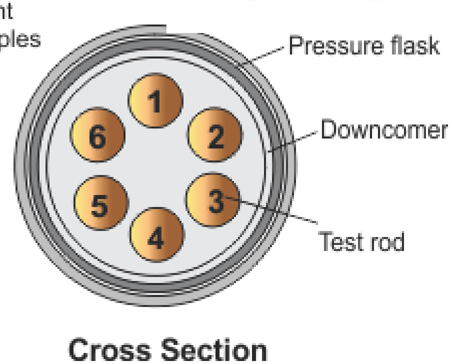
Integrity of fuel rods at base irradiation: Fuel swelling

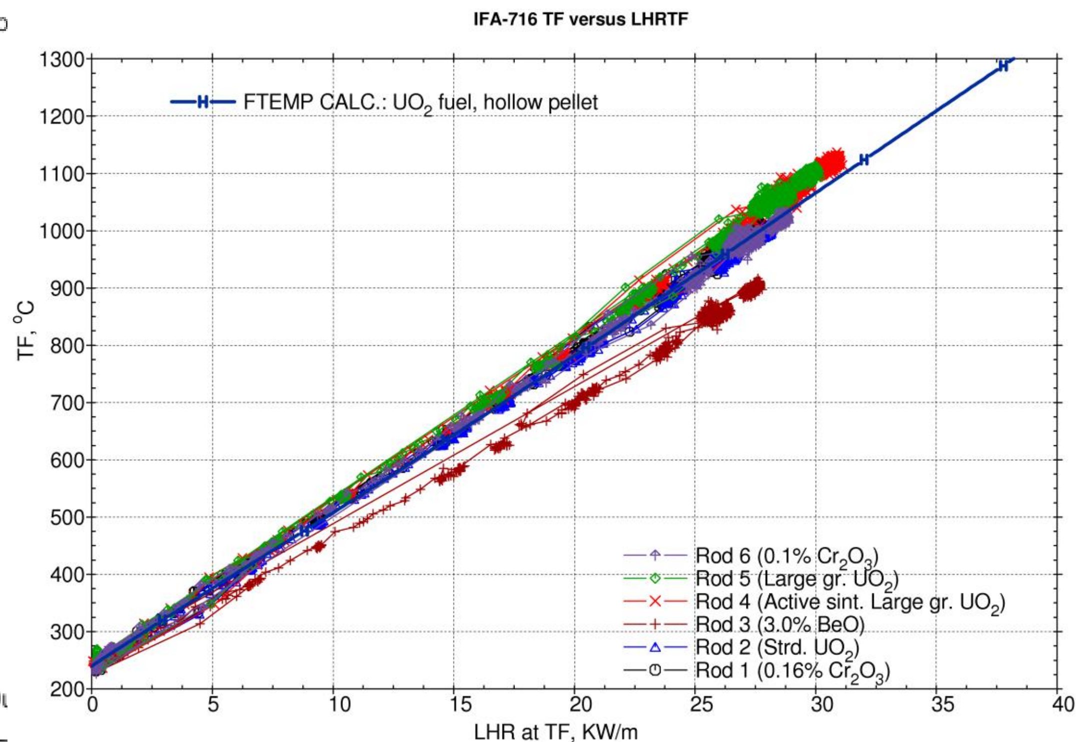




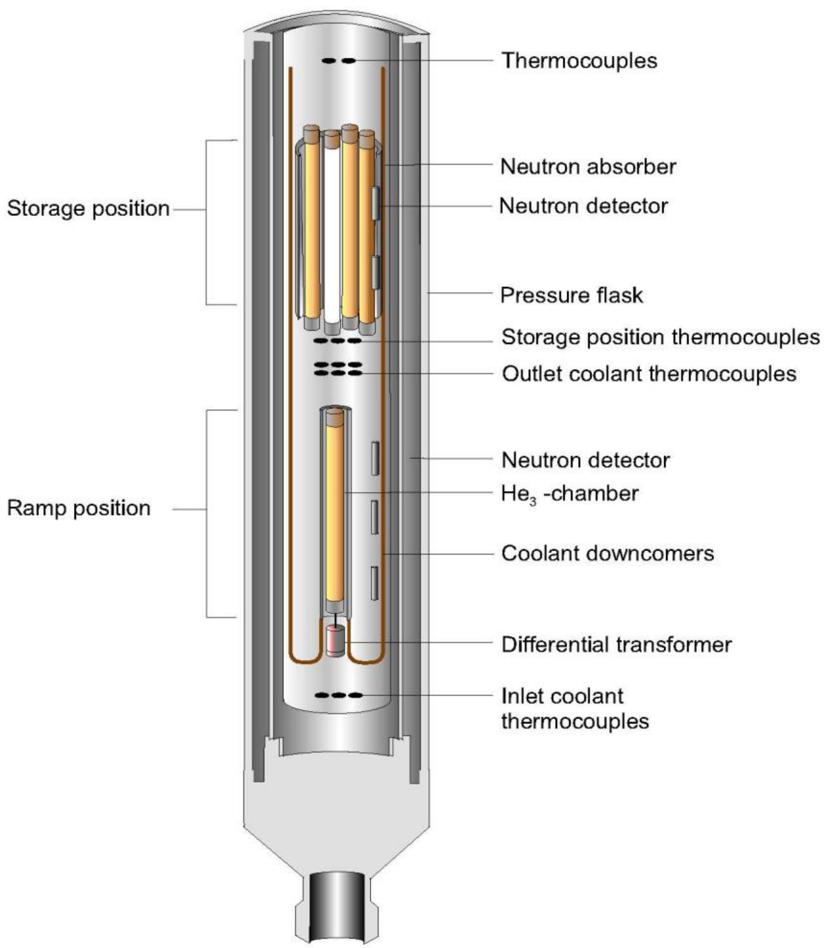
Integrity of fuel rods at base irradiation: Fuel swelling







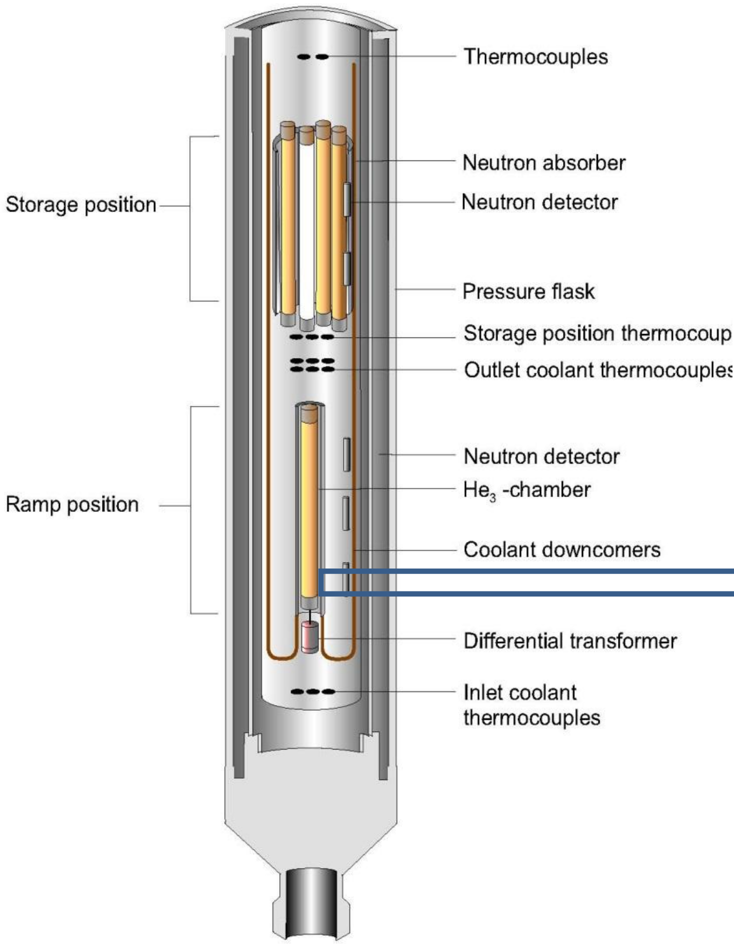
Integrity of fuel rods in power ramp conditions
Assessment of PCI and PCMI margin

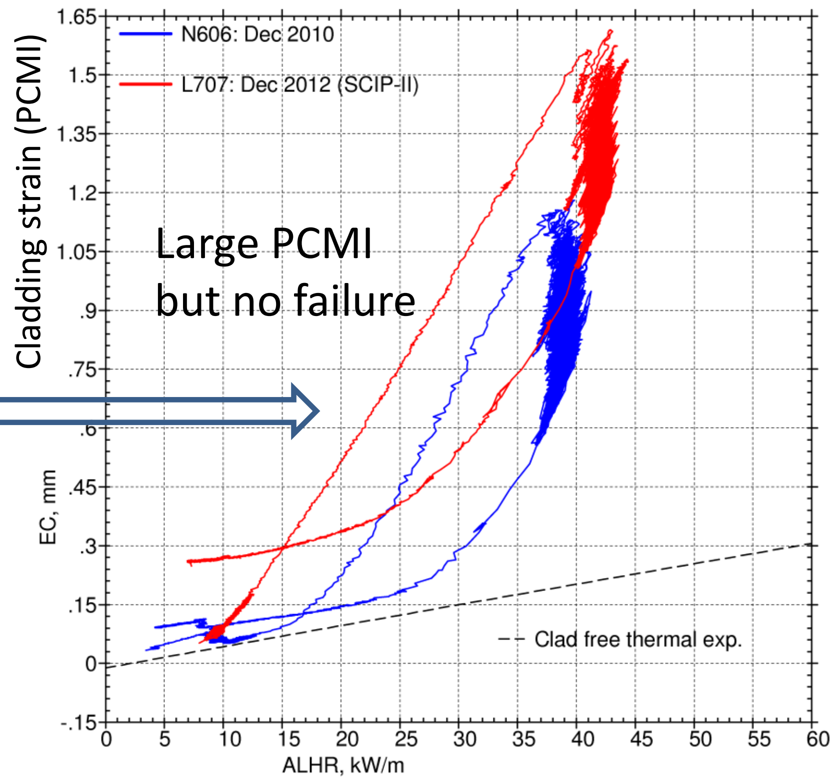


- A power ramp consists of a rapid and large power increment, e.g. 30kW/m in 1 minute
- Power ramps are currently used as standard test for assessing PCI/PCMI fuel margins.
- The Halden rig has four fuel rod storage positions in low-flux zone at the upper part of ramp rig (shielded).
  One fuel rod in ramp position.
- The rods can be moved (one at a time) from storage position to ramp position
- Helium-3 chamber surrounding fuel rod ramp position provides:
  - local power control (power level and ramp rate)
  - insulation for reducing power calibration uncertainties
- The ramp rig contains instrumentation for failure detection (cladding elongation detector) and for monitoring environmental parameters (neutron / gamma flux and coolant temperatures)
  - Monitoring of activity release to loop water
    - on-line gamma-monitor + water sampling

Integrity of fuel rods in power ramp conditions

Assessment of PCI and PCMI margin

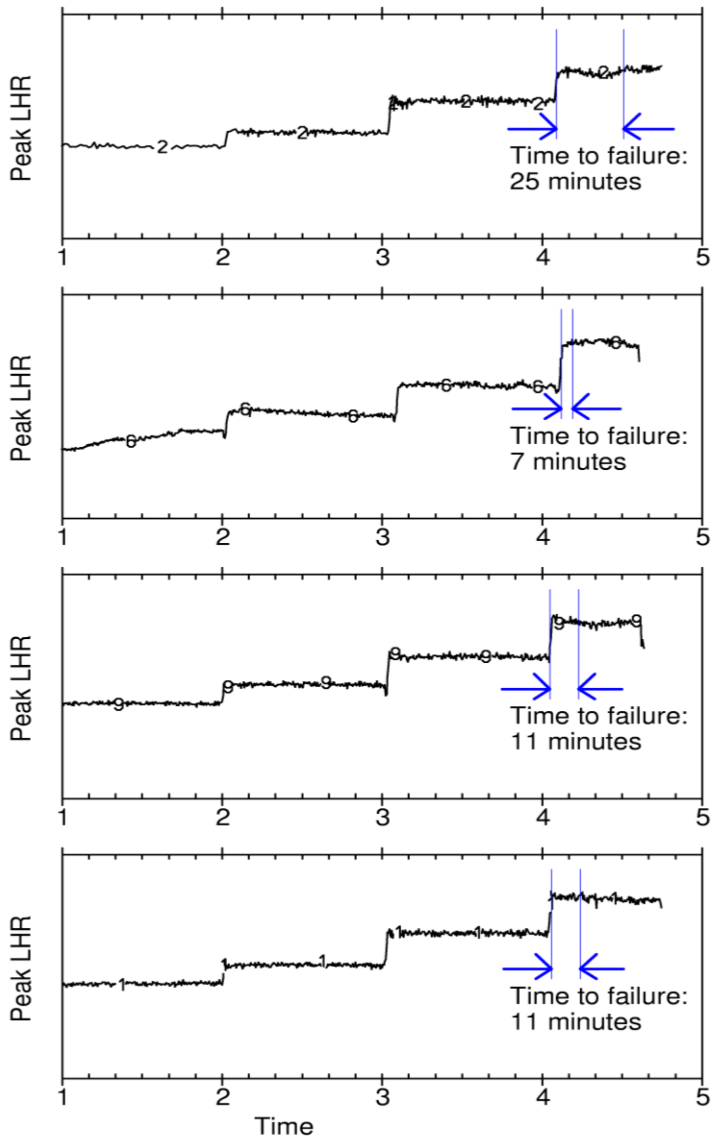




Example of failed rod

Integrity of fuel rods in power ramp conditions

Assessment of PCI and PCMI margin



Past experience: Repeatability

Test-1: 1 STD rod, 2 additive fuel rods

- The STD rod failed after 25 minutes
- The two additive fuel rods survived

Test-2: 1 STD rod, 2 additive fuel rods

- The STD rod failed after 7 minutes
- The two additive fuel rods survived

Test-3: 1 STD rod, 2 additive fuel rods

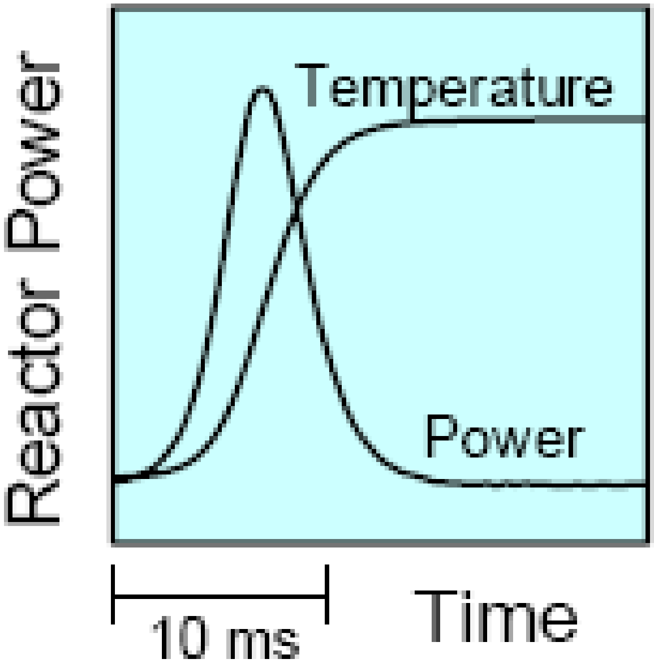
- The STD rod failed after 11 minutes
- The two additive fuel rods survived

Test-4: 1 STD rod, 2 additive fuel rods

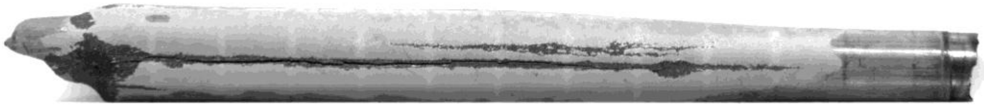
- The STD rod failed after 11 minutes
- The two additive fuel rods survived

RIA transients

- For new fuels, RIA tests may be required at some point in time.
- These tests are run in specialized reactors, such as NSRR in Japan



NSRR pulse reactor for RIA tests



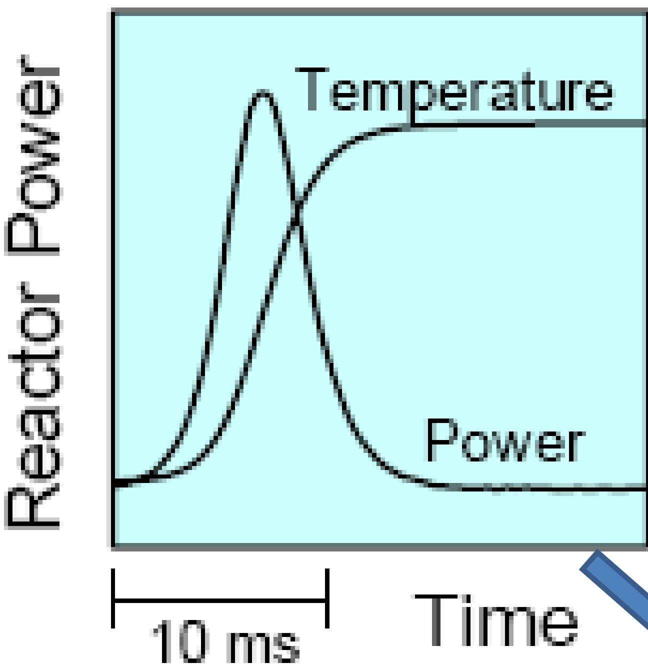
Example of failed rod

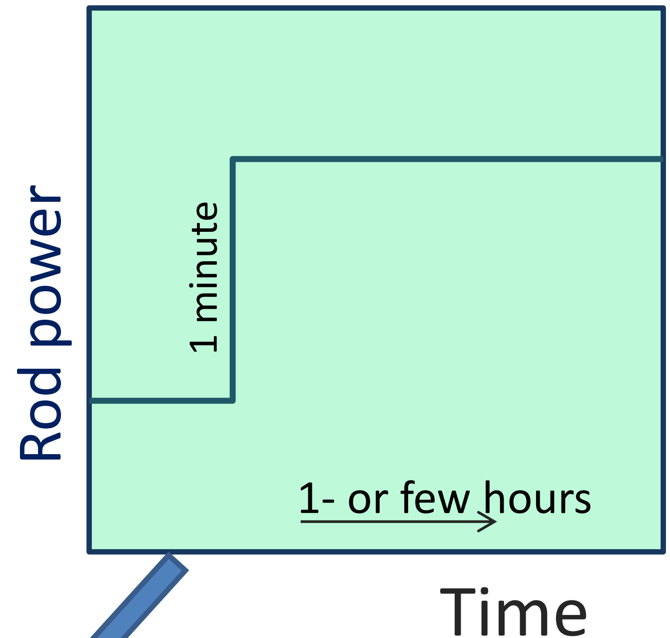
Rod behavior in transient operation

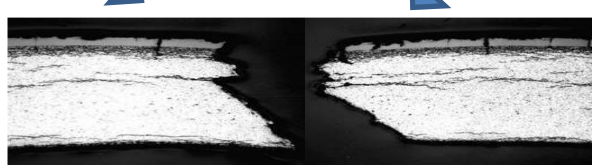
RIA transients

For new fuels, RIA tests may be required at some point in time. These tests are run in specialized reactors (not at Halden). Power ramps may cast much light on fuel tolerance of RIA

RIA transients ← analogy with → Power ramp





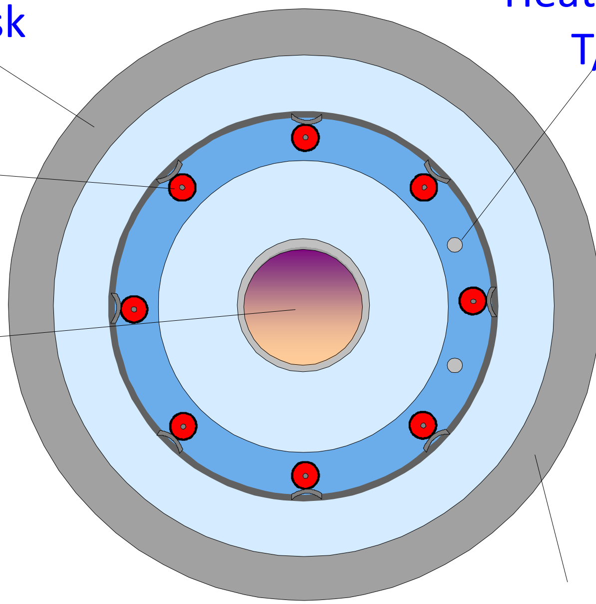


Both may result in PCMI-induced fuel cladding failure

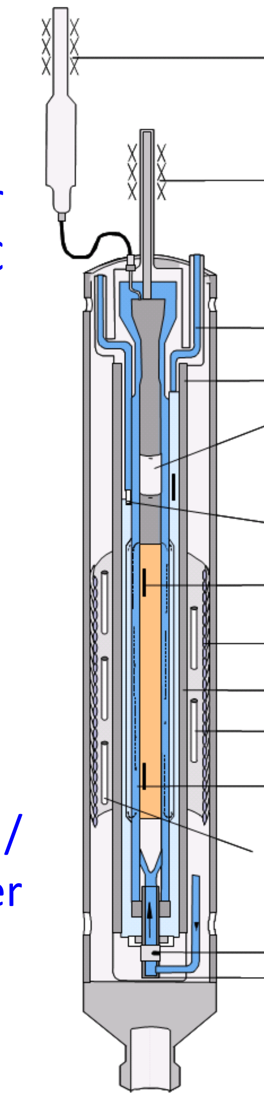
In-reactor simulation of LOCA

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

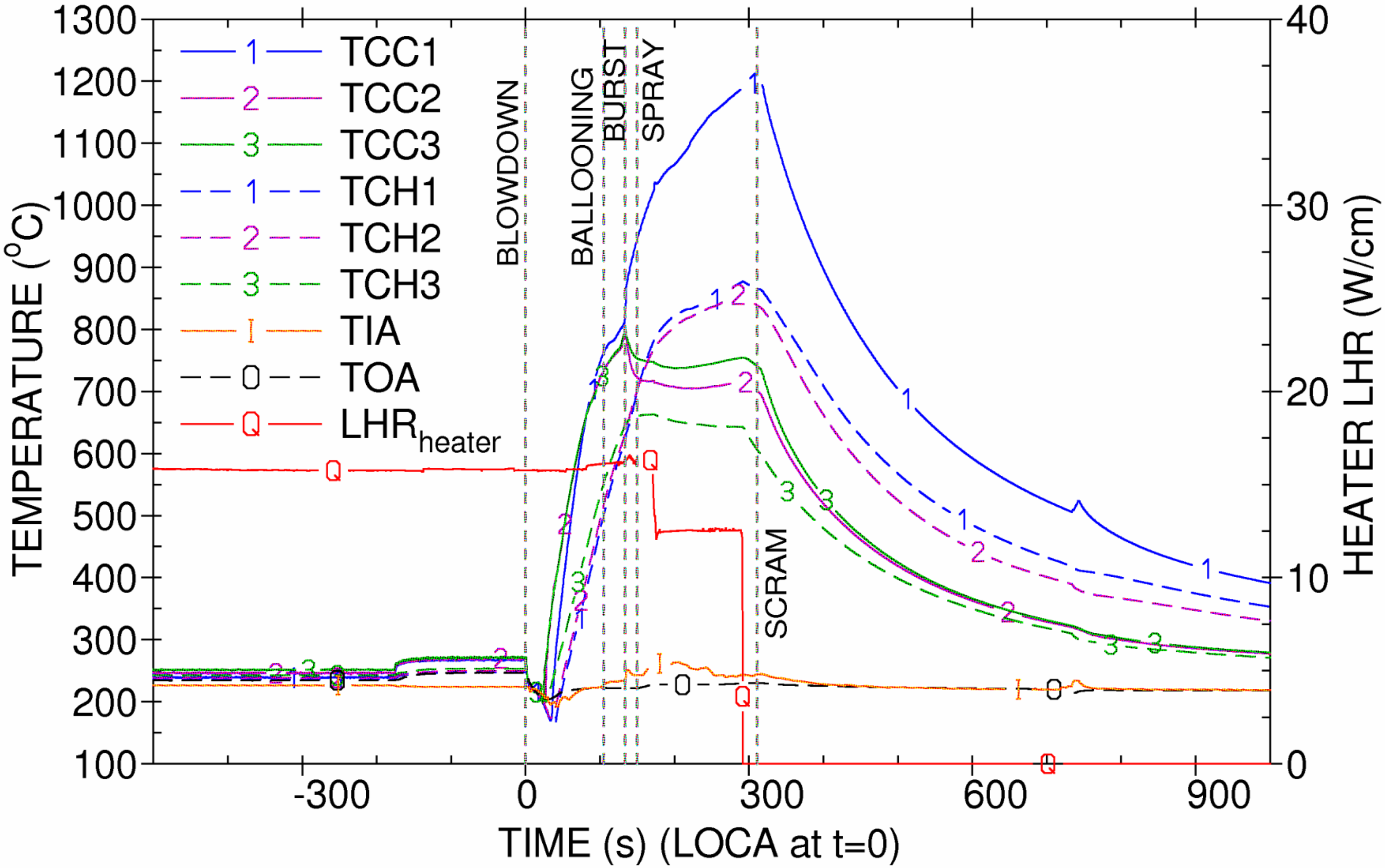


- 3 cladding TCs
- 3 heater TCs
- Fuel pressure transducer
- Cladding elongation detector



Rod behavior in transient operation

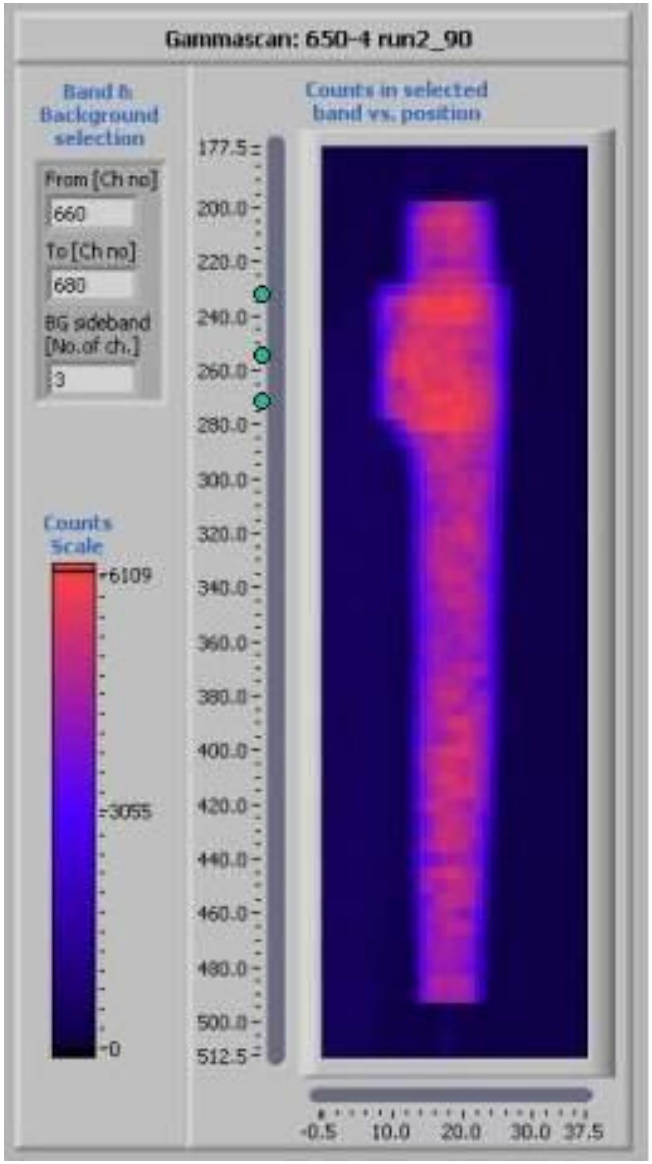
LOCA tests at Halden
Example of temperature measurements (Test 9)

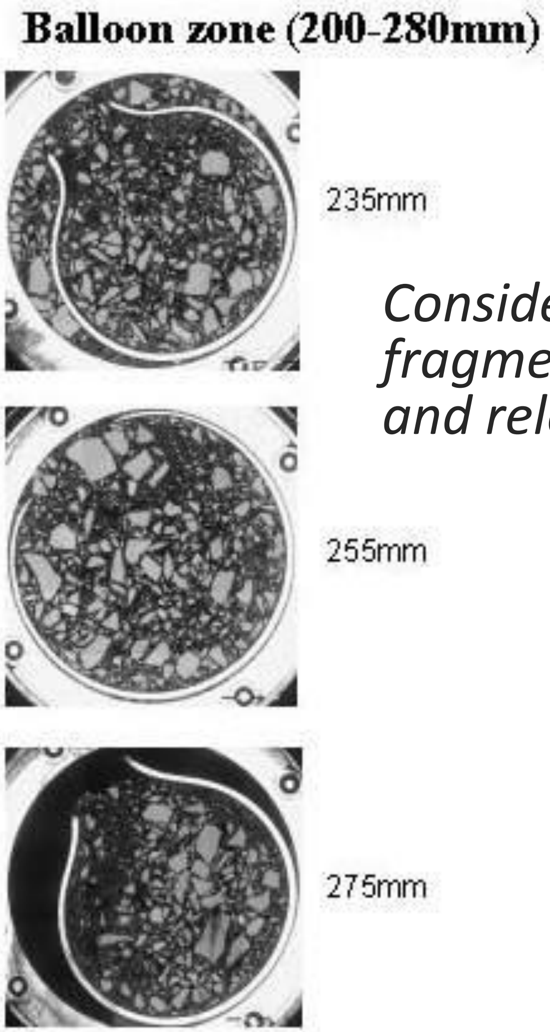


Hilary/Kolstad 56

Halden LOCA Test 4

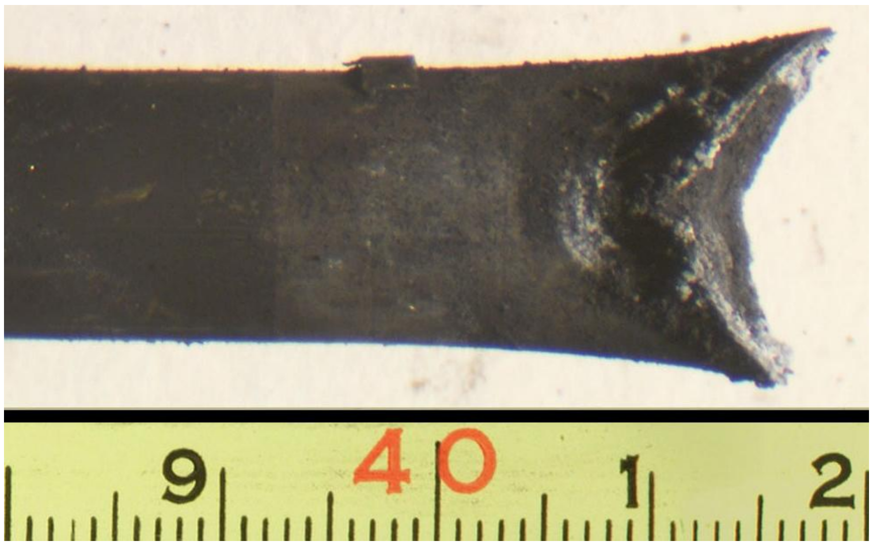
Large ballooning, extensive fuel fragmentation and ejection

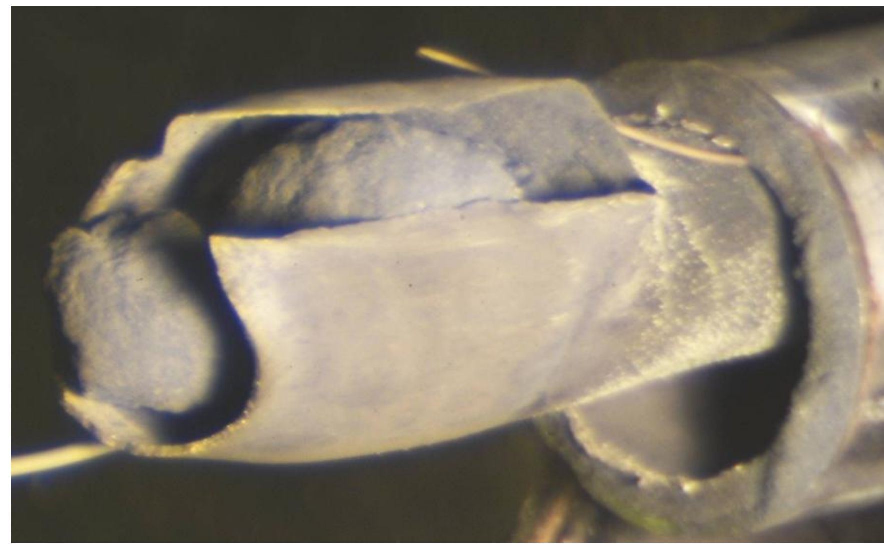




Considerable fuel fragmentation and relocation

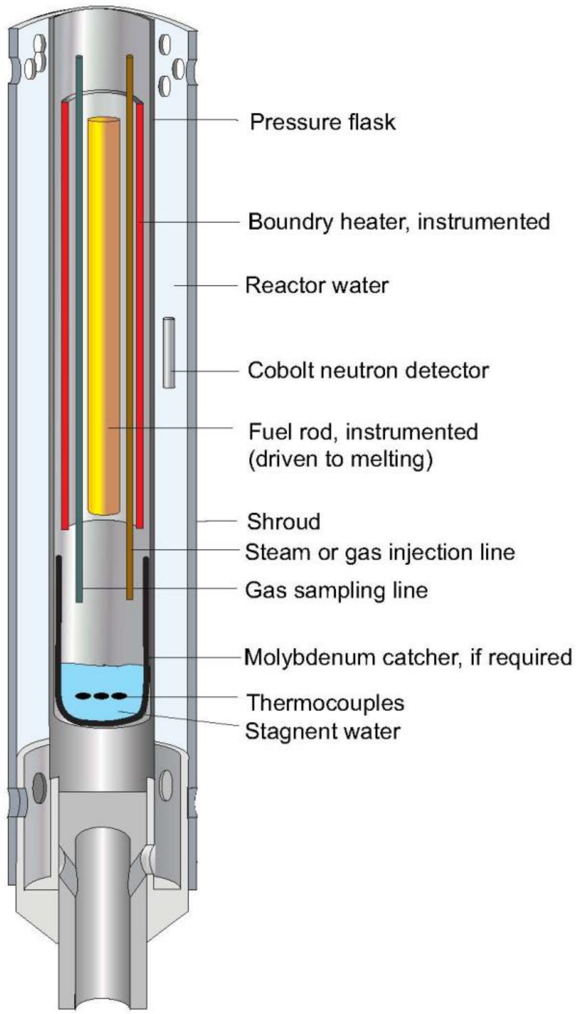
Appearance of rod cladding after LOCA (Halden test 9)
(rod probably broken during test)





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

Beyond Design Base LOCA: Possible Test Goals



Beyond-LOCA studies

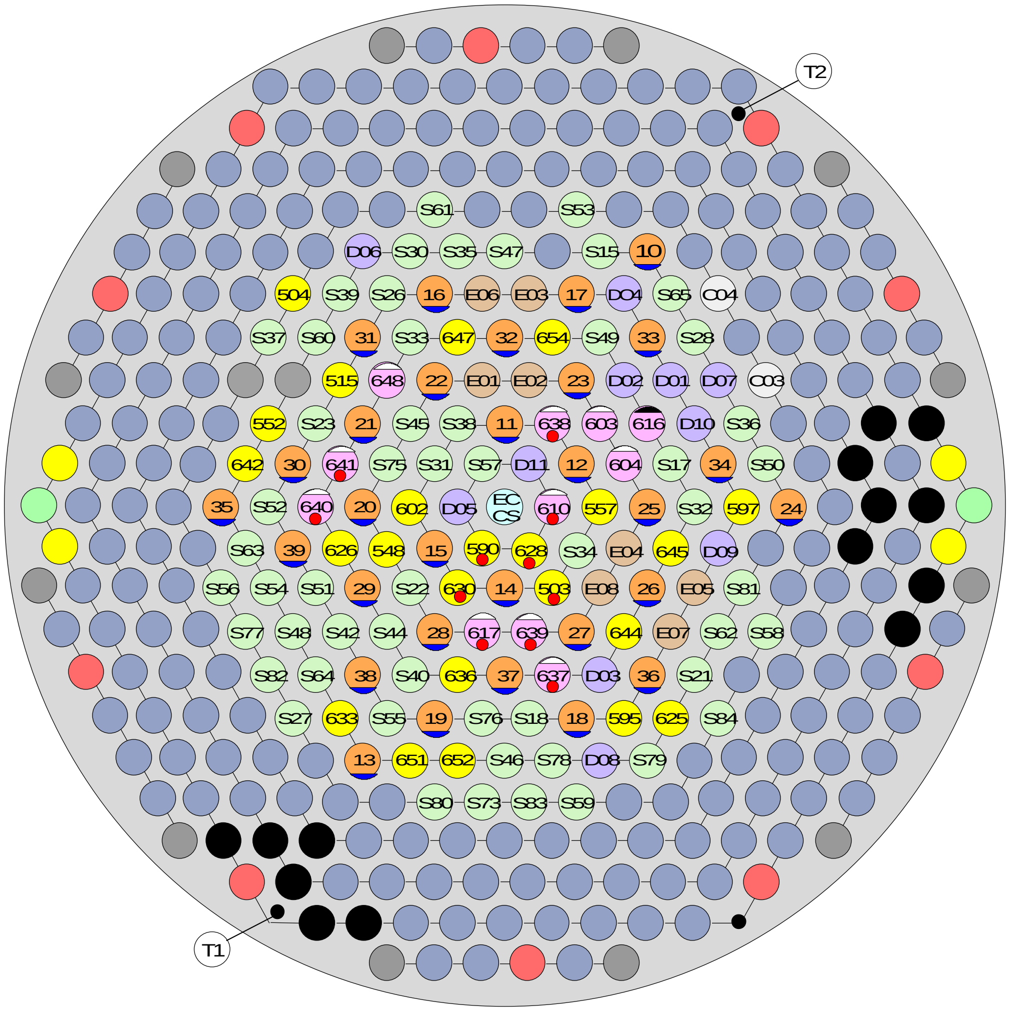
- Accident progression
  - Single rod
  - Temperature well beyond 1200 ^{o}C
  - Different environment (air, steam, mixed)
  - Valuable instrumentation to feasible extent

  - What happens to the fuel?
  - Until what stage it maintains its geometry?
  - How does the debris look like? Is it coolable?
  - Do we have H-release or other types of high temperature issues?
- Source term (including short-lived)

Summary

- The development of new accident tolerant fuel for LWRs is a long-term endeavor
- It will require a very extensive selection and validation process consisting of
  - Laboratory assessments
  - Out-of reactor testing
  - In-reactor testing
- Likely, the focus will be on new cladding materials, because the cladding is the "weakest" part in accident conditions
- The presentation summarizes the types of in-reactor testing that are likely to be required in ATF developments, based on previous experience at Halden and elsewhere, and addressing:
  - Material property assessment
  - Integrity in normal operation conditions
  - Behavior in accident conditions

HBWR Core Configuration



HBWR Core Configuration

- More than 300 positions individually accessible
- About 110 positions in central core
- About 30-35 positions for experimental purposes (any of 110)
- Height of active core 80cm
- Usable length within moderator about 140cm
- Experimental channel Ø:
  - 70mm in HBWR moderator
  - 35-45mm in pressure flask
- 15 loop systems for simulation of BWR/PWR conditions

Thank you

Overview of Halden LOCA experience (last 5 years)

	type	burnup MWd/kg	clad	treat-ment	oxide µm	PCT °C	pres. bar	when
1	PWR	0	commissioning test			1100	1	< 2007
2	PWR	0	commissioning test			850	40	-
3	PWR	82	Zry-4	SRA	24-27	850	40	-
4	PWR	92	Zry-4	SRA	10-11	850	40	-
5	PWR	83	Zry-4	SRA	65-80	1100	40	< 2007
6	VVER	55	E110		~5	850	30	4/2007
7	BWR	44	LK3/L		<10	1150	6	4/2008
8	PWR	0	commissioning test			1100	1	12/2008
9	PWR	90	Zry-4	SRA	7-8	1100	40	4/2009
10	PWR	61	Zry-4	SRA	20-30	850	40	5/2010
11	VVER	55	E110			1000	30	10/2010
12	BWR	72	LK3/L			<800	low	2011
13	BWR	72	LK3/L			<800	low	2012

NSRR RIA test data

Zr-based cladding alloys