CANDU Fuel Behavior in Limited and Severe Core Damage Accidents

Lawrence Dickson, Fuel Modeling Section Leader
Fuel and Fuel Channel Safety (F&FCS) Branch

Presented to US Nuclear Regulatory Commission
Washington DC
May 6-7, 2003

AECL TECHNOLOGIES INC.

Outline

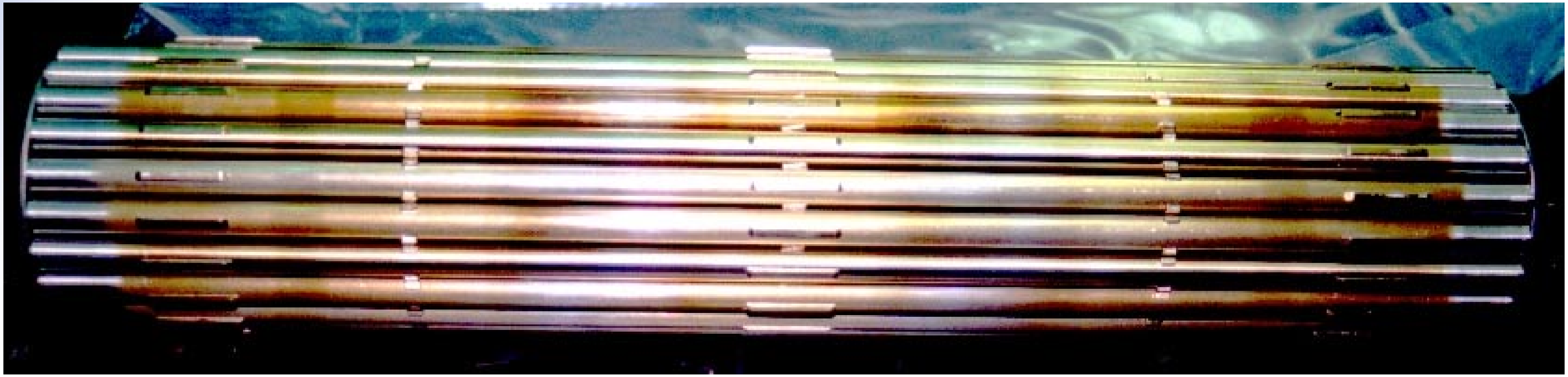
- ACR Fuel Design
- Fuel Behavior in Limited and Severe Core Damage Accidents
- Experimental Database
- Computer Codes

ACR Fuel Design

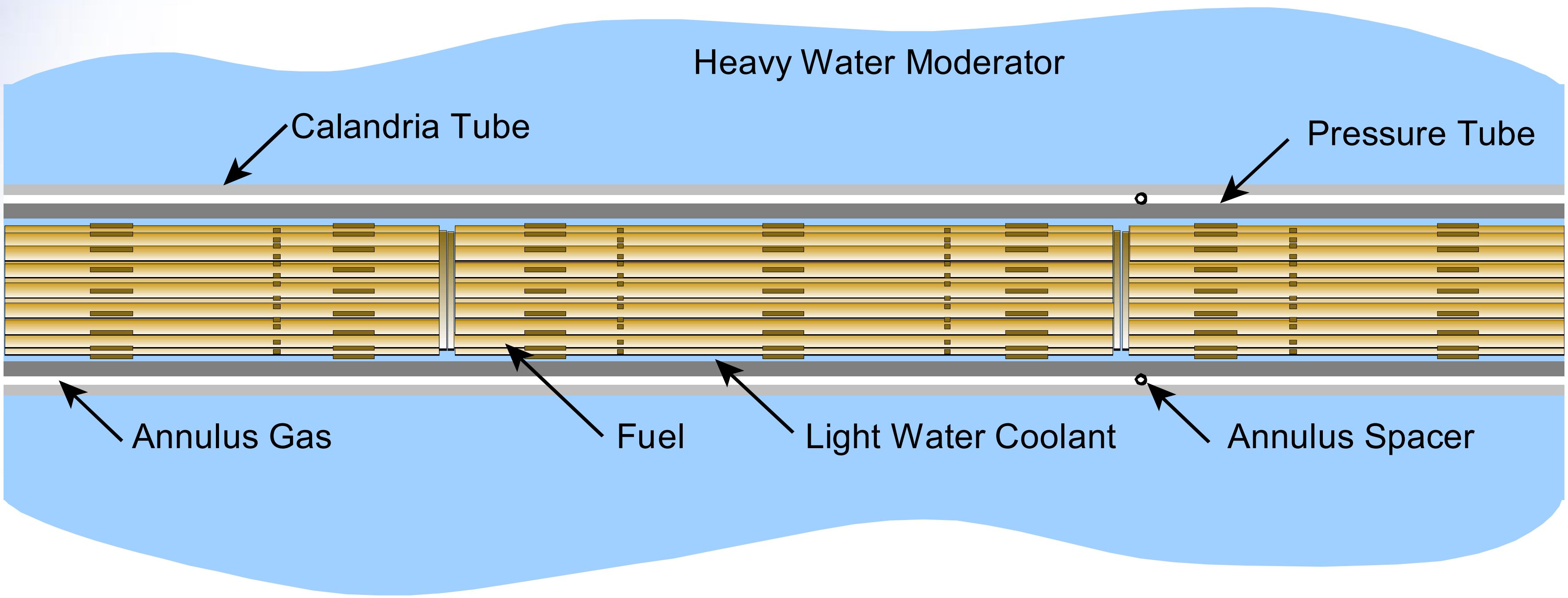
- UO_2 fuel pellets clad with Zircaloy-4
- Pellet diameters:
  - Center / inner elements: 12.58 mm
  - Intermediate / outer elements: 10.65 mm
- Enrichment:
  - Center element: 0.71 wt% ^{235}U (Dy-doped)
  - Inner / intermediate / outer element: 2.0 wt% ^{235}U
- Fuel cladding thickness: ~0.4 mm
- Bundle length: ~0.5 m
- Fuel elements in bundle: 43

CANFLEX Fuel Design

CANFLEX Fuel Design



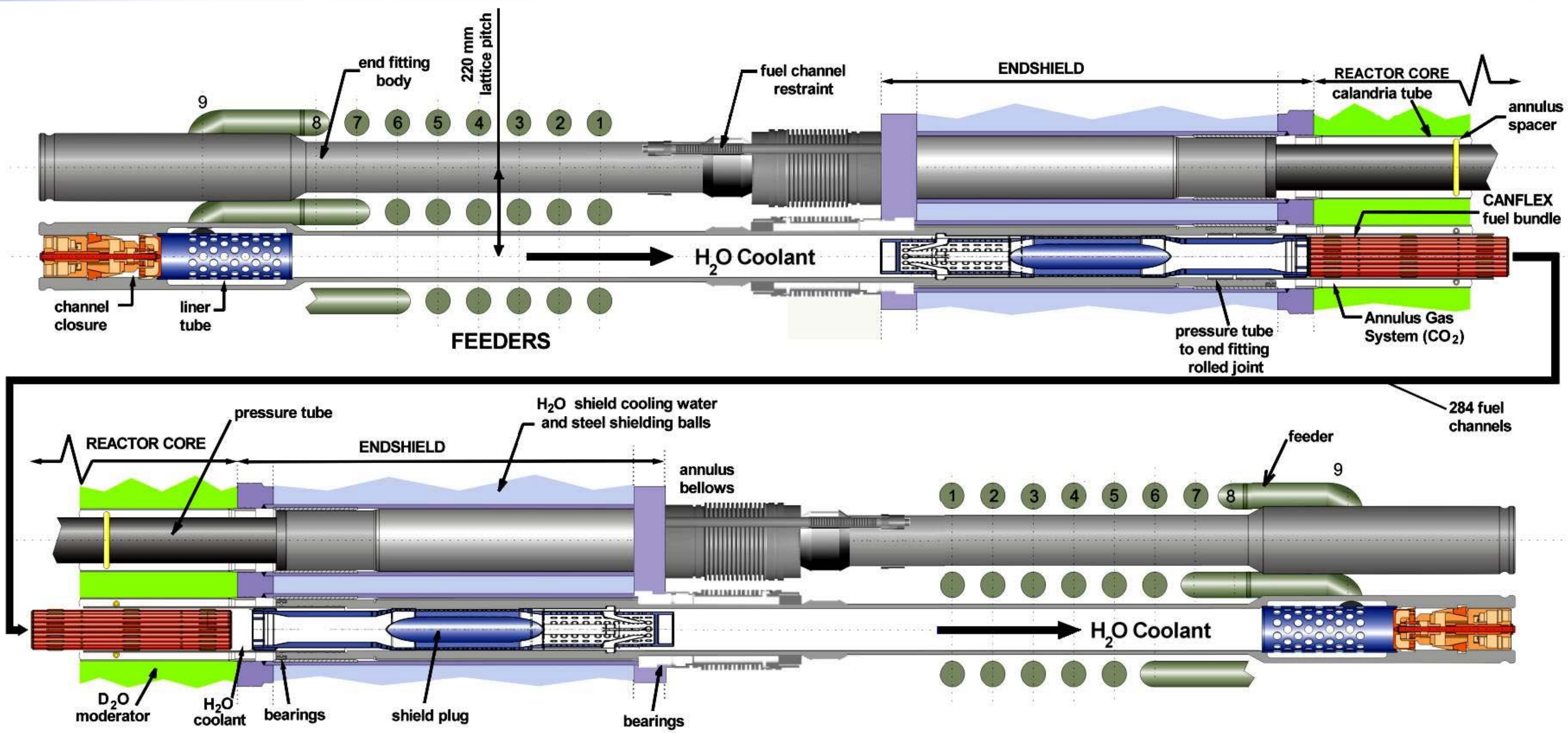
ACR Fuel Channel Details



Fuel is uranium oxide clad with Zircaloy-4

Moderator is unpressurized and below 100°C

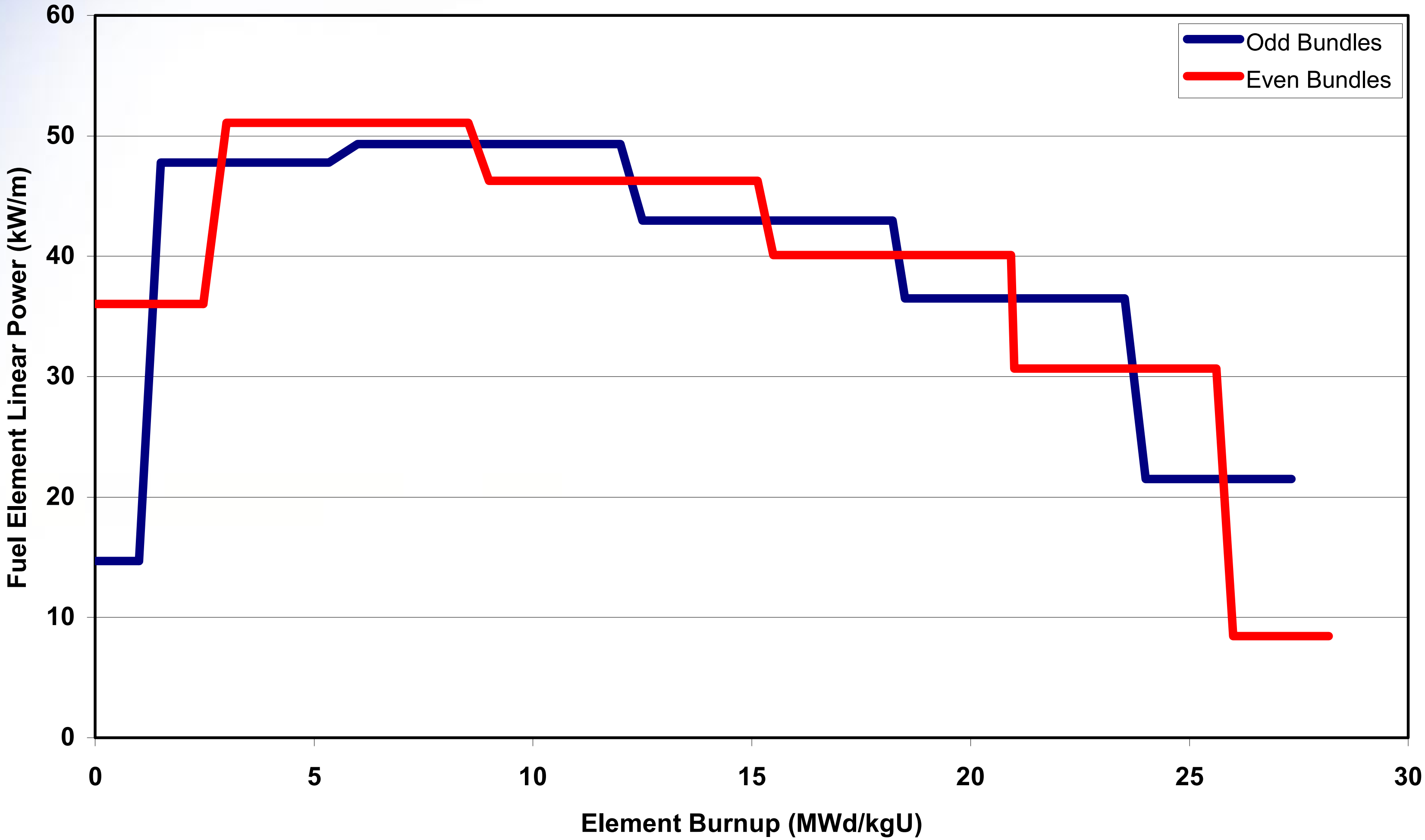
ACR Fuel Channel



ACR Fuel Operating Conditions

- Thin-walled cladding collapsed onto fuel pellets during normal operation
- Fuel element linear power ratings in high power bundle (kW/m)
  - Center: 15, inner: 40, intermediate: 35, outer: 47
- Maximum burnup ≤ 30 MWd/kgU
- Power ratings and burnups within current CANDU and LWR operating experience range
- On-power fueling
- 2-Bundle shifts

ACR Outer Element Power and Burnup in 8.0 MW Channel



Outline

- ACR Fuel Design
- **Fuel Behavior in Limited and Severe Core Damage Accidents**
- Experimental Database
- Computer Codes

Fuel Behavior in Limited and Severe Core Damage Accidents

- **Temperature increases due to reduced cooling, stored energy and decay power**
- **Fuel cladding balloons due to difference between internal gas pressure and coolant pressure**
  - **Localized due to small free volume**
- **Fuel cladding failure by various mechanisms**
  - **Over-strain, oxygen embrittlement, etc.**
- **Zircaloy fuel cladding oxidized by steam, generating H_2**
- **Fuel thermal cracking and mechanical relocation**
- **Fuel oxidized by steam ingress into fuel element**
- **Zircaloy/UO_2 interaction and dissolution**

Fuel Behavior Phenomena (1)

- Fission and Decay Heating
- Diffusion of Heat in Fuel
- Fuel-to-Cladding Heat Transfer
- Fuel-to-End Cap Heat Transfer
- Fission Gas Release to Gap and Internal Pressurization
- Cladding Deformation
- Cladding Failure
- Fuel Deformation

Fuel Behavior Phenomena (2)

- Cladding Oxidation or Hydriding
- Fuel Oxidation or Reduction
- Fuel or Cladding Melting and Relocation
- Bundle Mechanical Deformation
- Cladding-to-Coolant and Coolant-to-Pressure Tube Heat Transfer
- Element-to-Pressure Tube Radiative Heat Transfer

Outline

- ACR Fuel Design
- Fuel Behavior in Limited and Severe Core Damage Accidents
- **Experimental Database**
- Computer Codes

Experimental Database (1)

- Laboratory separate-effects tests
  - Fuel cladding ballooning tests
  - Cladding oxidation tests
  - UO_2 – Zircaloy interaction and dissolution tests
  - Cladding embrittlement tests
- In-reactor tests under normal operating conditions
  - Fuel-centerline temperature measurements
  - Fuel rod internal gas pressure measurements
  - Defected fuel rod behavior tests

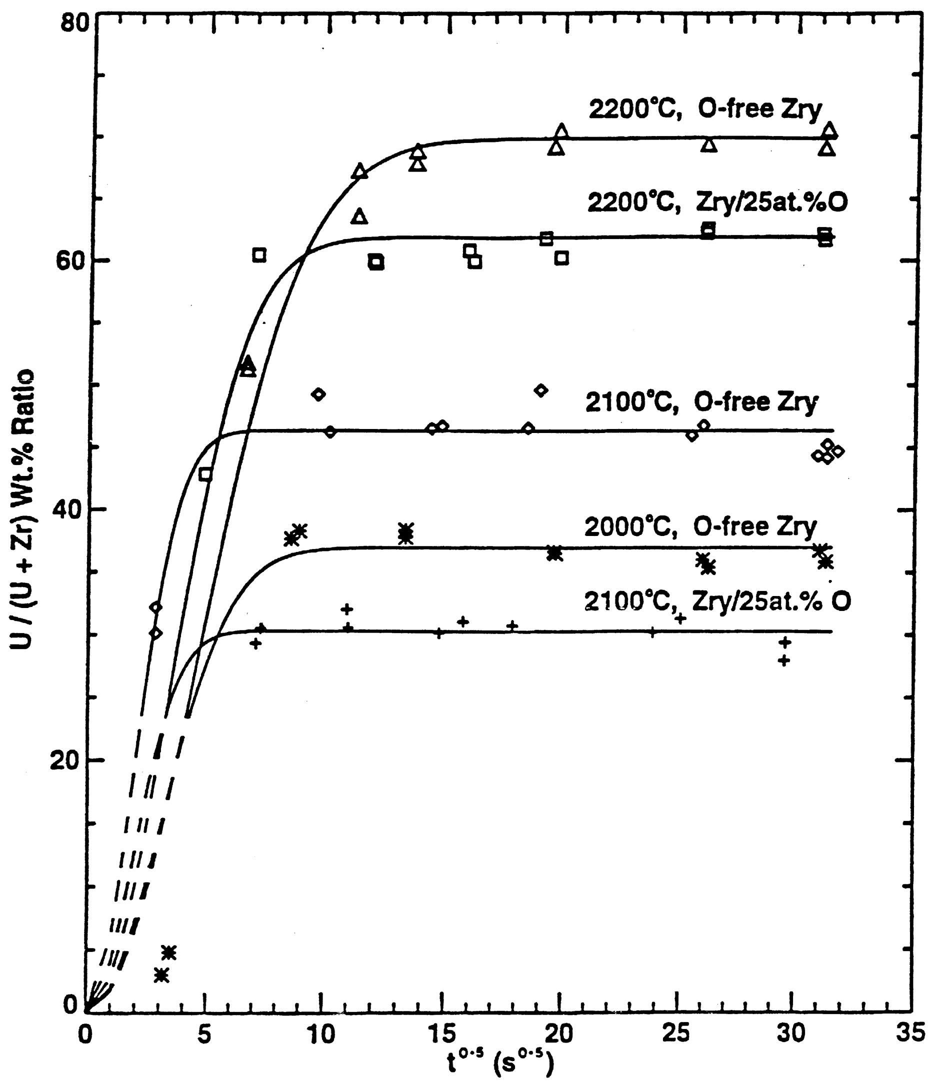
Experimental Database (2)

- **In-reactor tests under accident conditions**
  - **Canadian in-reactor blowdown tests (X-2)**
  - **Canadian severe-fuel-damage tests (BTF)**
  - **International severe accident tests (PBF SFD, FLHT, Phebus)**

UO_2 – Zircaloy Dissolution Tests

- Dissolution of UO_2 in molten Zircaloy measured
  - Temperatures: 2000 to 2500°C
  - Zircaloy and Zircaloy containing 25 at% oxygen

UO_2 – Zircaloy Dissolution Tests



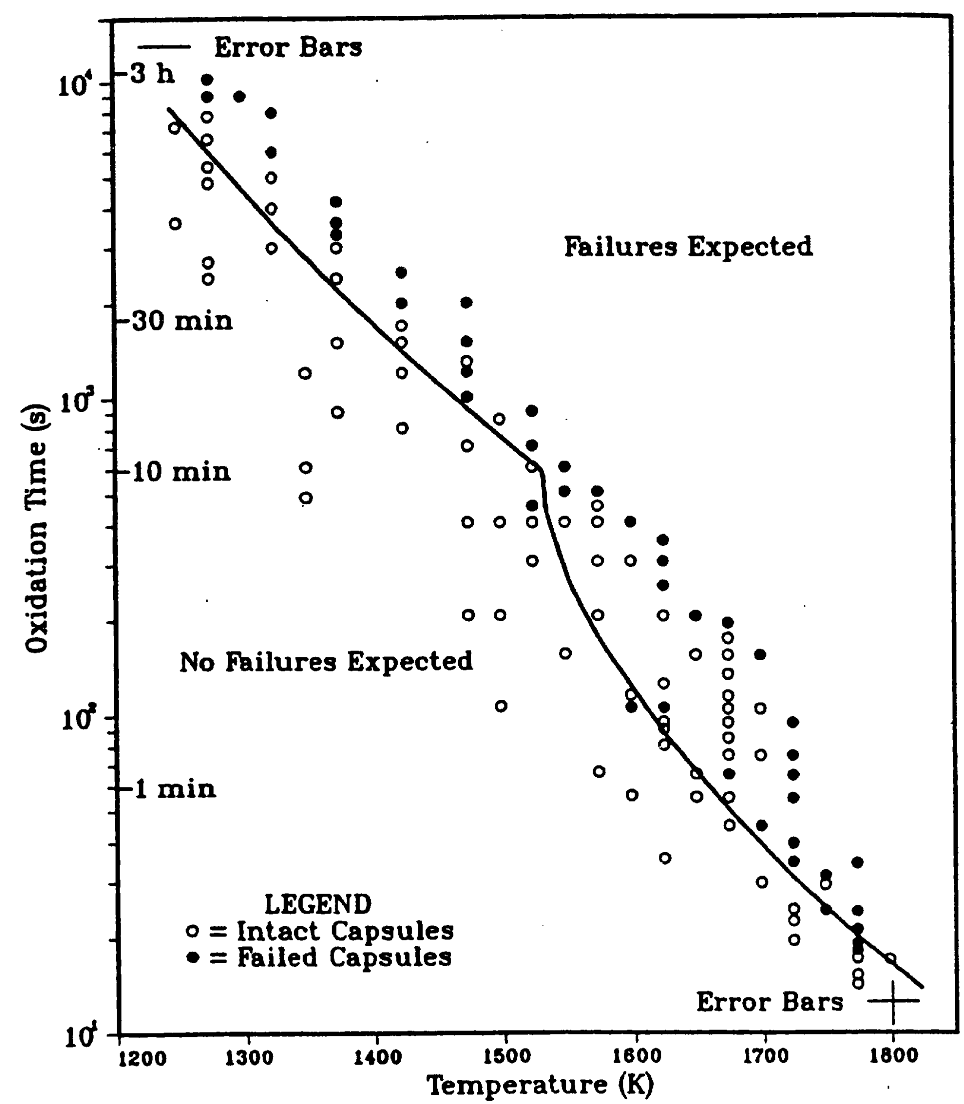
UO_2 – Zircaloy Dissolution Tests

- UO_2 dissolves in molten Zircaloy and oxygenated Zircaloy within a few minutes at temperatures of 2000 to 2500°C
- No subsequent increase in uranium content of the melt
- Higher solubilities are observed at higher temperatures

Cladding Embrittlement Tests

- **Tests performed to develop failure criteria for CANDU fuel cladding degraded by oxidation and subjected to thermal-quench loads**
- **Fuel cladding oxidized in steam at temperatures between 1250 and 1800 K and quenched with water**

Cladding Embrittlement Tests



Cladding Embrittlement Tests

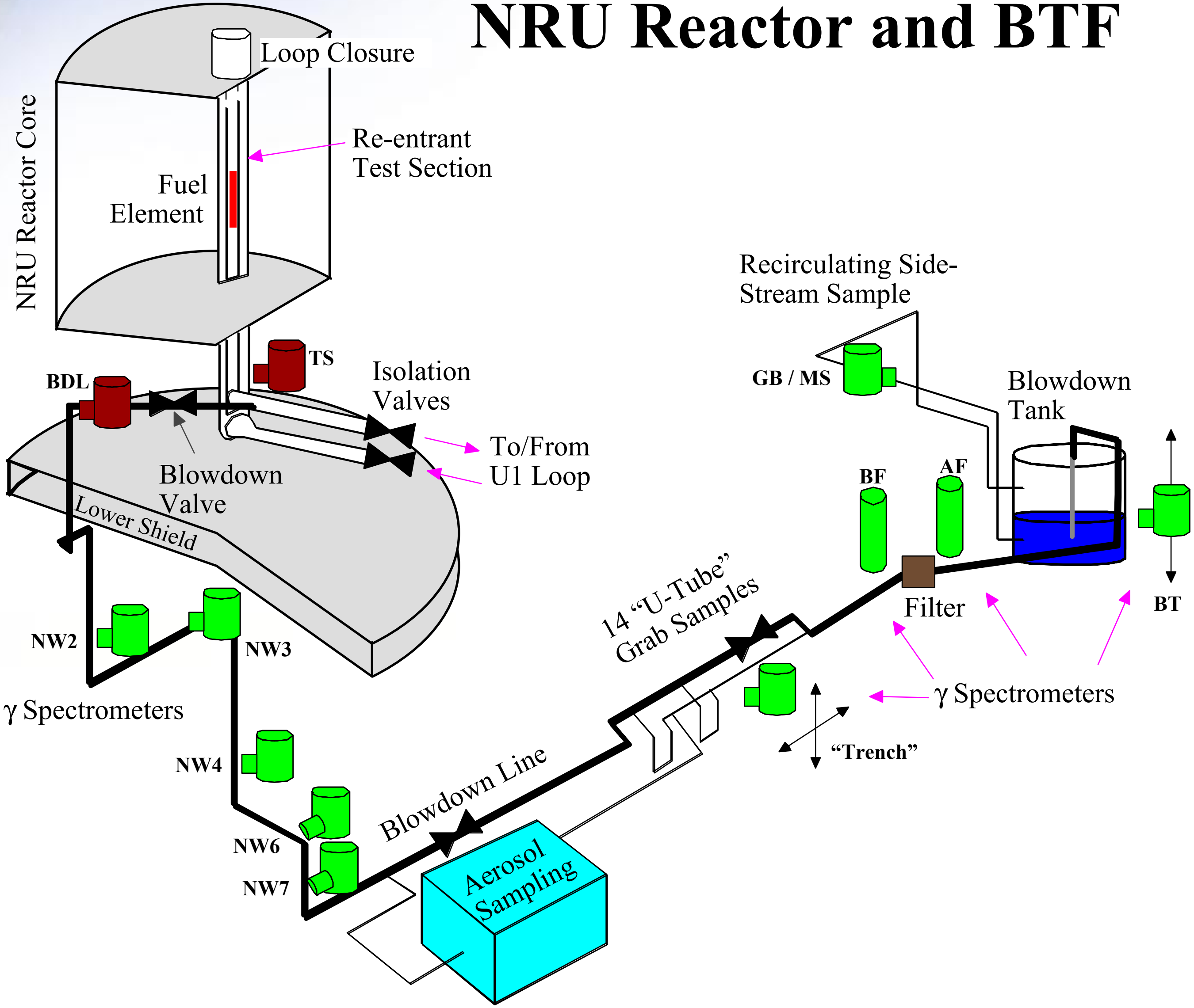
- Results support Sawatzky's criterion
  - Cladding may fail if less than half the cladding thickness has an oxygen concentration less than 0.7 wt%

Blowdown Test Facility

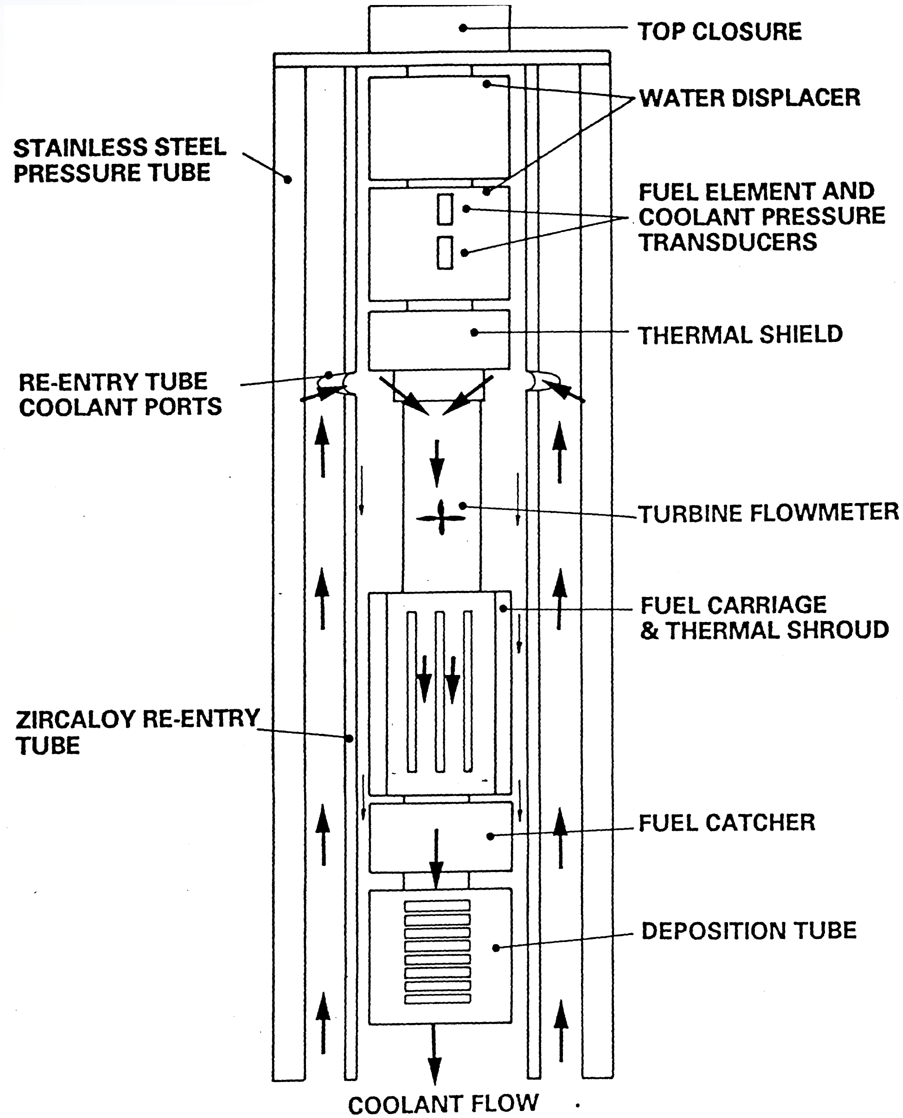
Research Program Goals

- Provide data from integral in-reactor experiments for use in the validation of computer codes used for safety analyses and licensing of CANDU reactors
- Verify our understanding of CANDU fuel behavior and FP release & transport under high temperature conditions representative of severe-fuel-damage accident scenarios

NRU Reactor and BTF



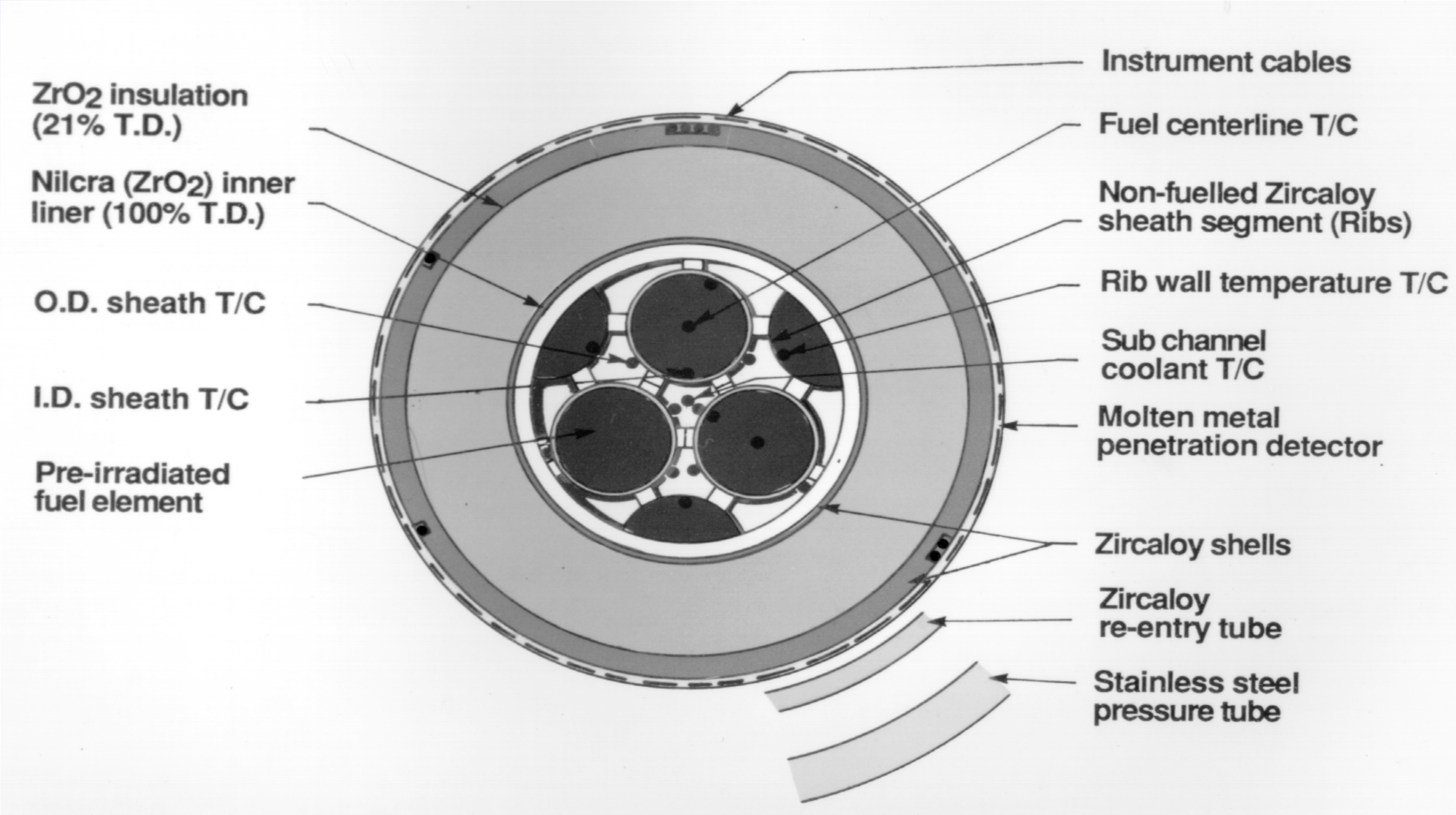
BTF Test Section



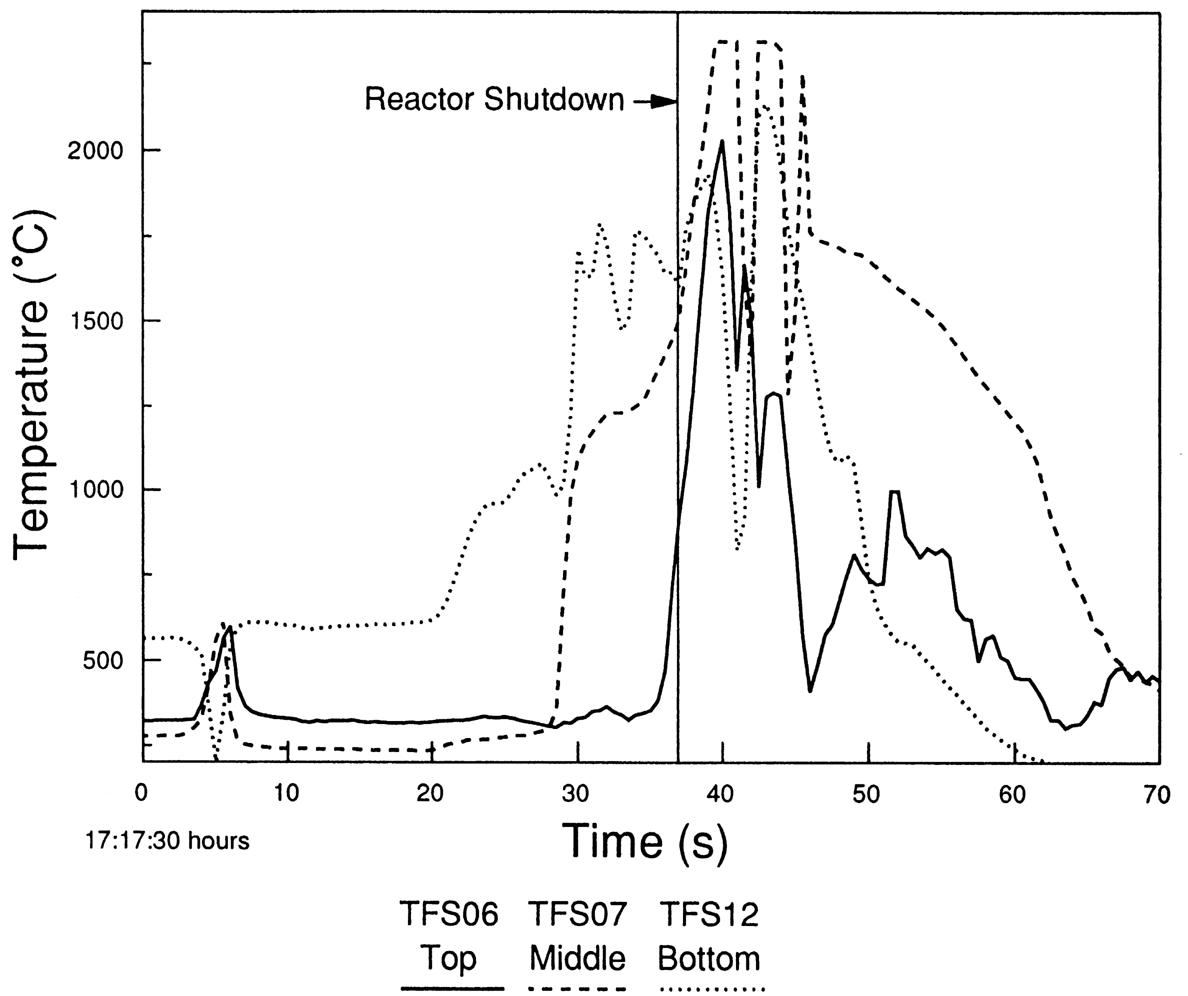
Summary of BTF Test Conditions

Parameter	BTF-107 Test	BTF-104 Test	BTF-105A Test	BTF-105B Test
Fuel elements	1 pre-irradiated, 2 fresh	1 pre-irradiated	1 fresh	1 pre-irradiated
Pre-transient cooling	Pressurized water	Saturated steam	Saturated steam	Saturated steam
Maximum fuel temperature (K)	≥ 2770 (peak)	~ 2100 (volume-average)	~ 2100 (volume-average)	~ 2100 (volume-average)
Transient duration (s)	~ 70	~ 2100	~ 2900	~ 4200
Time at high temperature after fuel failure (s)	~ 20	~ 1500	< 60	~ 2400

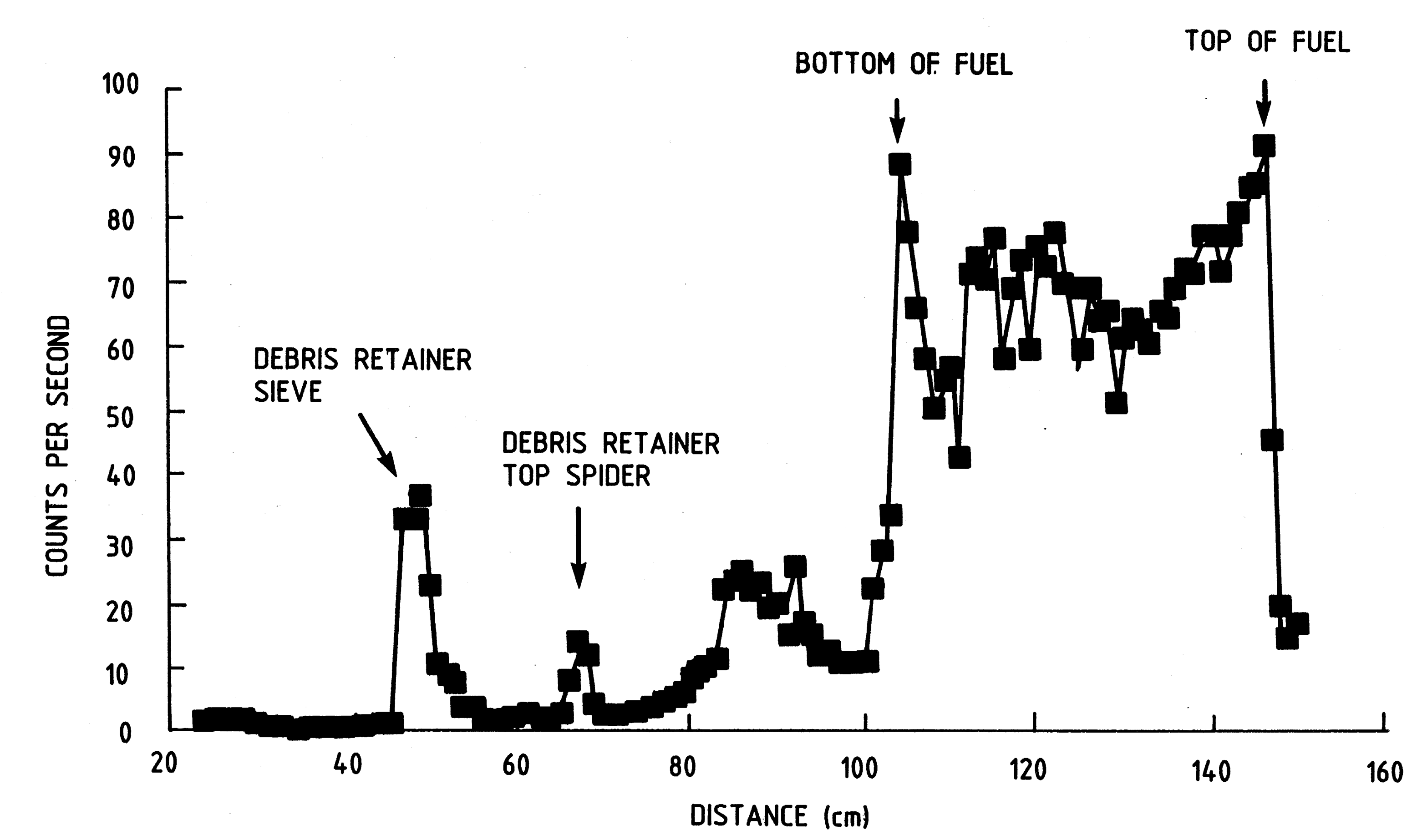
BTF-107 Fuel Assembly



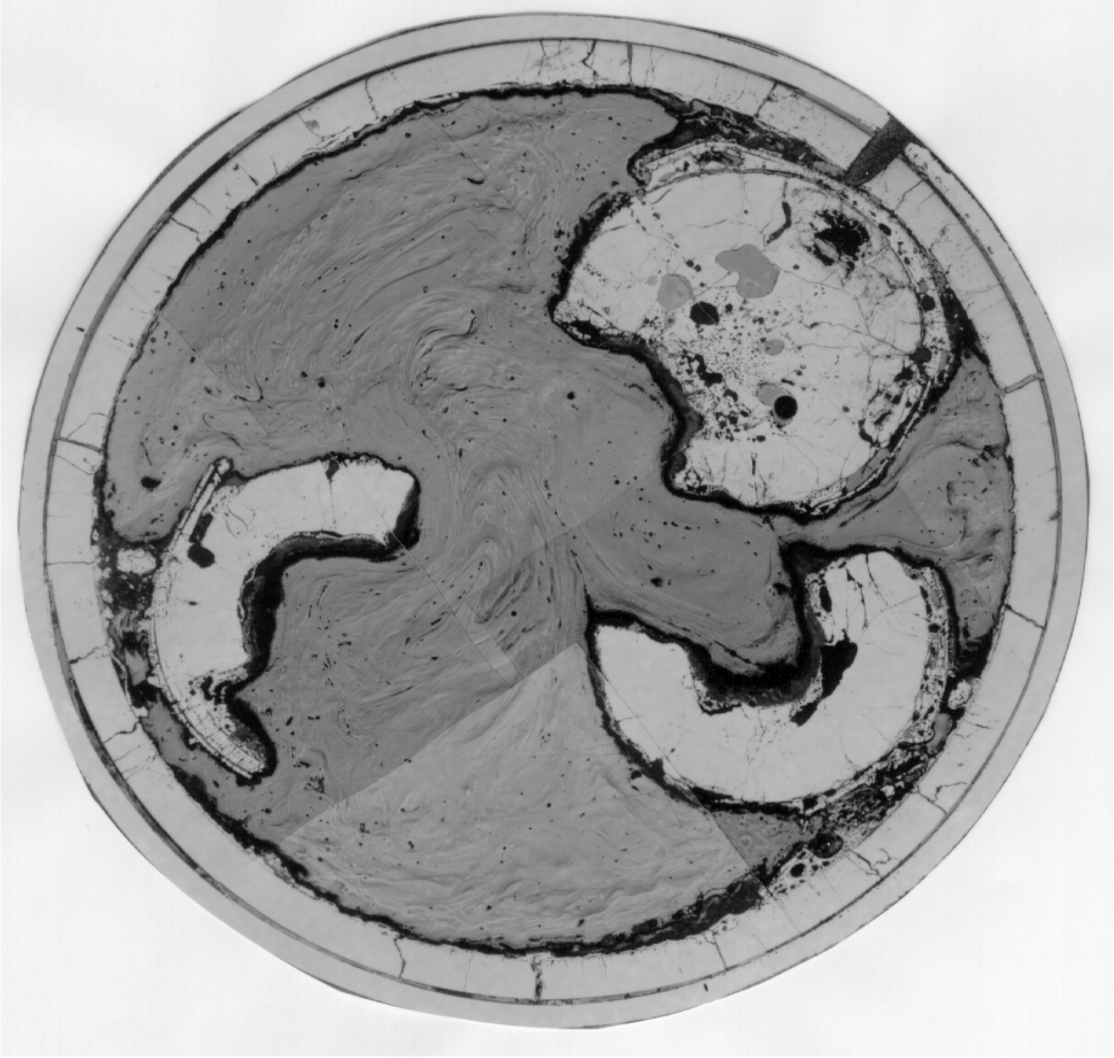
BTF-107 Cladding Temperatures



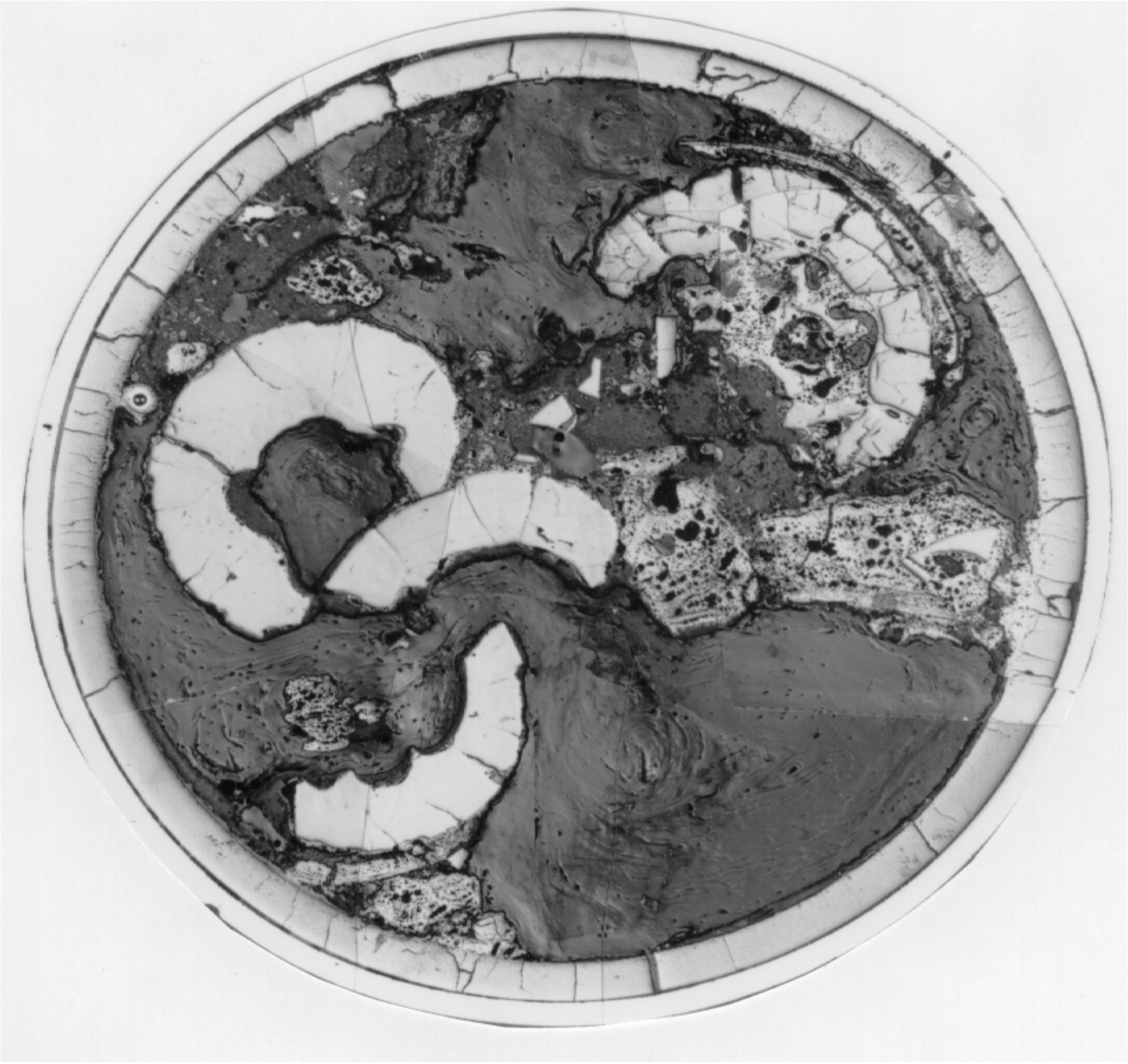
BTF-107 Post-Test ^{140}La Gamma-Scan



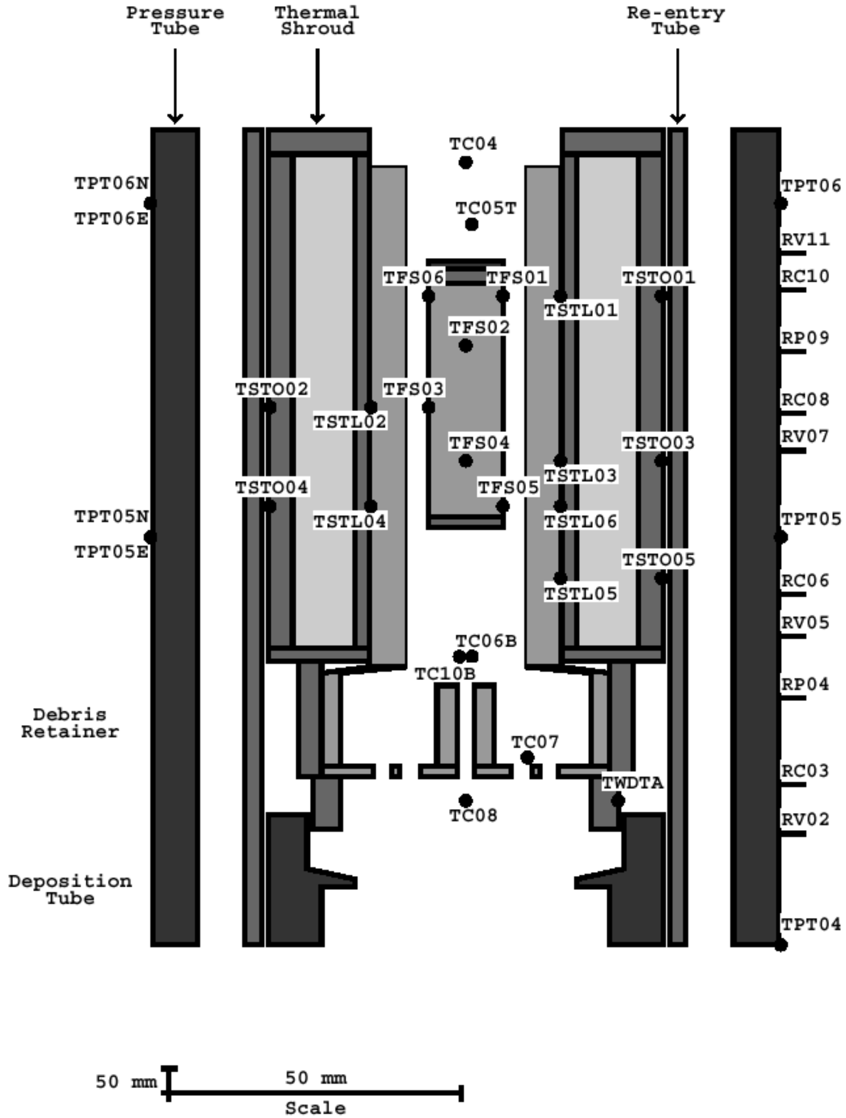
BTF-107 Post-Irradiation Examination (PIE)



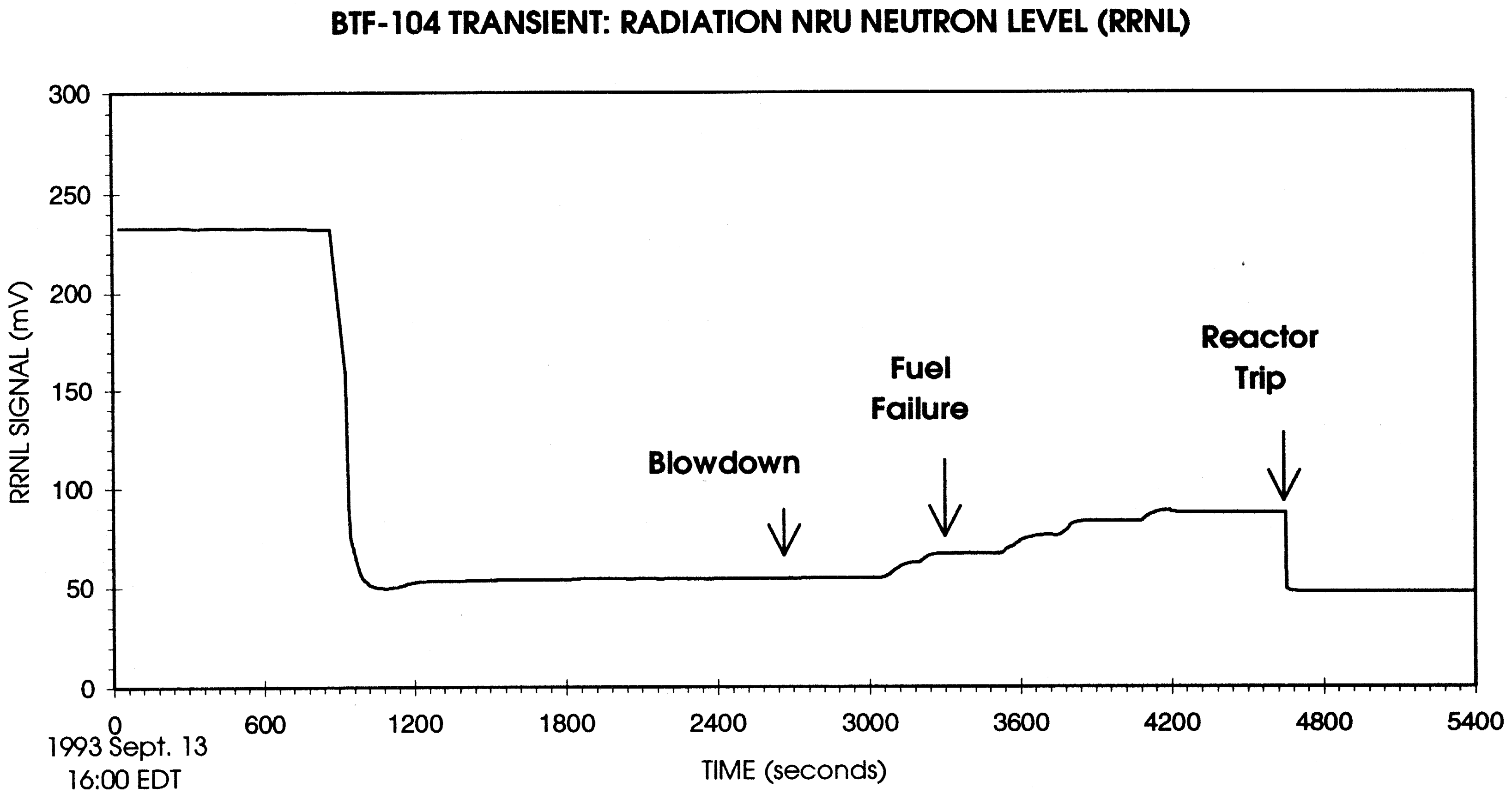
BTF-107 PIE



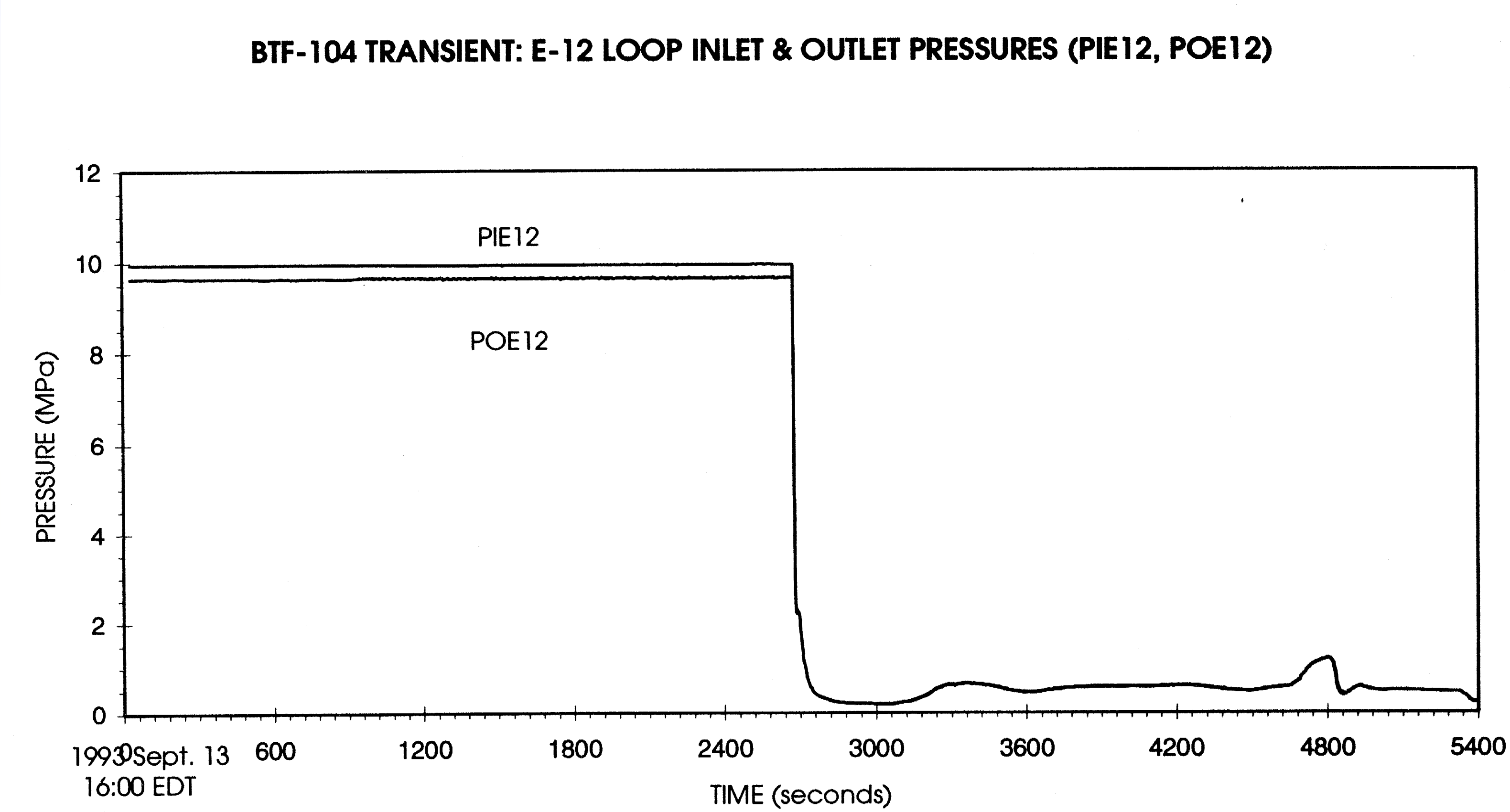
BTF-104 Fuel Assembly



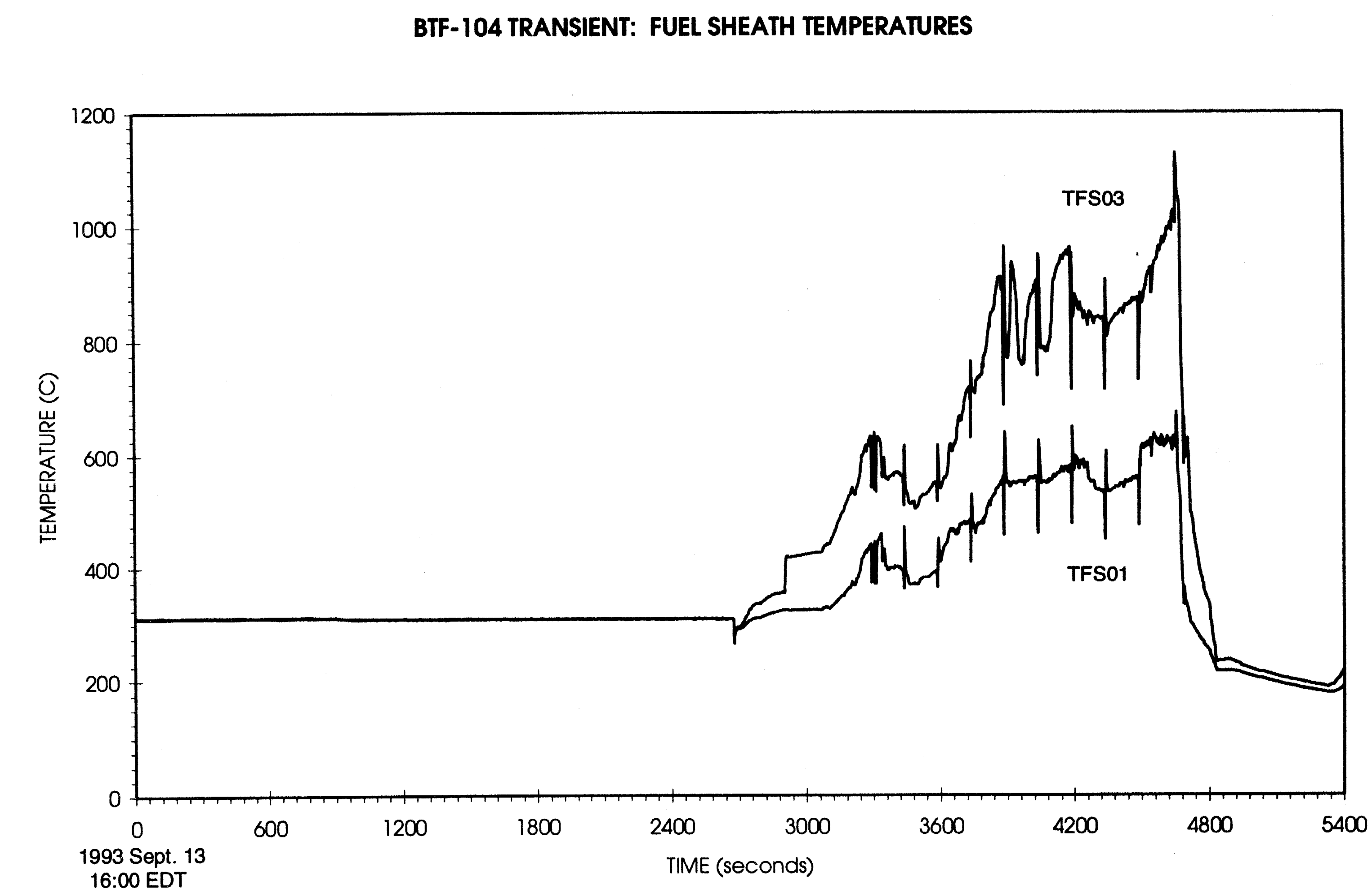
BTF-104 Reactor Power



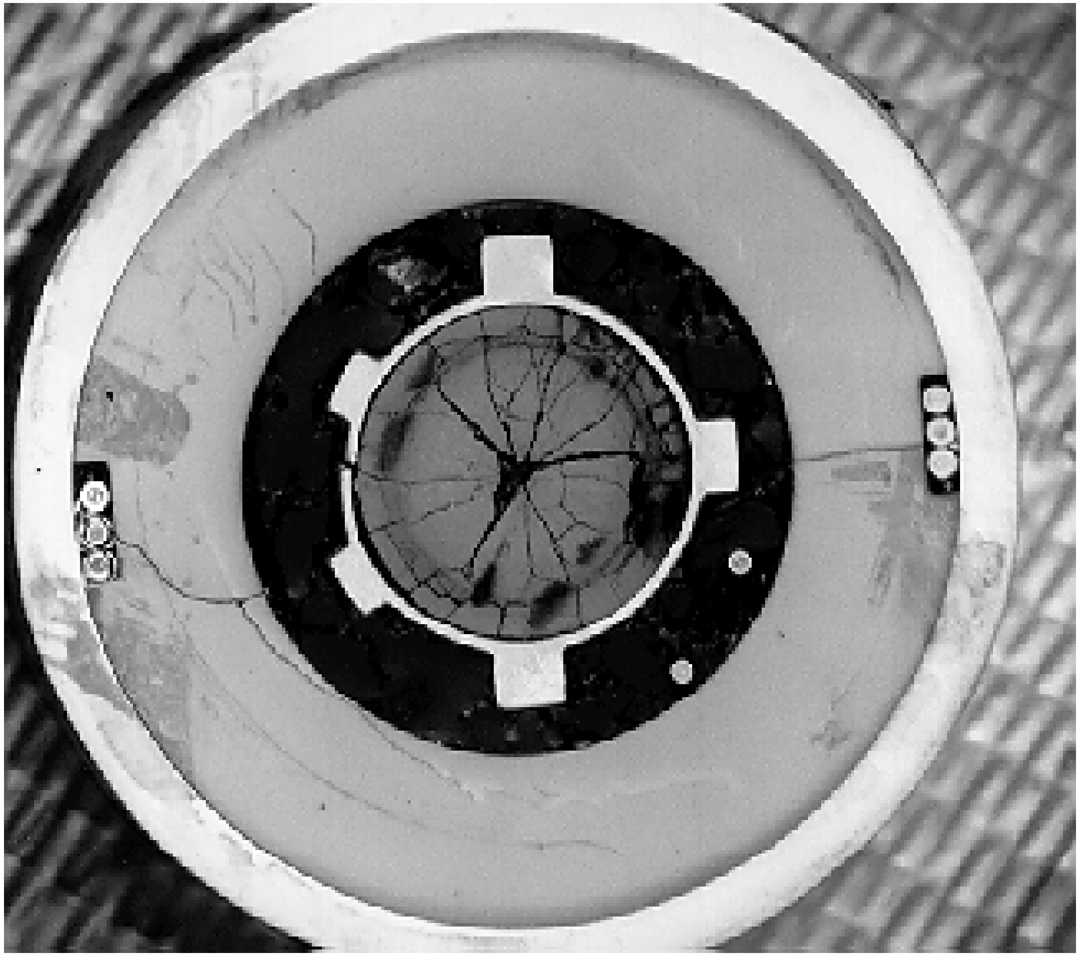
BTF-104 Coolant Pressure



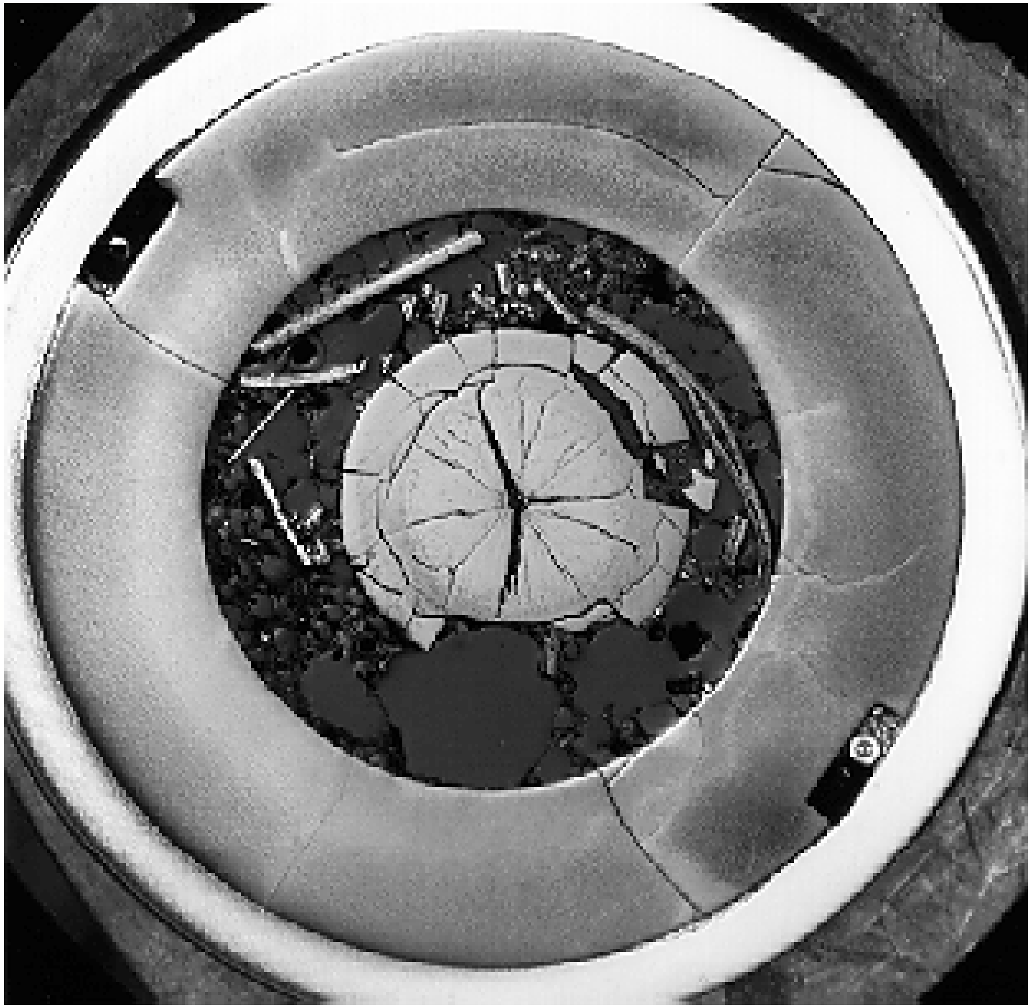
BTF-104 Fuel Cladding Temperature



BTF-104 PIE, Elevation 252 mm



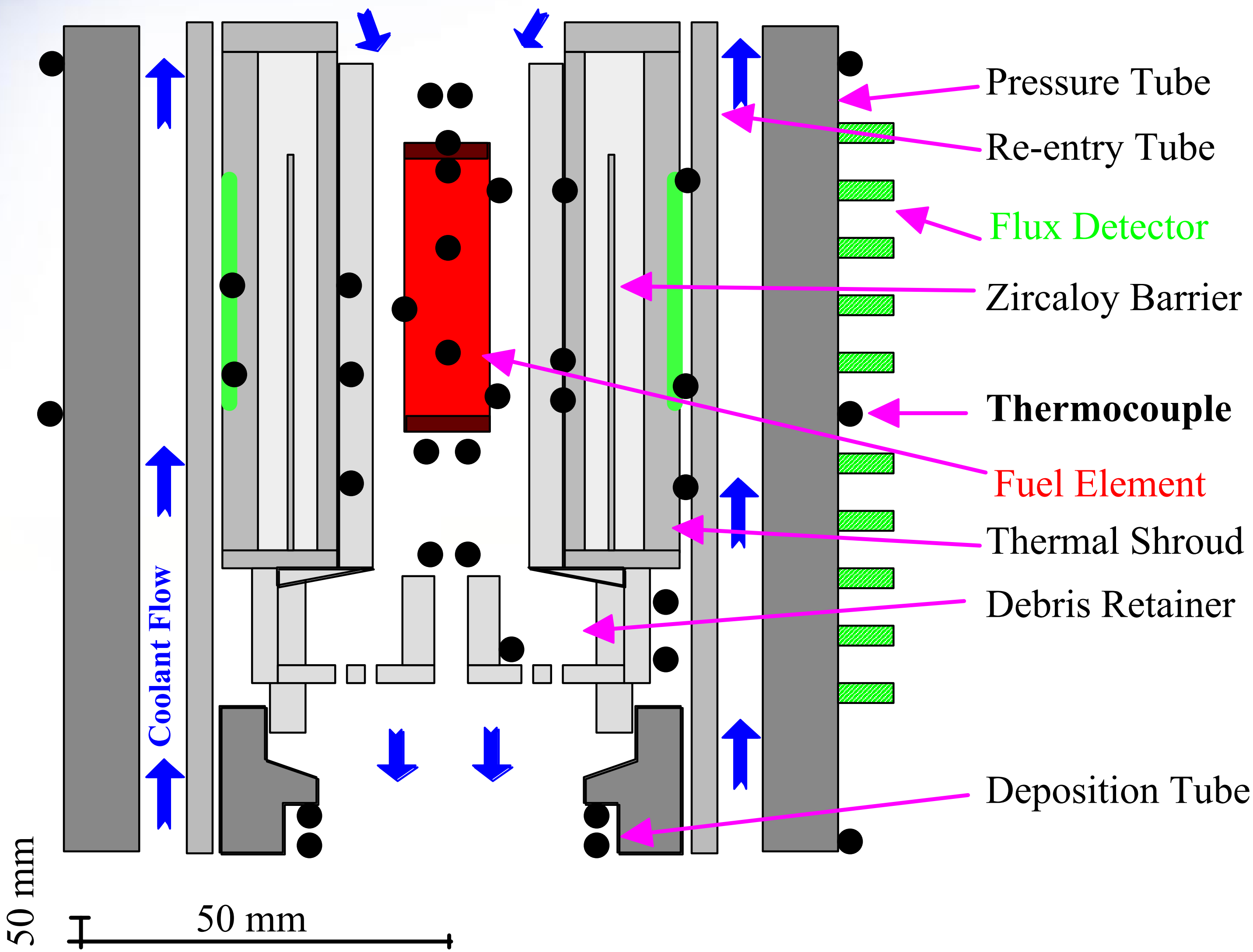
BTF-104 PIE, Elevation 36 mm



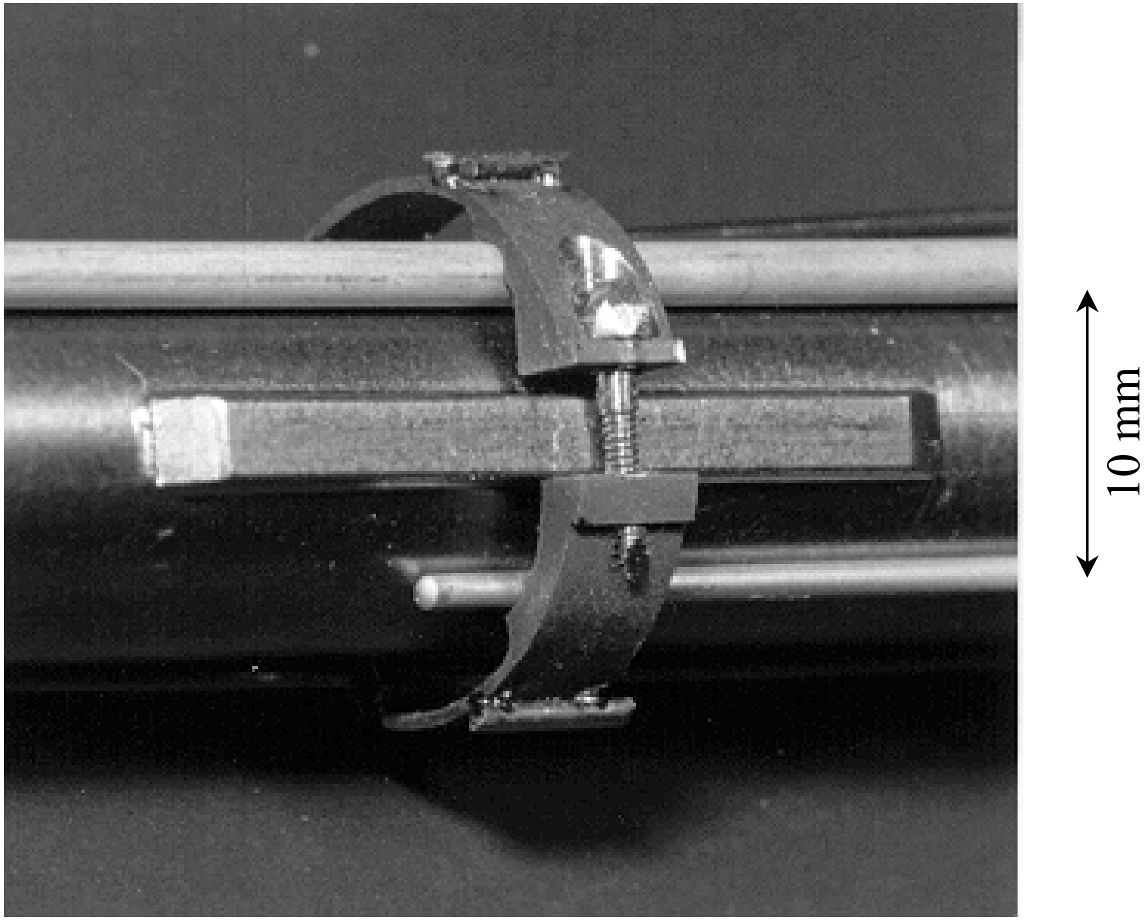
BTF-105A Objectives

- Test instrumentation and procedures planned for use in BTF-105B
- Obtain data on the relationship between fuel-centerline and cladding temperatures under transient conditions with steam cooling

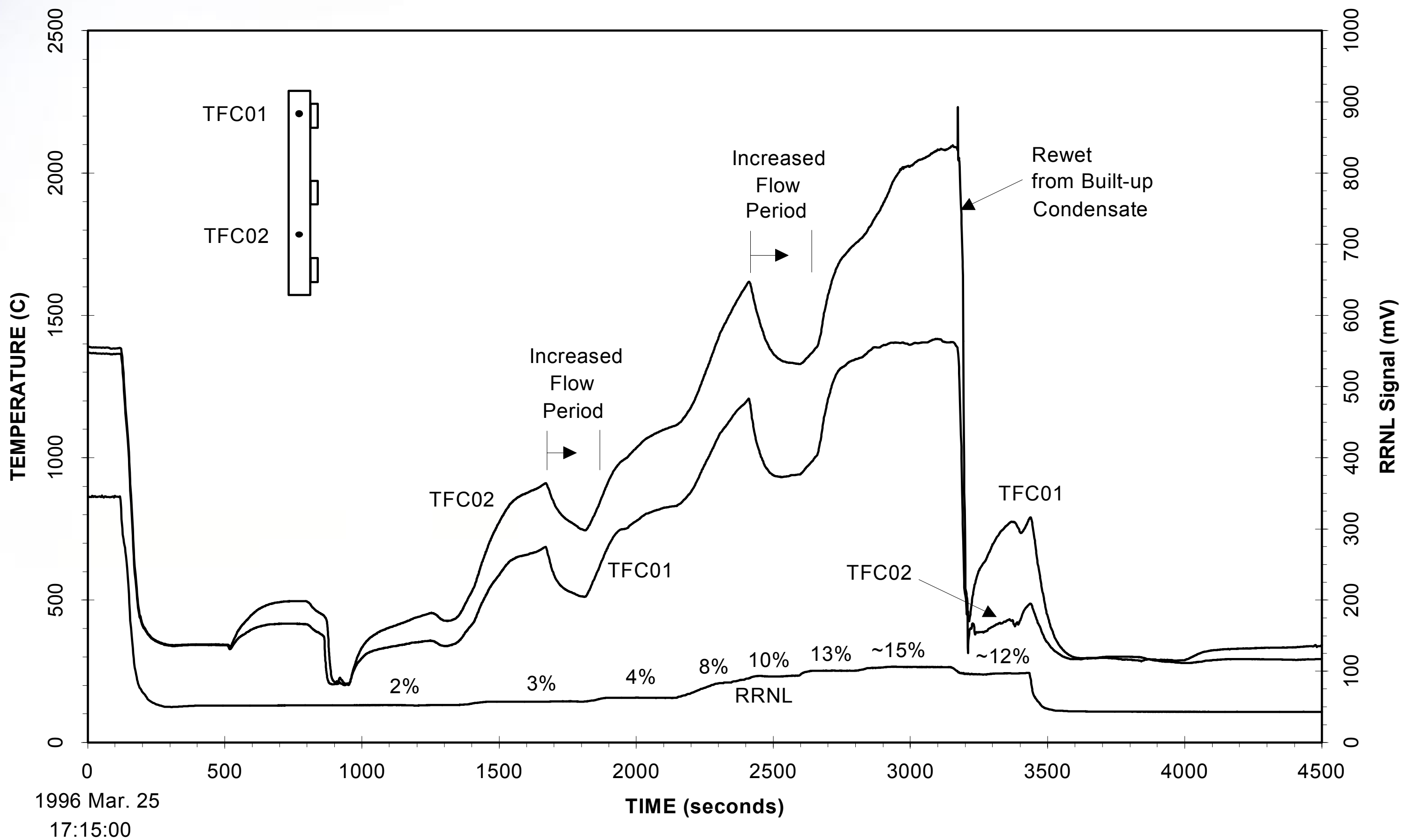
Thermocouples & Flux Detectors



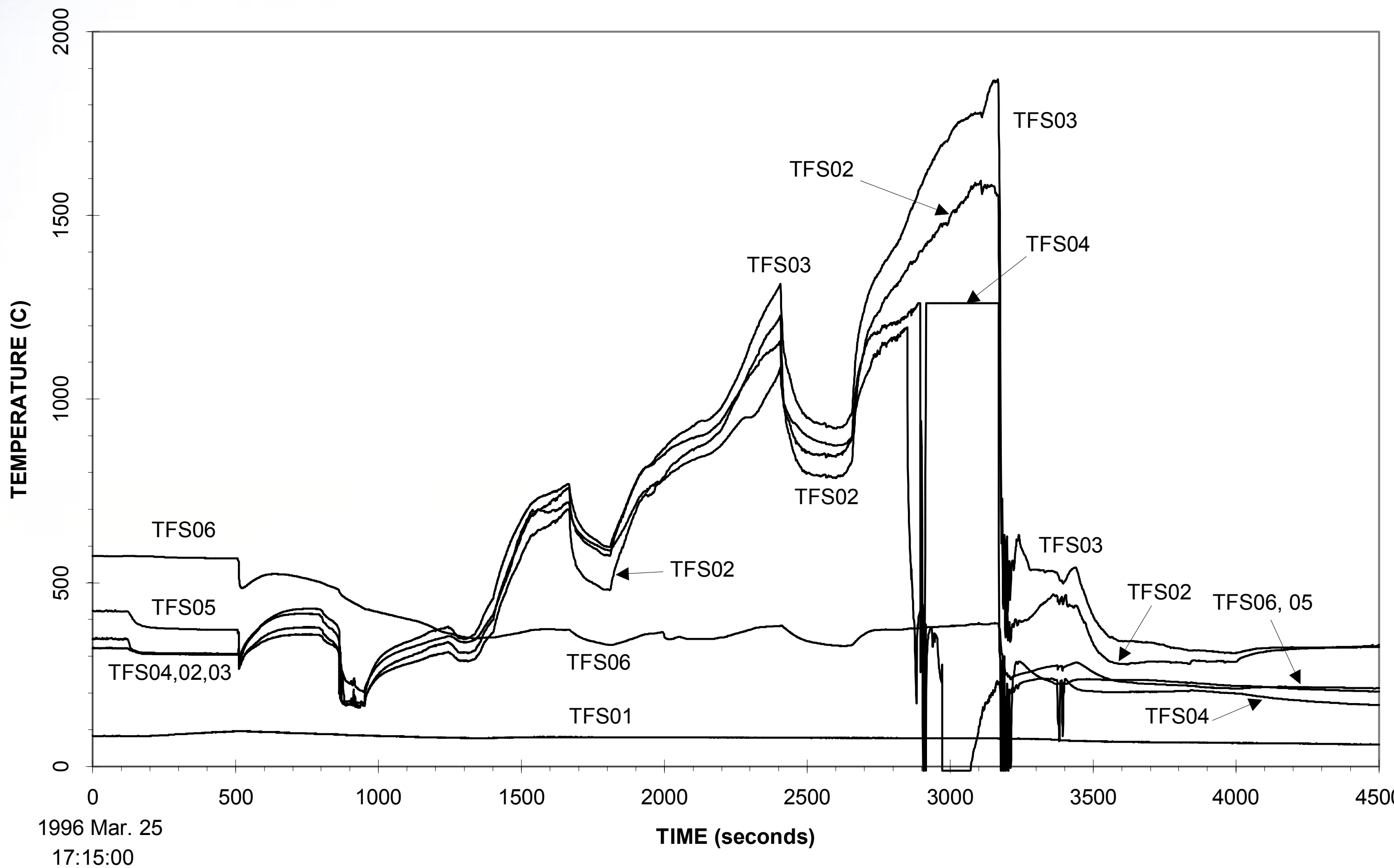
Thermocouple Clamp



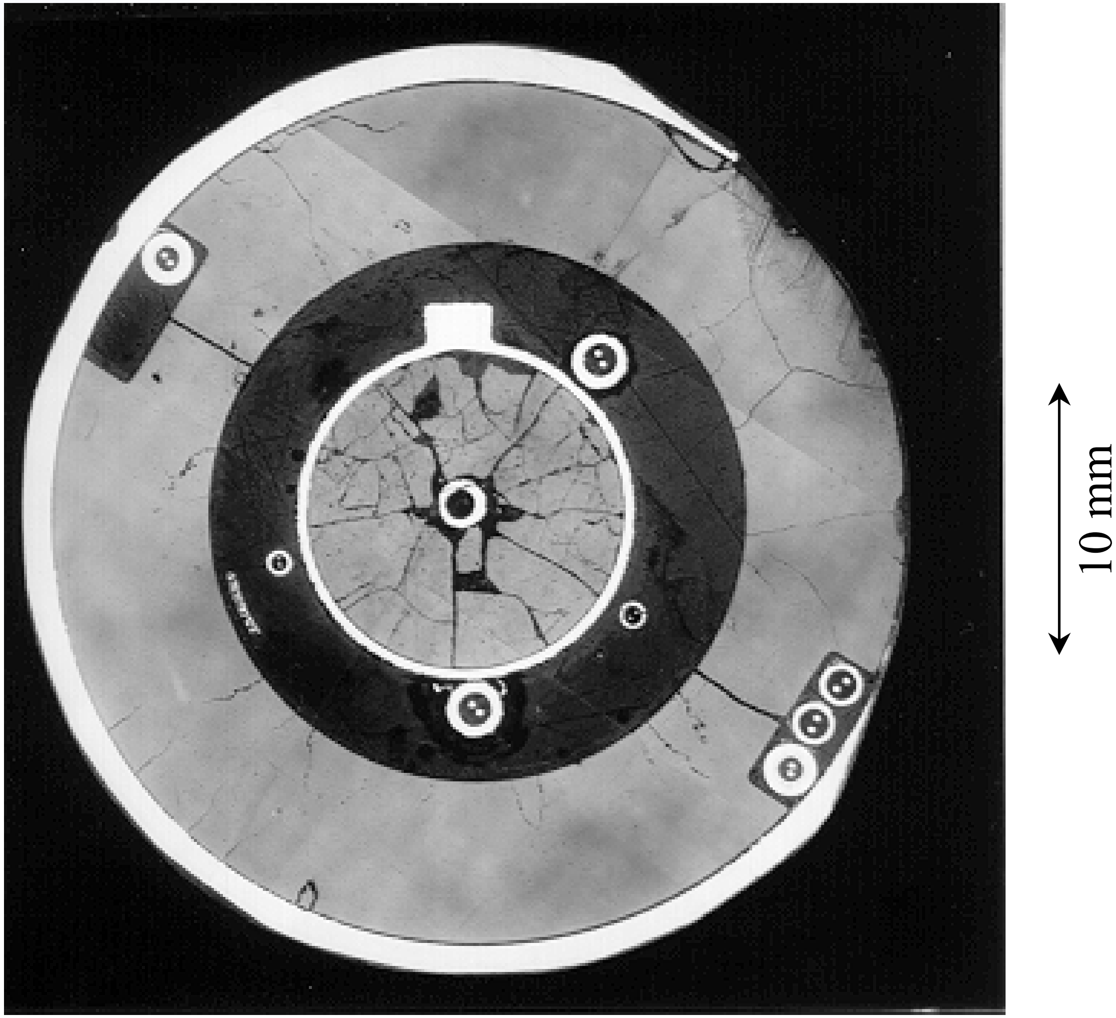
BTF-105A Fuel Centerline Temperatures



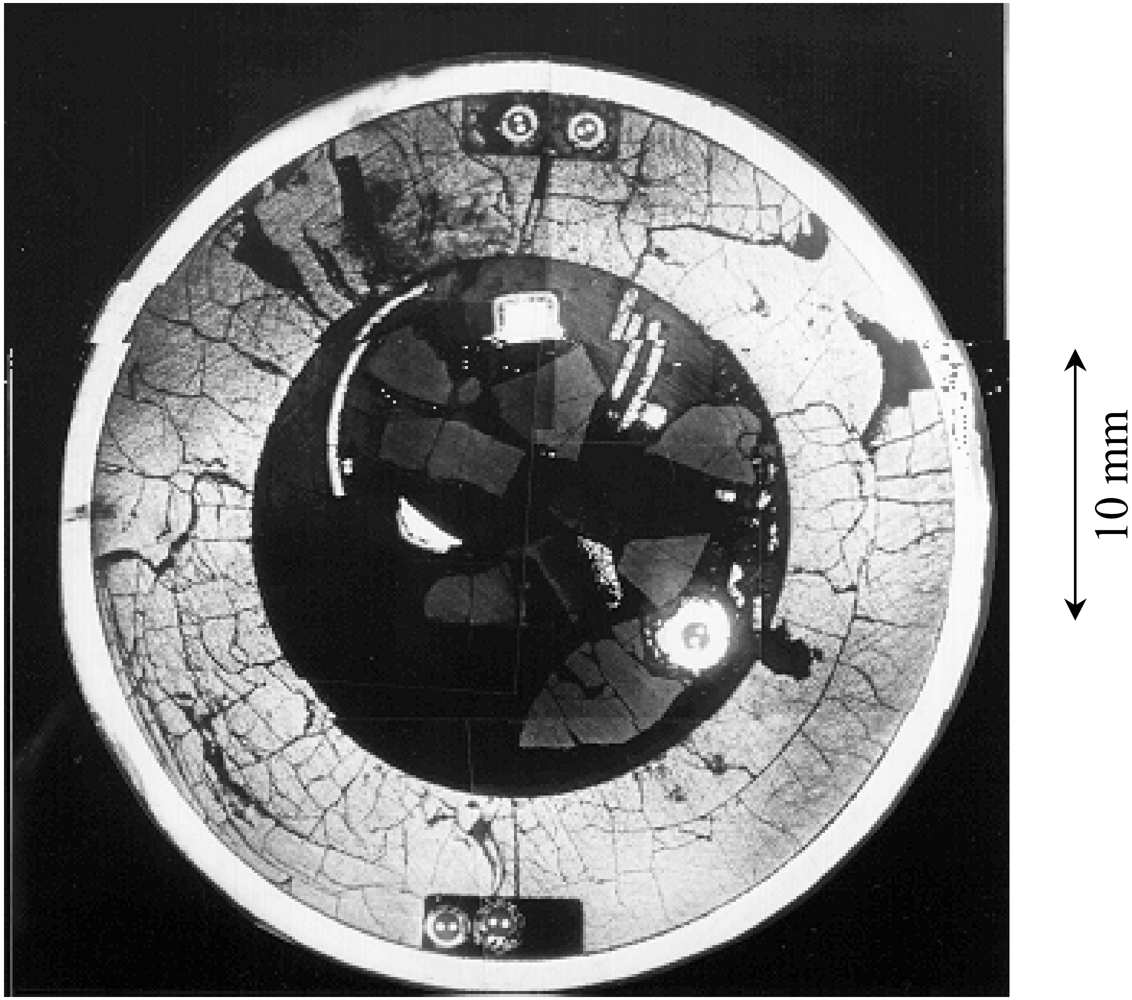
BTF-105A Fuel Cladding Temperatures



BTF-105A PIE, Elevation ~400 mm



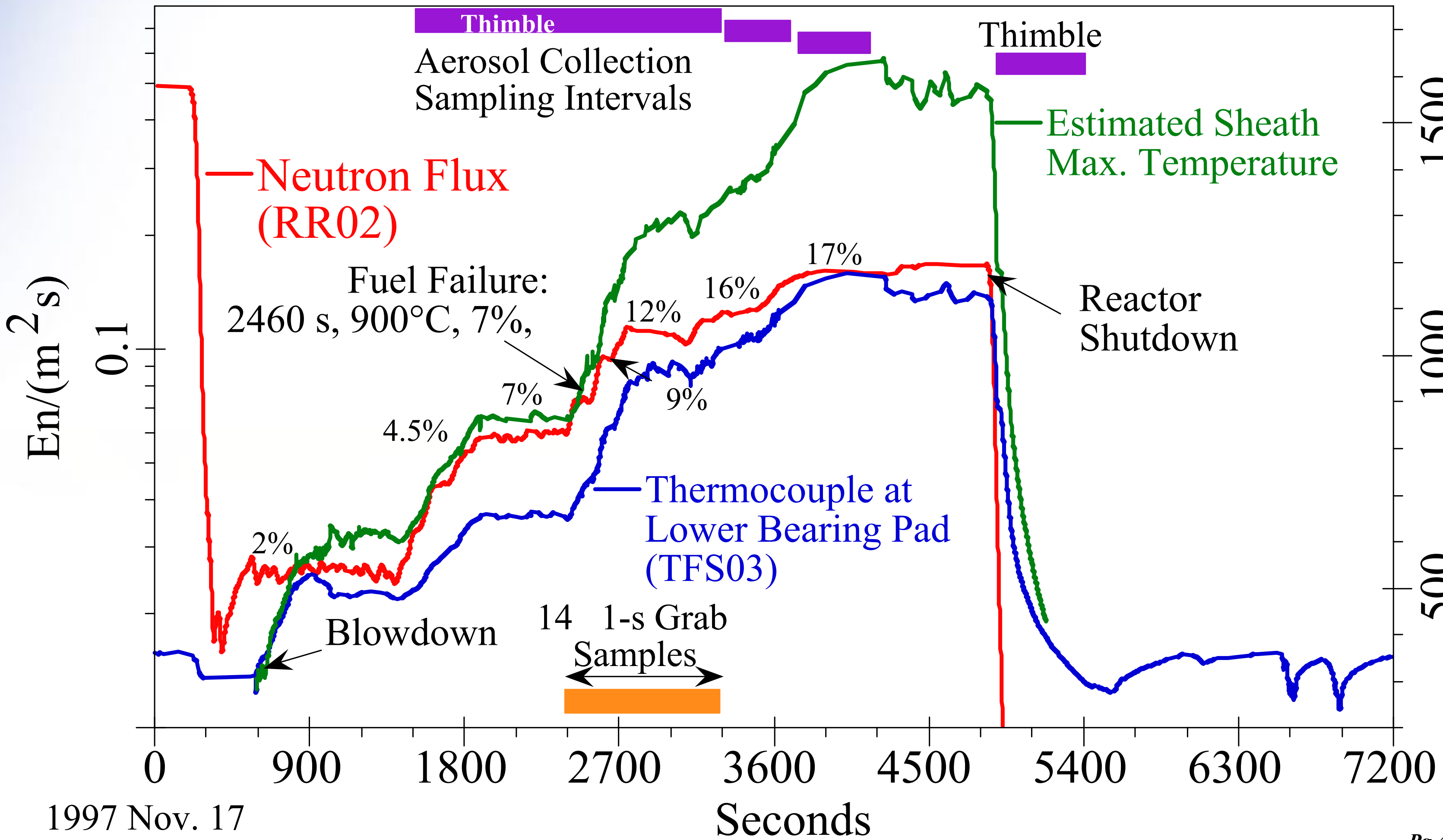
BTF-105A PIE, Elevation ~250 mm



BTF-105B Objective

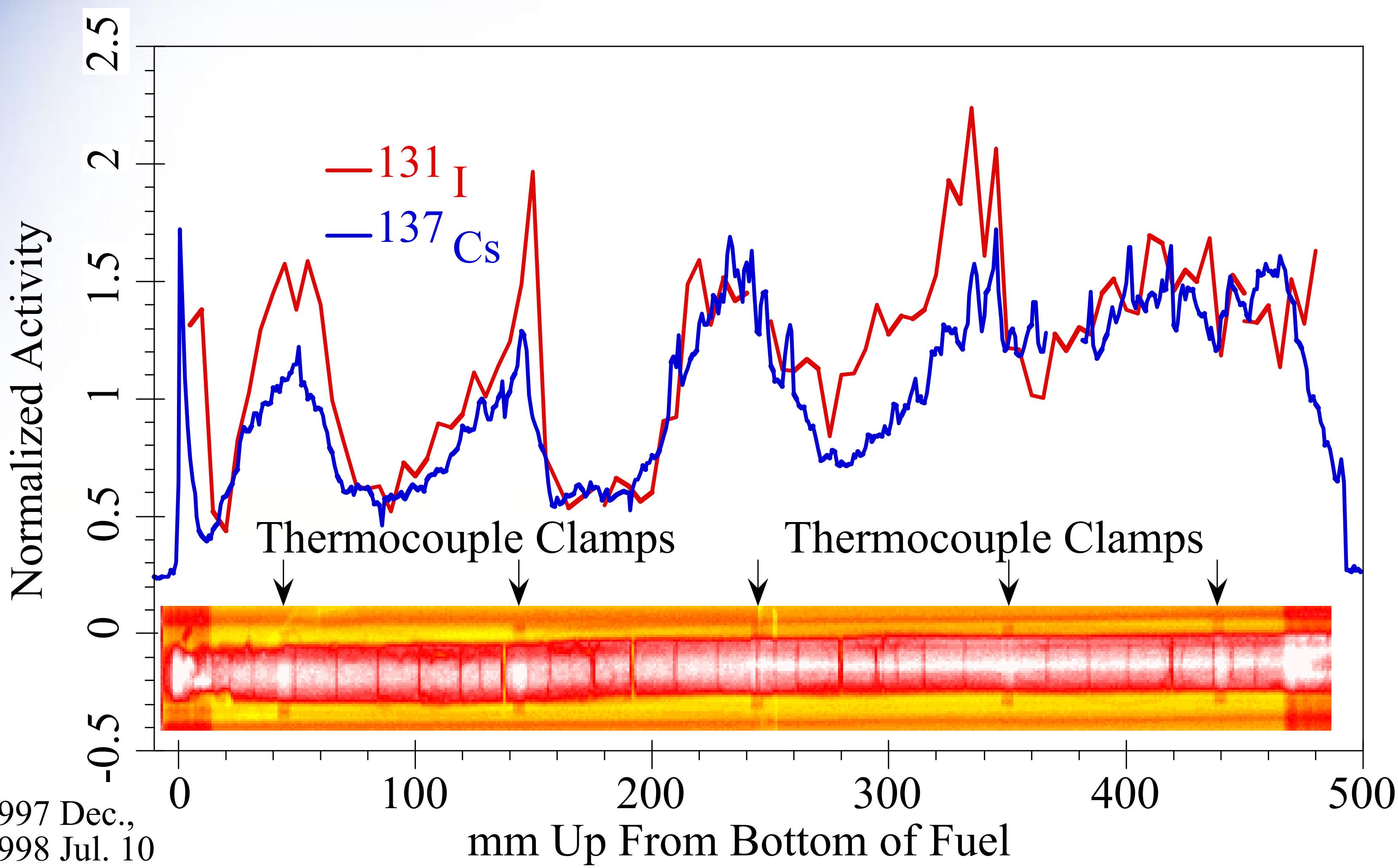
- **Measure fission product release under high temperature conditions**
  - fuel-averaged temperature target of 1800-2000°C
  - try to preserve element geometry to measure retained fission products and fuel performance
  - compromise resulted in a target fuel-averaged temperature about 1800°C for 15 minutes

Neutron Flux, Cladding Temperature



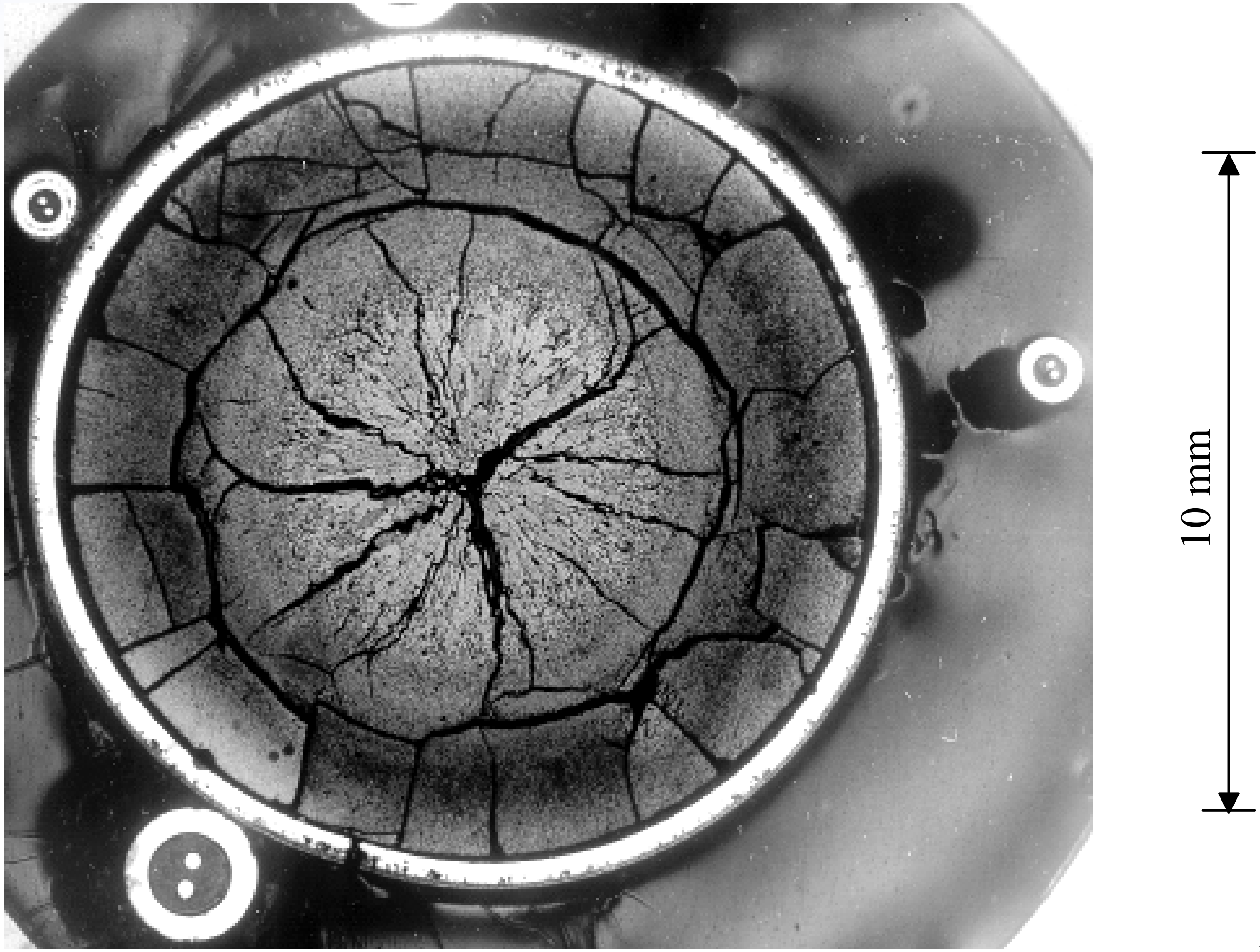
1997 Nov. 17
21h00

^{131}I, ^{137}Cs Along Fuel Element

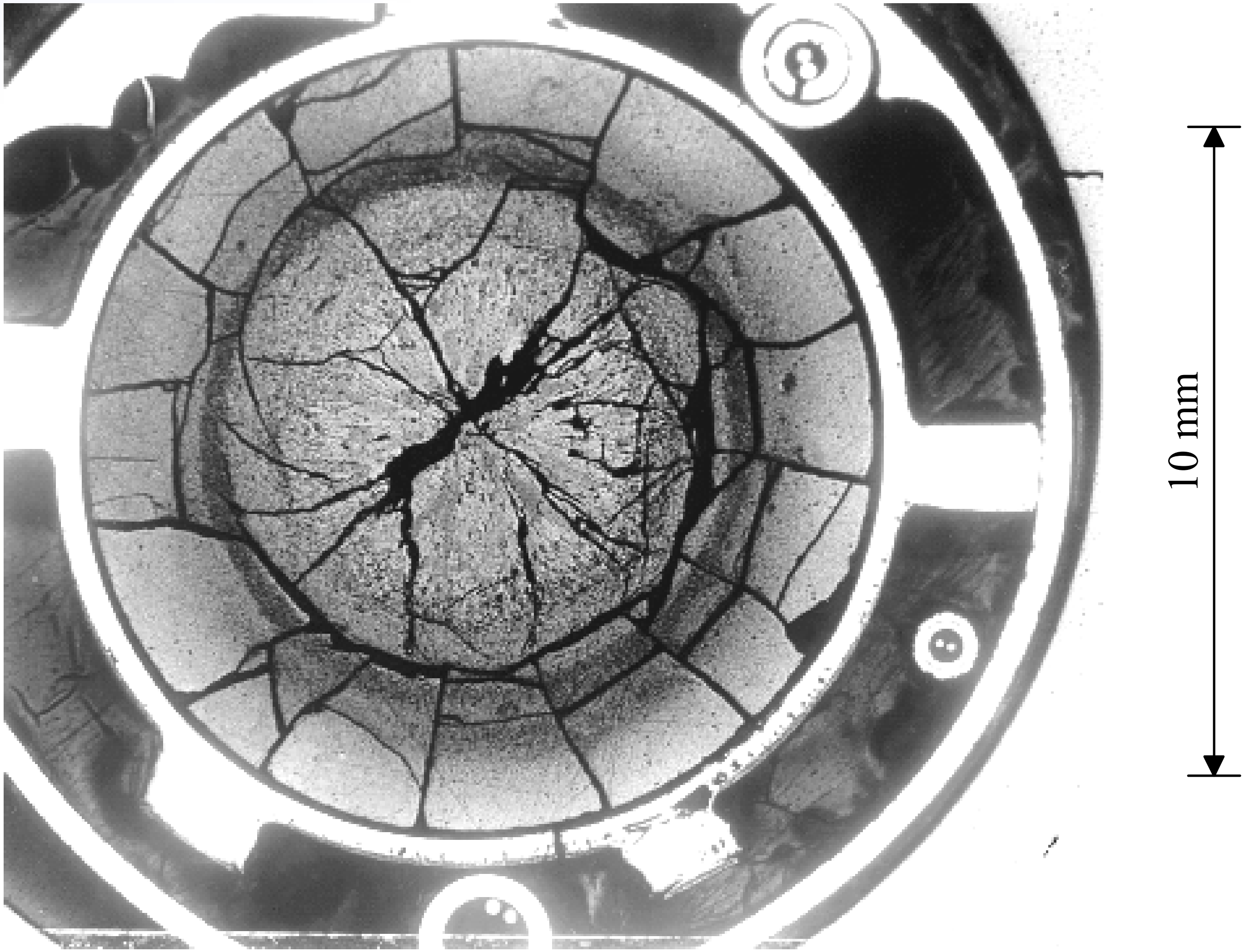


1997 Dec.,
1998 Jul. 10

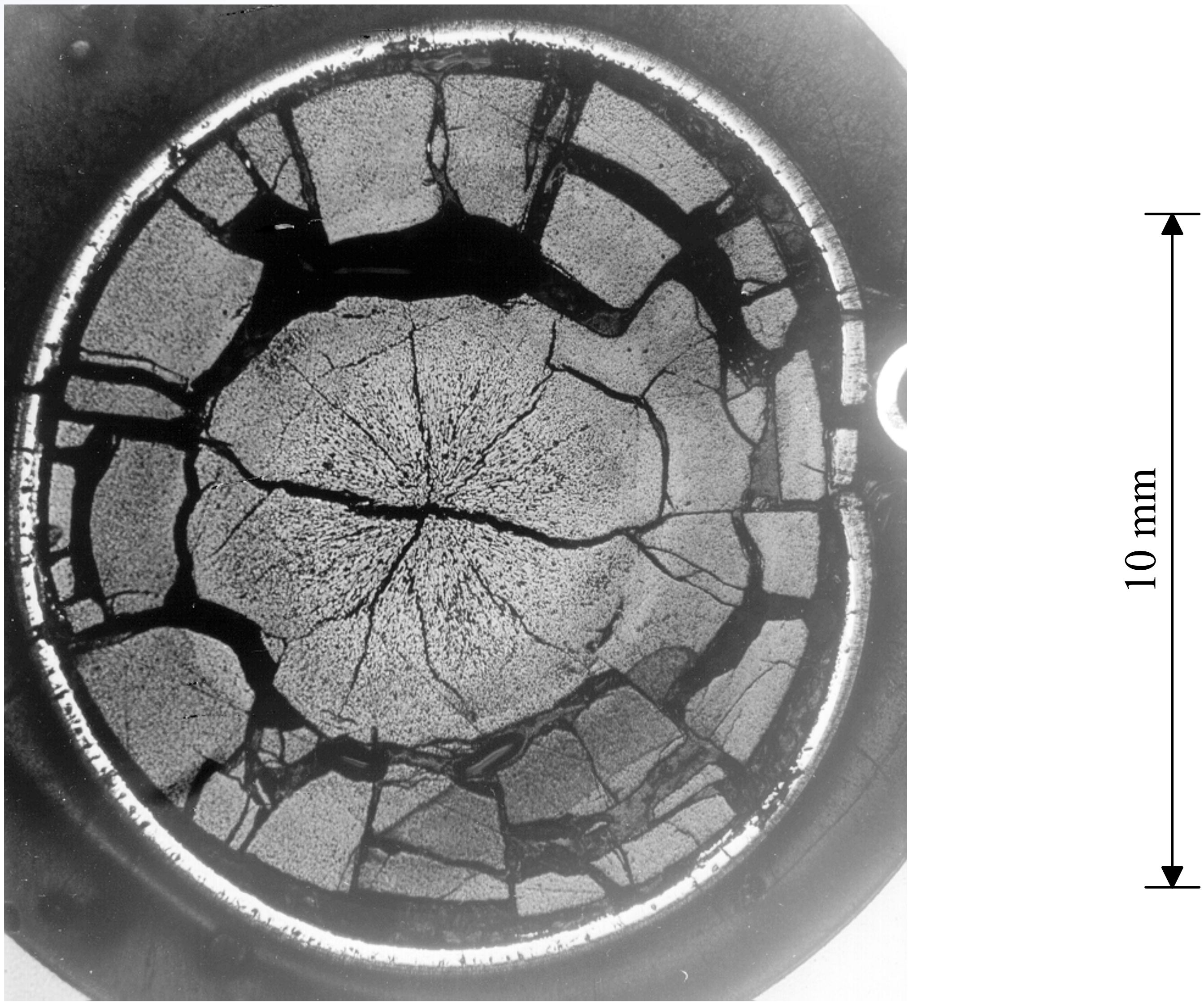
BTF-105B PIE, Elevation 373 mm



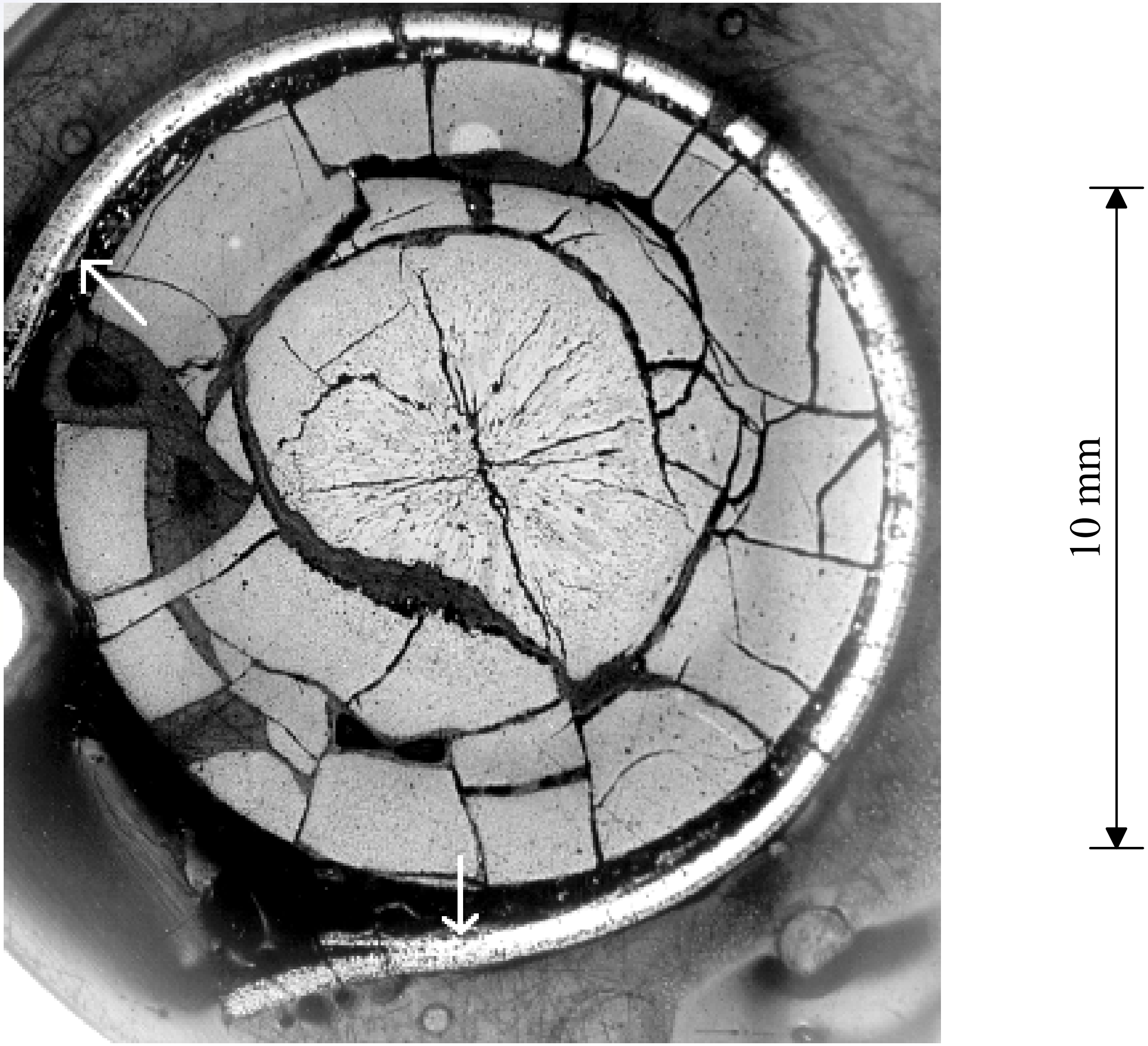
BTF-105B PIE, Elevation 247 mm



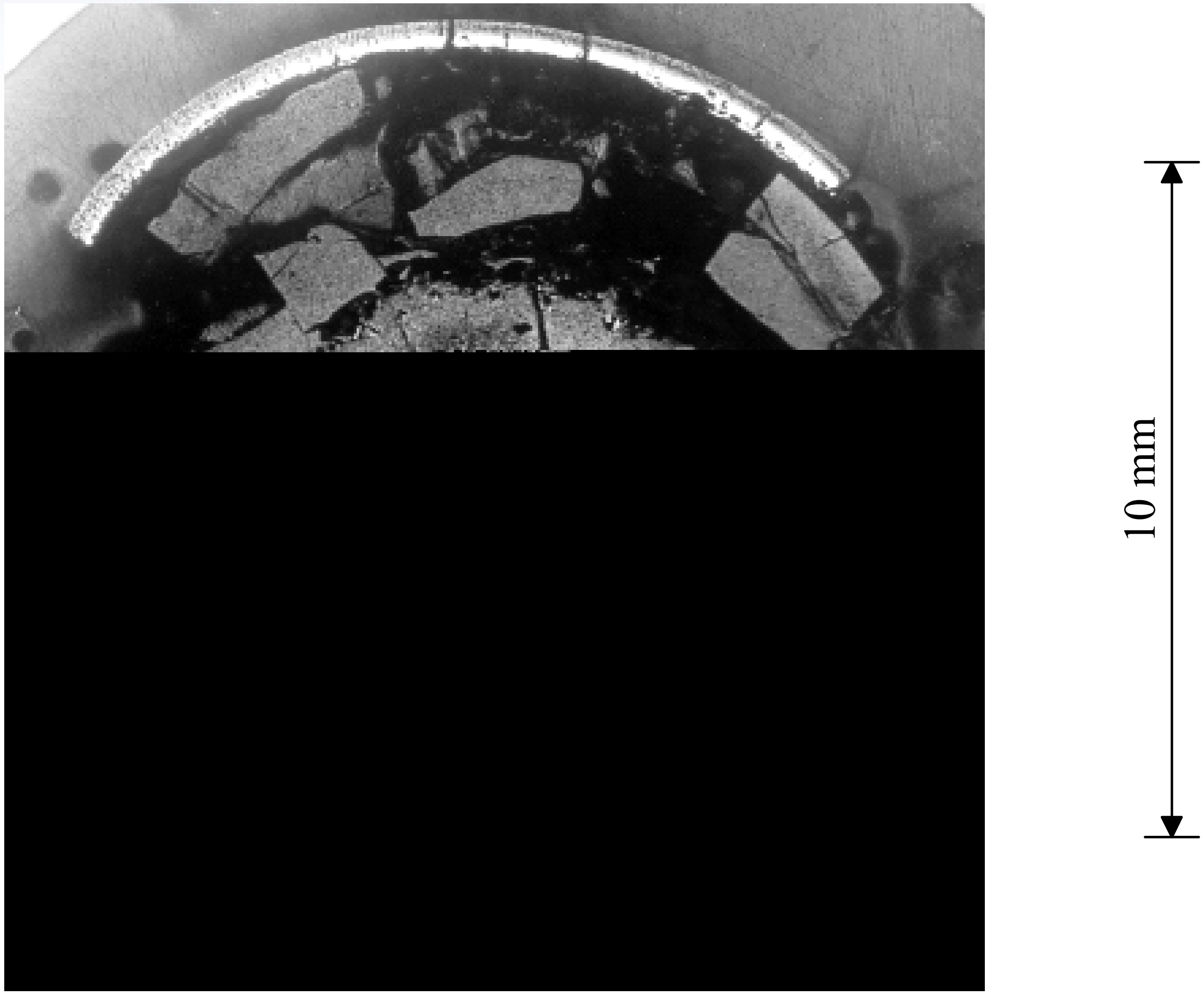
BTF-105B PIE, Elevation 105 mm



BTF-105B PIE, Elevation 69 mm



BTF-105B PIE, Elevation 20 mm



BTF Program Conclusions

- Data obtained for validation of CANDU fuel behavior codes under severe-fuel-damage accident conditions
- Post-test simulations performed using CANDU safety analysis computer codes (CATHENA, ELOCA, SOURCE and SOPHAEROS)
- No new phenomena or phenomena interactions identified

Outline

- ACR Fuel Design
- Fuel Behavior in Limited and Severe Core Damage Accidents
- Experimental Database
- **Computer Codes**

ELOCA-IST 2.1

- ELOCA-IST models the thermo-mechanical behavior of the fuel and fuel cladding under the transient conditions of an accident
- The model was first developed in the mid 1970s and has under gone continuous development since this time
- The model was chosen as part of the Industry Standard Tool Set (IST) in 1998
- The current version, ELOCA-IST 2.1, completed validation in 2001 and was released for use in 2002

Phenomena Modeled

- Expansion, contraction, and melting of the fuel
- Variations in the element internal gas pressure
- Deformation of the cladding
- Changes in the fuel/cladding heat transfer
- Zircaloy/coolant chemical reaction (oxidation)
- Cladding failure by over strain, oxidation, and beryllium-assisted cracking

Key Output Parameters

- **ELOCA-IST calculates:**
  - Fuel temperature
  - Cladding temperature
  - Internal gas pressure
  - Cladding strain
  - Axial and radial gaps (or contact pressure) between the fuel and the cladding
  - Time of cladding failure (if strain at failure is specified)
  - Oxide layer thickness on the outside of the cladding

Boundary Conditions

- The initial conditions required by ELOCA-IST are supplied by ELESTRES-IST and include:
  - the geometry and physical condition of the fuel and cladding
  - the initial radial profile of the heat generation rate of the fuel
- The time dependent conditions are:
  - the coolant temperature
  - the coolant pressure
  - the cladding-to-coolant heat transfer coefficient
  - the relative power (i.e. expressed as a fraction of the initial power)

Solution Method

- The ELOCA-IST thermal calculation allows for up to 100 radial annuli within the fuel pin and 20 axial segments
- The transient temperature distribution is calculated by an implicit finite-difference scheme
- The stress within the cladding is calculated from the strains imposed by the fuel in both the radial and axial directions
- The cladding stress model allows for anisotropic material properties, Zircaloy phase changes, and relaxation due to creep
- Cladding oxidation is calculated using the mechanistic finite element model FROM_SFD

Intended Uses of ELOCA-IST 2.1

- Large Break Loss of Coolant Accident
- Small Break Loss of Coolant Accident
- Secondary coolant failures
- Fuel handling accidents
- Loss of regulation accidents
- Auxiliary system failures
- Loss of Flow Accidents
- LOCA combined with failure of Emergency Core Coolant

Validation of ELOCA-IST

- The following 10 phenomena from the Fuel and Fuel Channel Validation Matrix have been identified as relevant to ELOCA-IST
  - Fission and Decay Heating
  - Heat Diffusivity in Fuel
  - Fuel-to- Cladding Heat Transfer
  - Fuel-to-End Cap Heat Transfer
  - Fission Gas Release to Gap and Internal Pressurization
  - Cladding Deformation
  - Cladding Failure
  - Fuel Cladding Deformation
  - Cladding Oxidation
  - Fuel and Cladding Melting

Validation Exercises

Validation exercises have been conducted against:

- **Cladding Oxidation Experiments**
- **Cladding Ballooning Experiments**
- **In-reactor Experiments on Fuel with CANDU type Geometry**
- **In-reactor Experiments on Fuel with Non-CANDU type Geometry**
- **Semi-Analytical Solution to the Radial Heat Distribution in the Fuel**

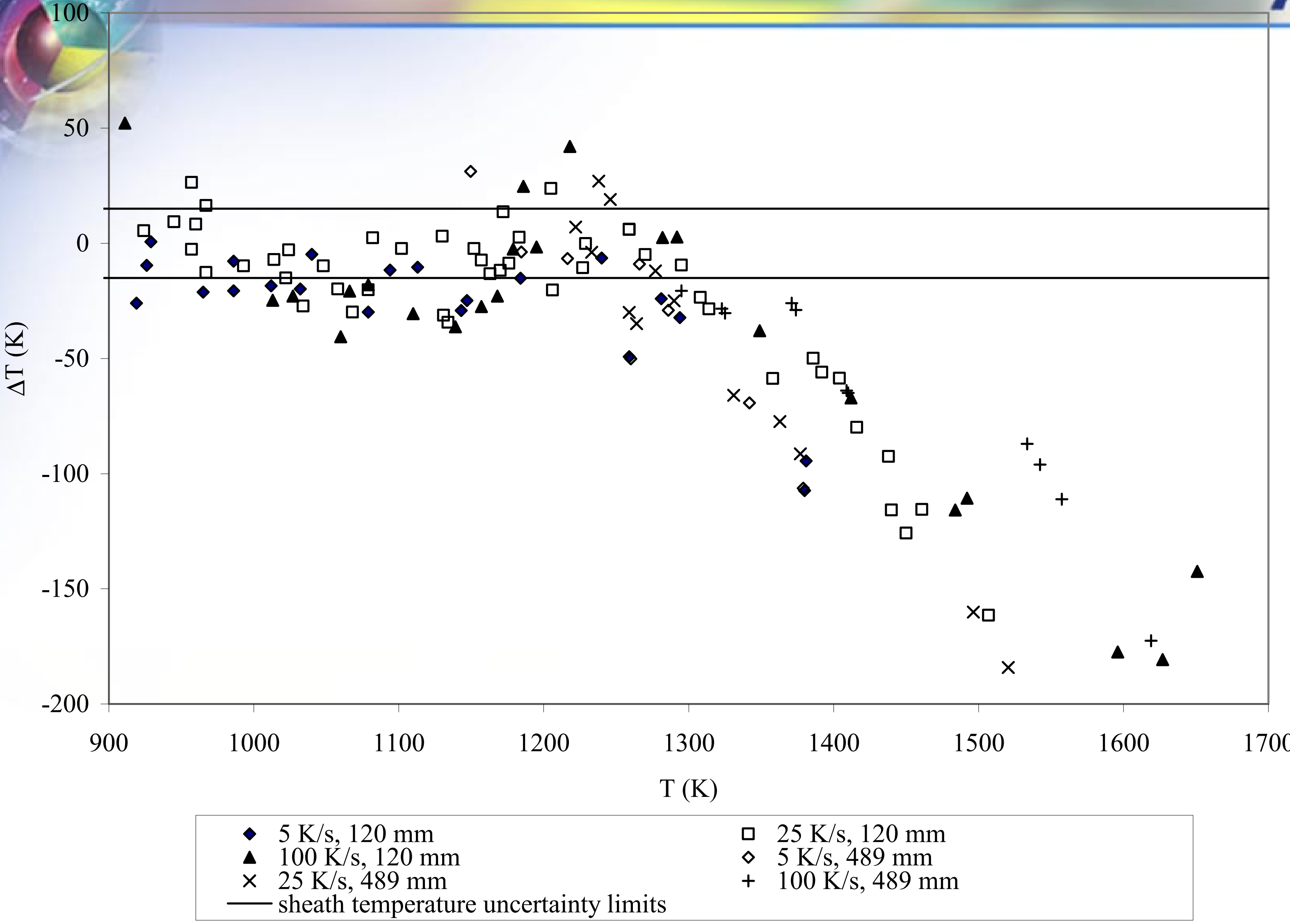
Cladding Oxidation Experiments

Validation of the FROM_SFD mechanistic oxidation model

- **Validation against ~500 individual tests**
- **In general the code performed well**
  - **Exhibited a positive bias of < 10% when calculating oxide layer thickness**
  - **Exhibited a positive bias of < 24% when calculating the thickness of the oxygen-stabilized alpha layer**

Cladding Ballooning Experiments

- Validation conducted against ~ 190 cladding ballooning tests
- Comparisons were made against predictions of failure temperature
- For non-oxidizing conditions: calculated failure temperature exhibited a bias of –21 K with an uncertainty of ±30 K
- For oxidizing conditions (steam): the predicted failure temperatures were up to 130 K lower than measured (at 1500 K)
- Use of the oxide strengthening model apparently improved the agreement for oxidizing conditions



Difference Between ELOCA Calculated and Observed Failure Temperatures for All Samples of As-Received Sheath Material in Steam as a Function of Observed Failure Temperature.

In-Reactor Experiments

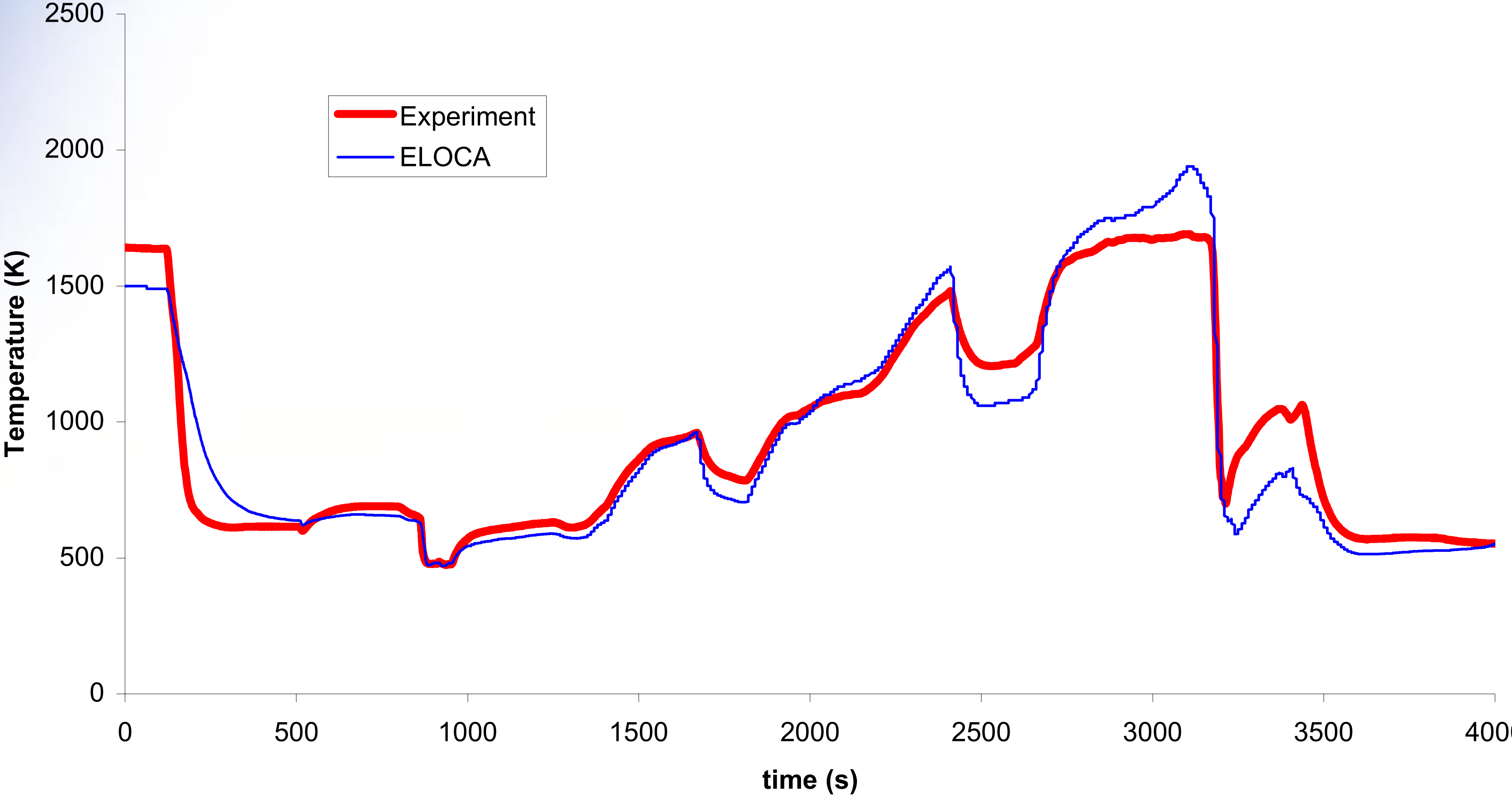
Validation against integrated in-reactor experiments included:

- Eight Experiments for CANDU-type fuel:
  - FIO-138: High-temperature transient in NRX
  - BTF-107 Loss of coolant test from full reactor power
  - BTF-104 Blowdown of Zircaloy-clad fuel in steam
  - FIO-142 BTF-105 Normal Operating Conditions Pre-Test
  - FIO-131 LOCA transient fresh Zircaloy-clad fuel
  - FIO-130 LOCA transient irradiated Zircaloy-clad fuel
  - CANDU-PBF test at the Power Burst Facility (PBF), Idaho National Engineering Labs (INEL)
  - BTF-105A Blowdown test on Zircaloy-clad fresh fuel

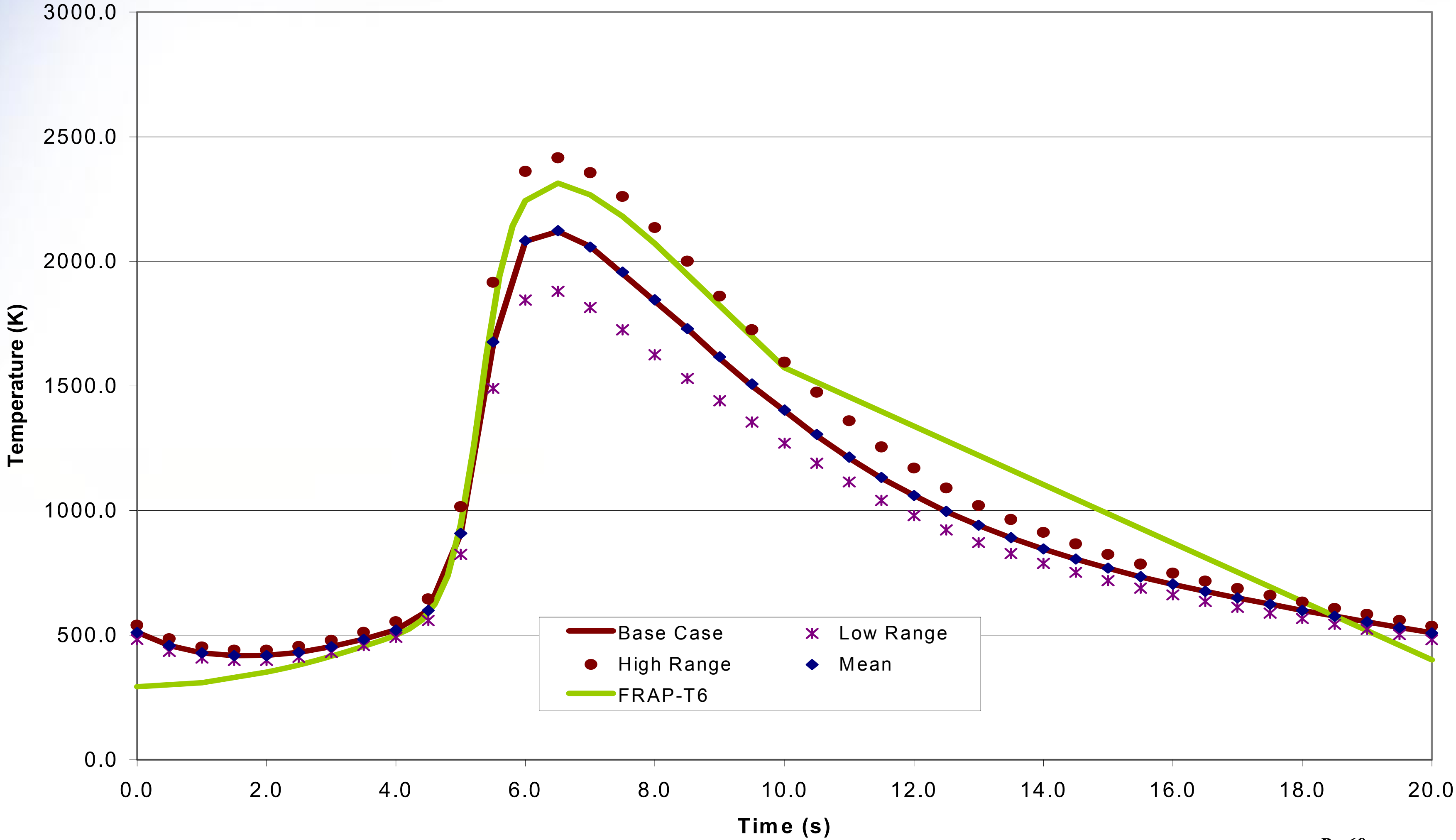
In-reactor Experiments (cont.)

- **Five experiments on non-CANDU type fuel:**
  - **SFD test 1-1, conducted at PBF, INEL**
  - **SFD test 1-4, conducted at PBF, INEL**
  - **SFD-ST Severe fuel damage scoping test conducted at PBF, INEL, and**
  - **PHEBUS FP FPT0 and FPT1 Tests at IPSN, France**
  - **Russian IGR fuel power-pulse tests**

BTF-105A Fuel Centerline Temperature



ELOCA Validation – Russian IGR Test H16T Fuel Centerline Temperature



In-reactor Tests Conclusions

- Comparison against fuel temperatures
  - The ELOCA code performed within the estimated uncertainty of the experiments
- Comparison against internal gas pressure
  - Within experimental uncertainty, only a small number of measurements
- Comparison against cladding strain
  - Within experimental uncertainty, only one in-reactor experiment

Analytical Solution

- ELOCA-IST 2.1 was compared against an analytical solution for the transient radial heat distribution in a composite cylinder (i.e., fuel and cladding)
- There was a close match (<0.5 K fuel centerline temperature difference) between the ELOCA-IST calculation and the analytical solution

Summary

- Good technology base for understanding of CANDU fuel behavior in accidents
  - Phenomena
  - Experimental database
  - Computer codes
- Extension to ACR is straightforward

AECL

TECHNOLOGIES INC.