1. Review of ACR Fuel Report

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AECL

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Chalk River Laboratories
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AECL TECHNOLOGIES INC.

Outline

- Overview of ACR
- Description of CANDU fuel
- Description of ACR fuel
- Technology base for ACR fuel
  - CANFLEX
  - Extended burnup fuel
  - Low Void Reactivity Fuel (LVRF)
- Overview of computer programs

(Fuel design criteria and qualification covered Wed & Thurs)

ACR

- ACR is evolutionary extension of the CANDU 6 plant, which has ten units in operation on four continents, and one unit currently under construction

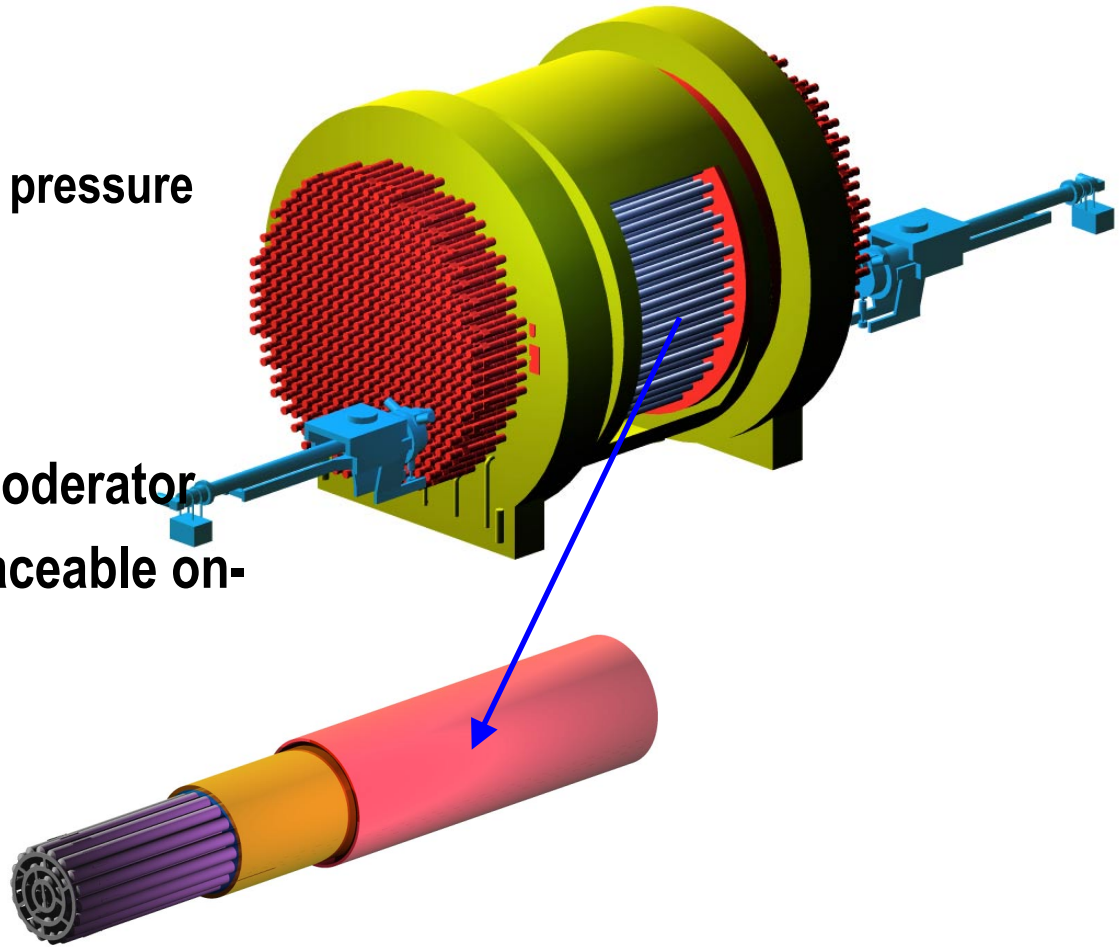
- provides a sound basis for projecting highly reliable performance, low project risk, assured costs, licensability
- fuel based on 3 underlying technologies: CANFLEX, extended burnup, and LVRF

ACR Optimizes the Channel Concept

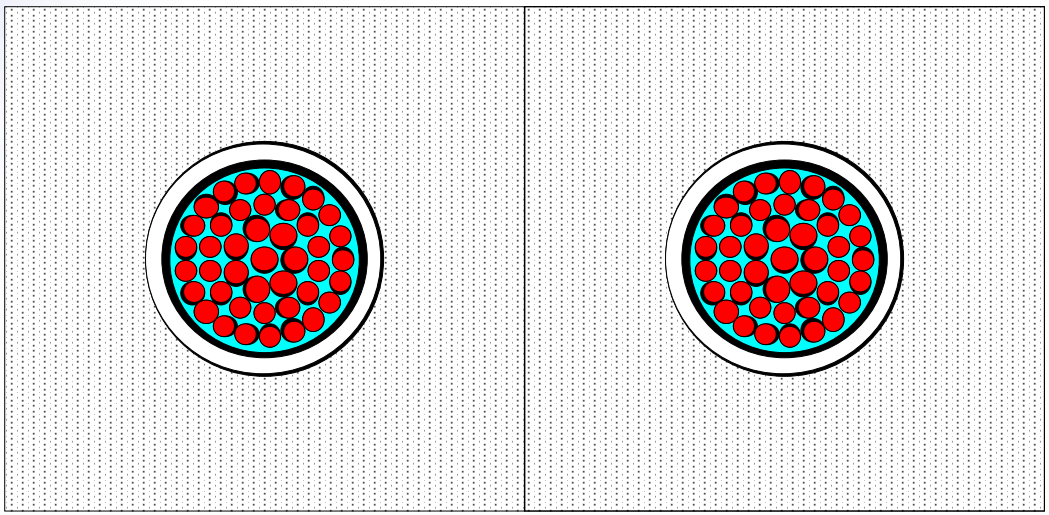
- Current operating CANDU reactors
  - natural uranium (NU) fuel
  - heavy water (D_2O) coolant
  - heavy water (D_2O) moderator
- ACR – relaxes constraint of natural uranium fuel and –
  - uses Slightly Enriched Uranium (SEU) fuel
  - uses light water coolant
  - reduces core size and reduce amount of heavy water moderator
    - increase reactor coolant system (RCS) pressure
    - increase thermal efficiency
- Retains intrinsic proven CANDU features

Intrinsic CANDU Features

- Channel reactor
  - horizontal channels
  - pressure tube as core pressure boundary
  - water cooled
  - water moderated
- Separate coolant and moderator
- Short fuel bundles replaceable on-line

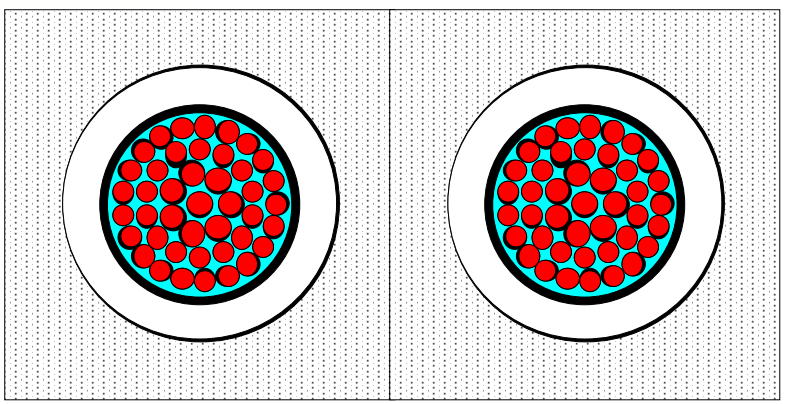


Comparison of CANDU and ACR Lattices



CANDU 6

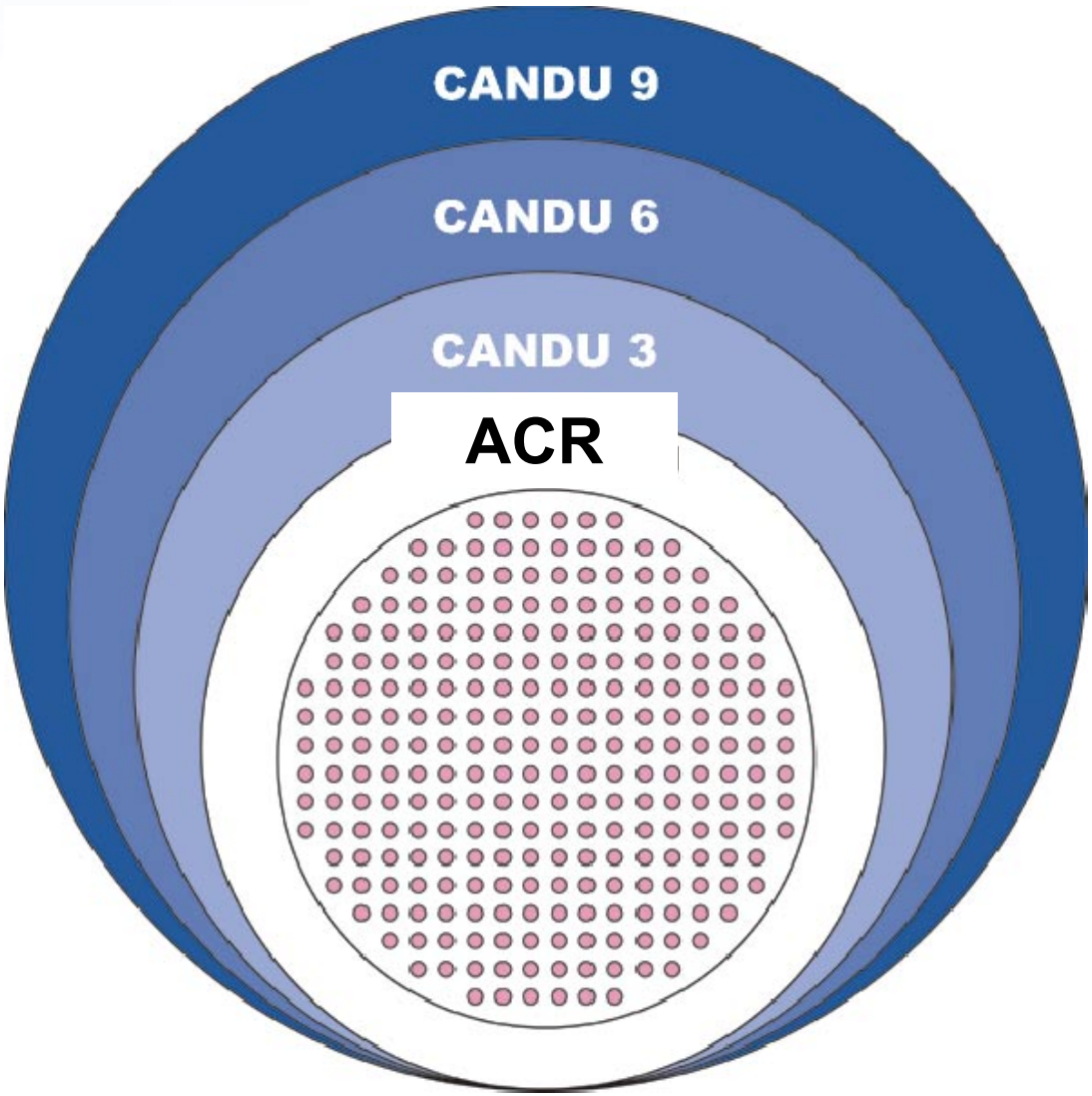
Lattice Pitch = 28.58 cm

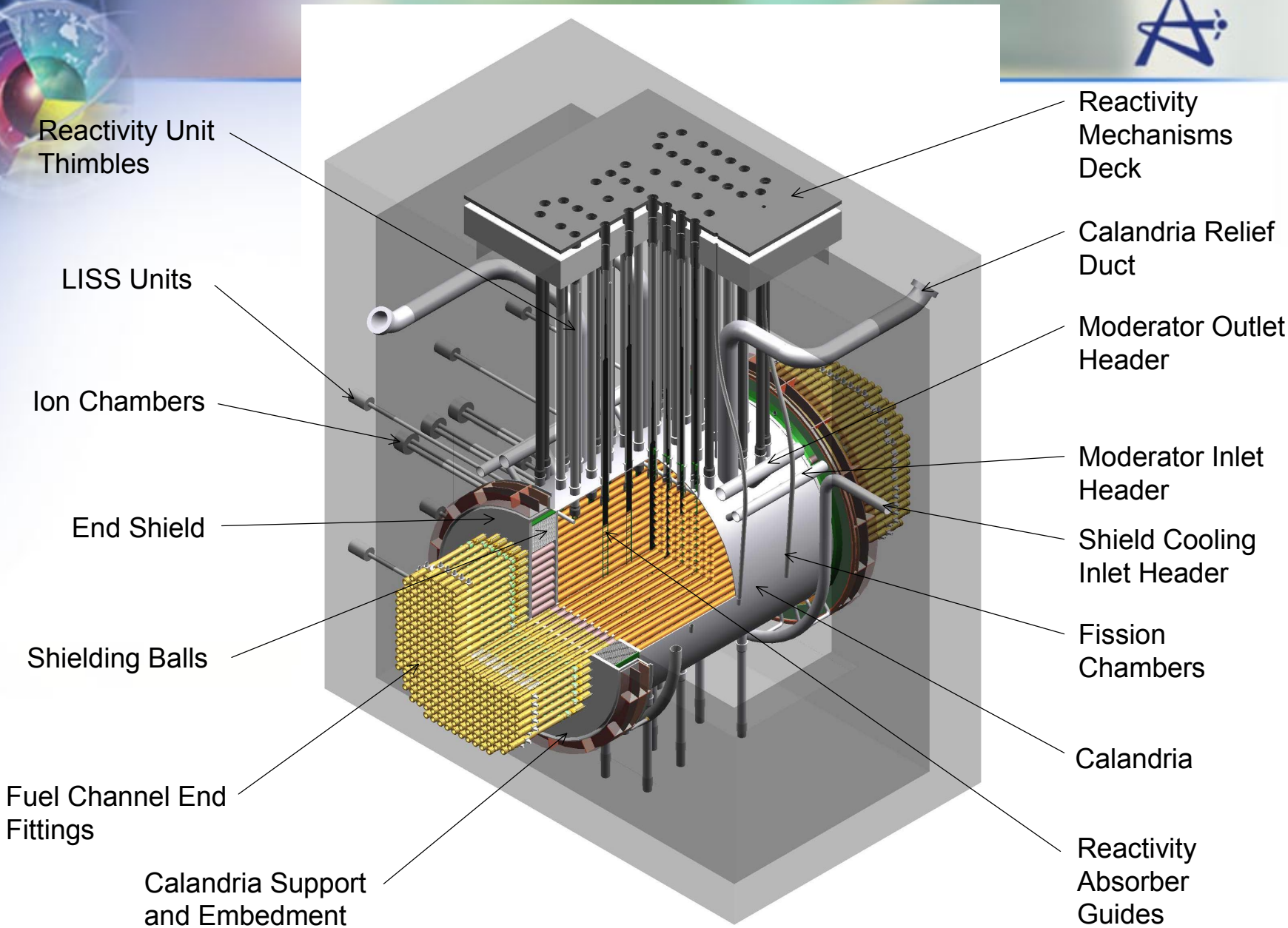


ACR

Lattice Pitch = 22.0 cm

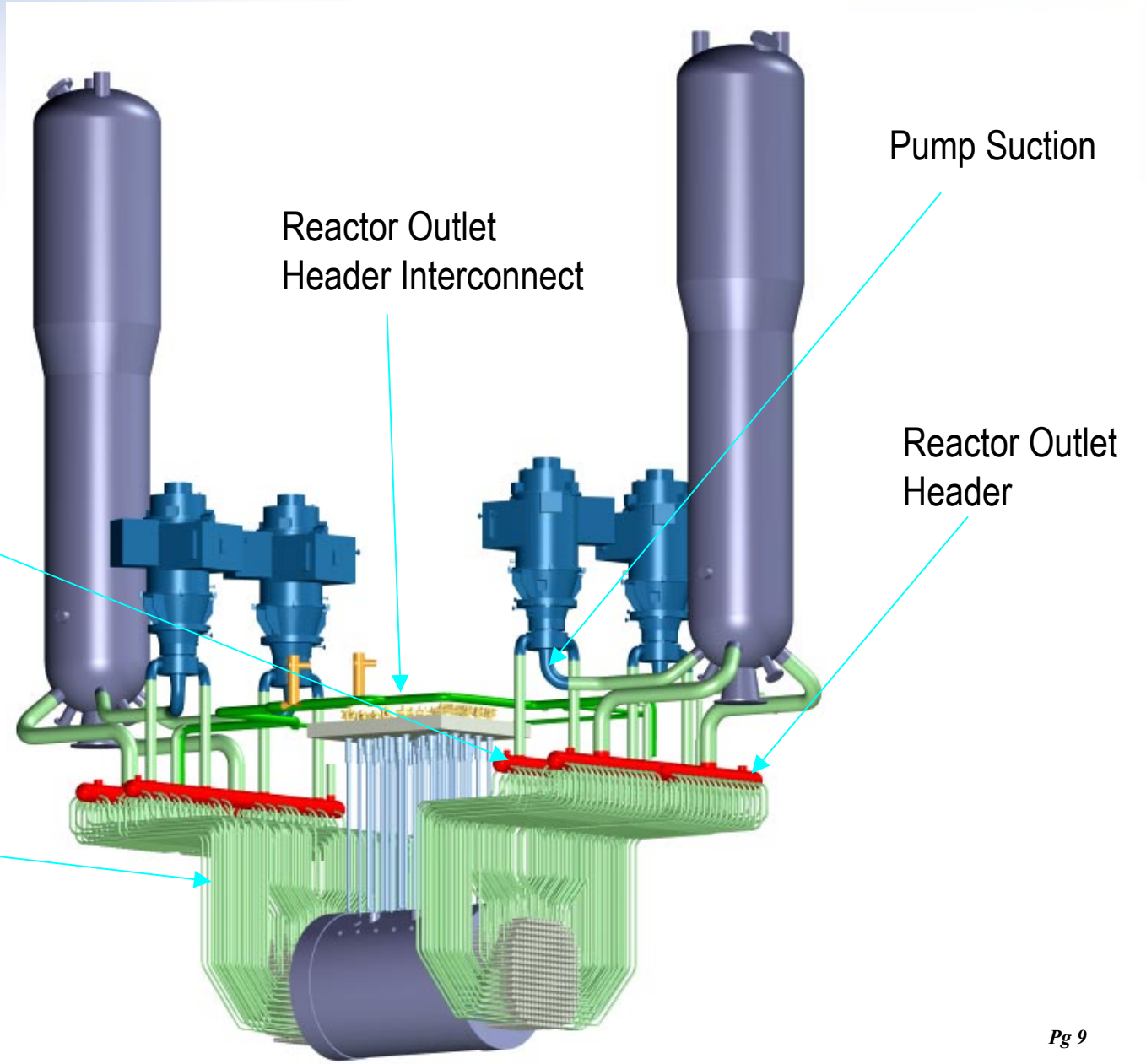
Smaller Calandria – Less Heavy Water





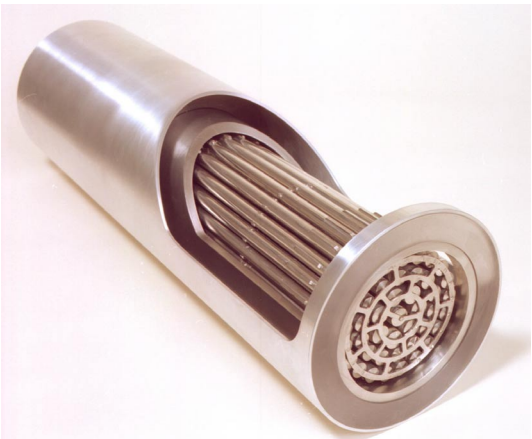
ACR –700
Reactor Assembly

Reactor Cooling System



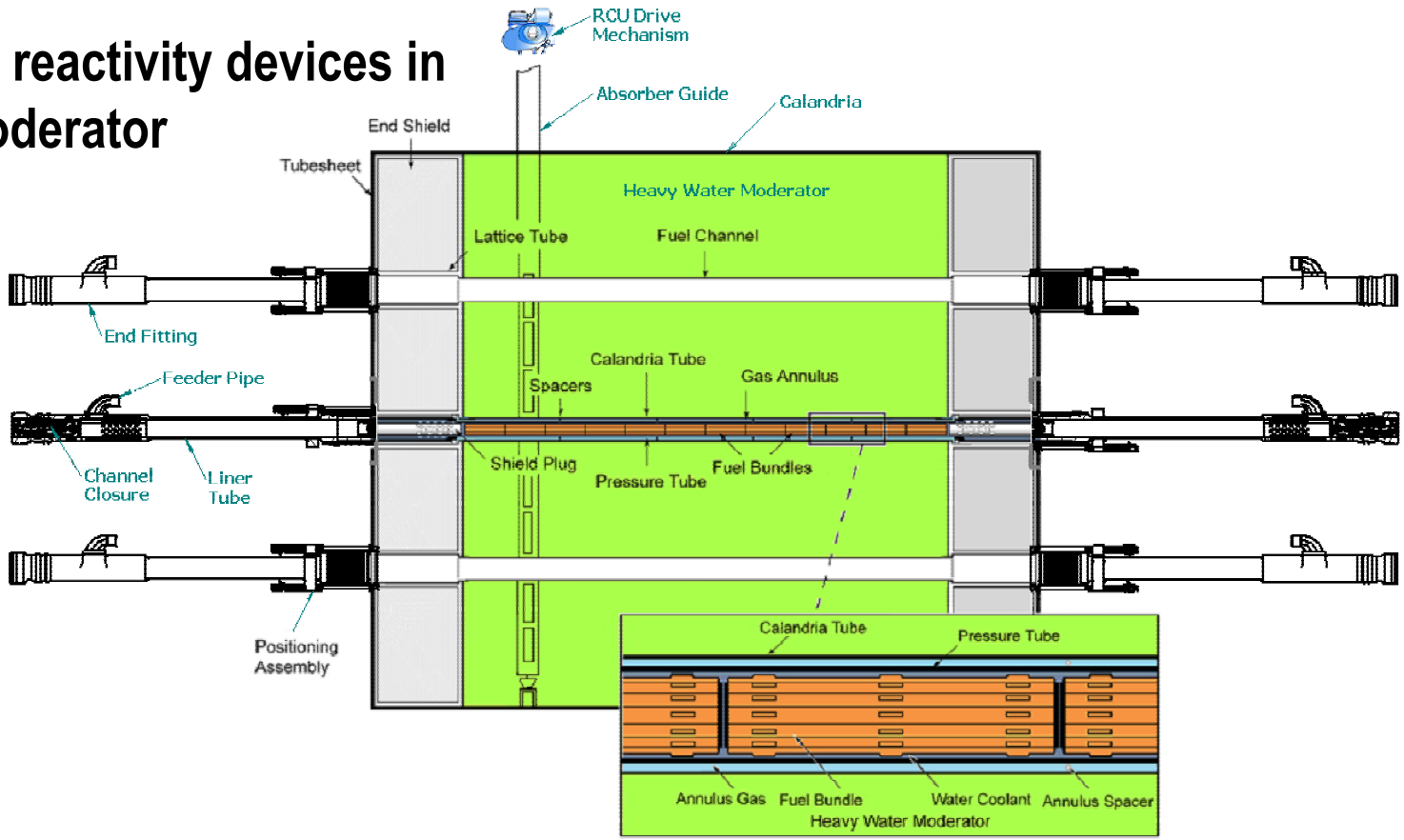
ACR Fuel Channel Design

- **Pressure tube**
  - thicker to reduce stresses during normal operation
  - chemistry specification optimized to reduce deformation and corrosion
- **Calandria tube**
  - Zr-4 has materials properties equivalent to Zr-2
  - thicker tube to withstand spontaneous pressure tube failure

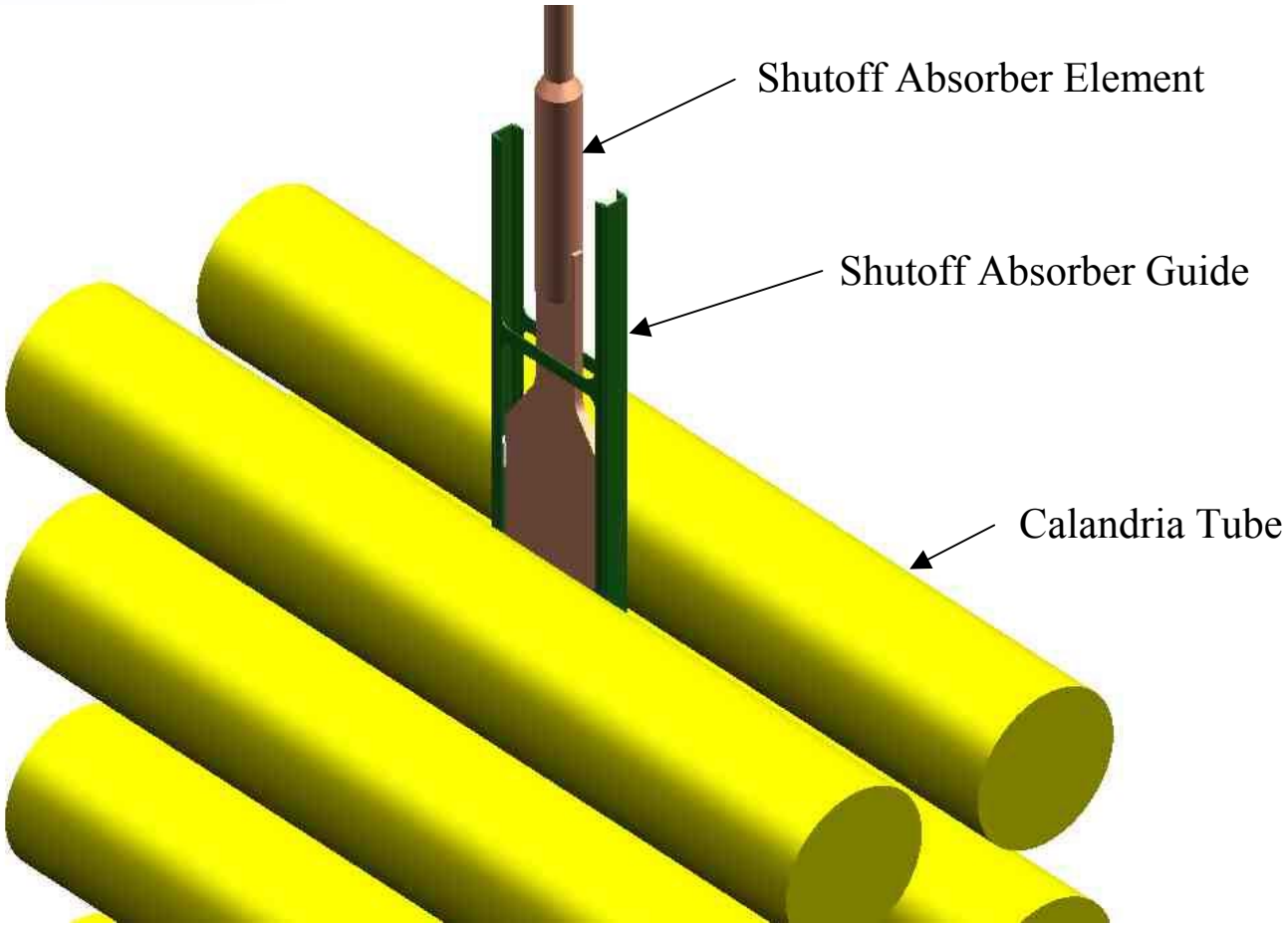


ACR Reactor

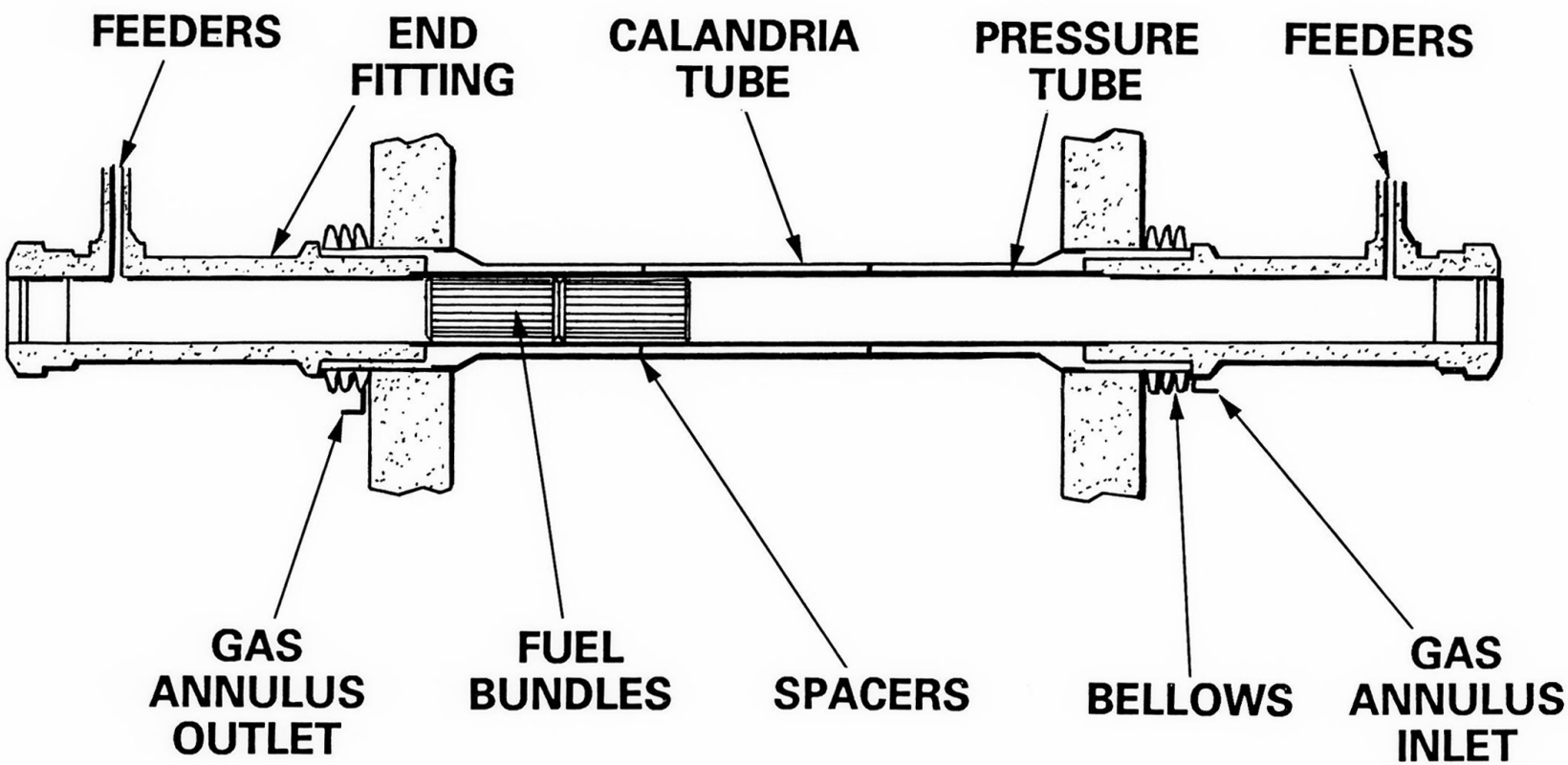
- Horizontal fuel channels surrounded by low temperature, low pressure moderator
- Steel calandria contains moderator and supports fuel channels
- All reactivity devices in moderator



Schematic of ACR-700 Shutoff Rod

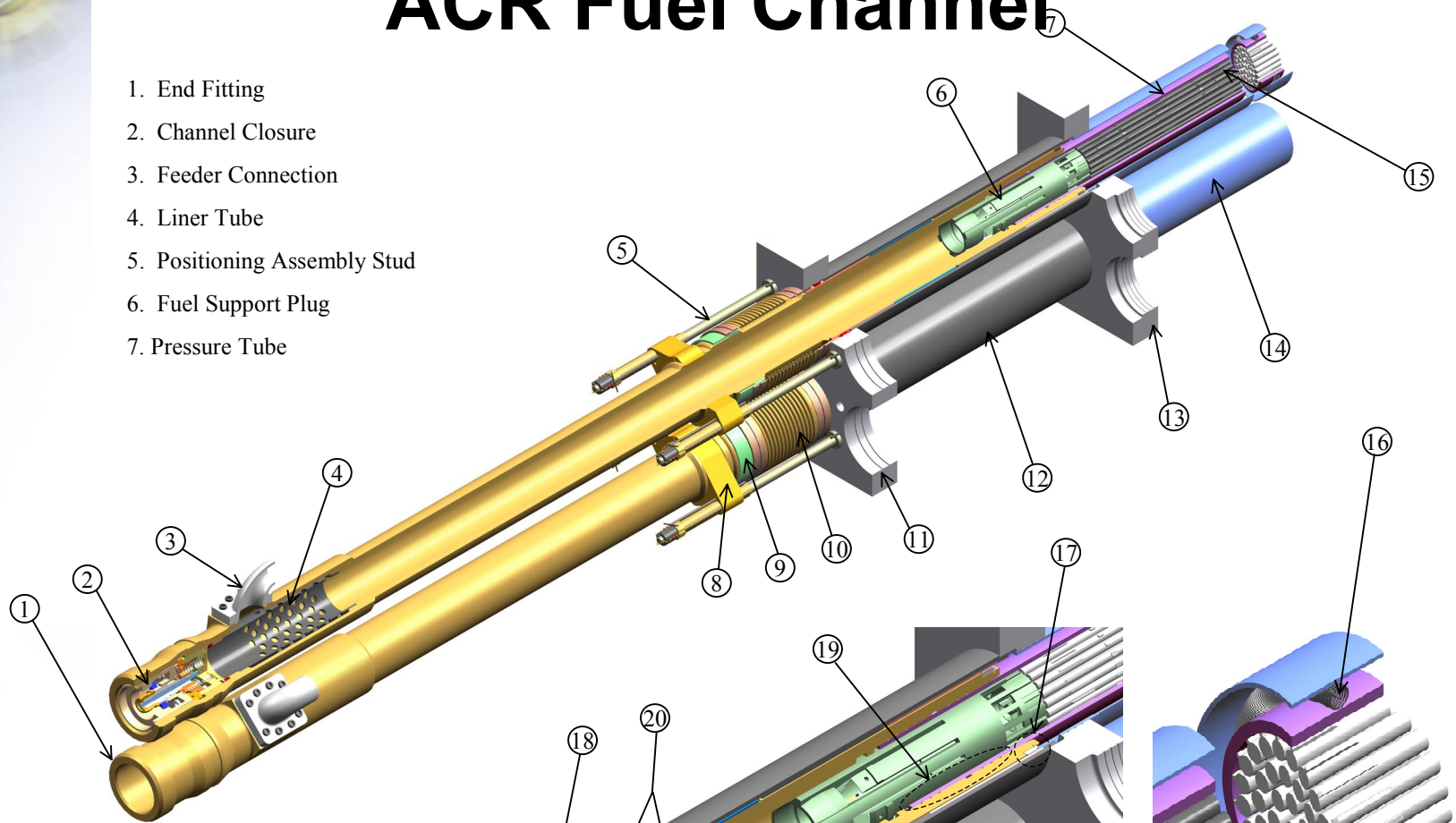


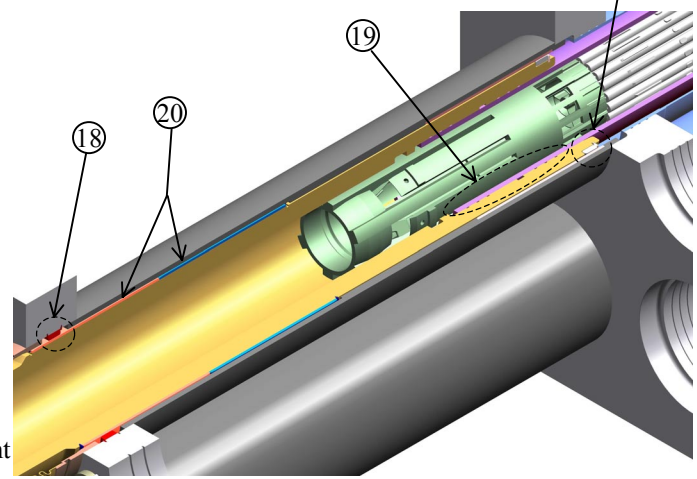
FUEL CHANNEL COMPONENTS



ACR Fuel Channel

1. End Fitting
2. Channel Closure
3. Feeder Connection
4. Liner Tube
5. Positioning Assembly Stud
6. Fuel Support Plug
7. Pressure Tube

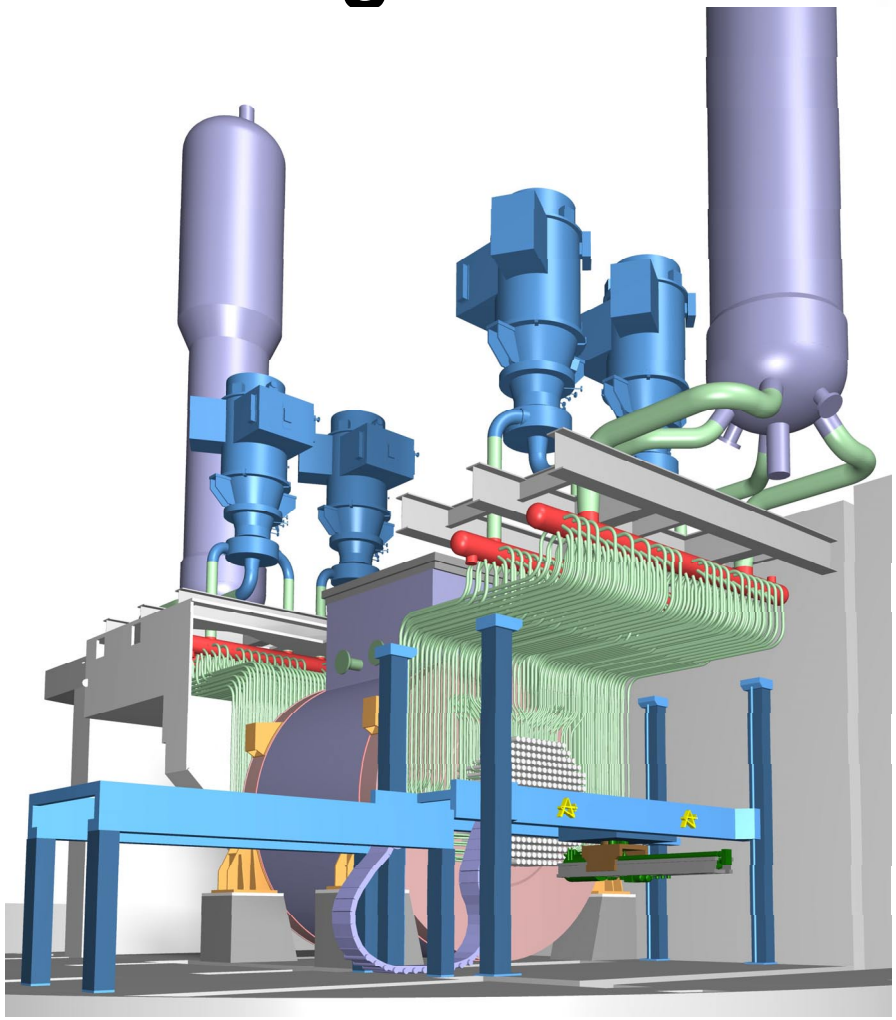




8. Position Assembly Yoke
9. Bellows Attachment Ring
10. Bellows Assembly
11. Fuelling Tubesheet
12. Lattice Tube
13. Calandria Tubesheet
14. Calandria Tube
15. Fuel Bundle
16. Channel Annulus Spacer
17. Inboard Journal Bearing
18. Outboard Journal Bearing
19. Pressure Tube Rolled Joint
20. Shielding Sleeves

On-Power Fueling

- Each fuel channel contains 12 fuel bundles
- 2 bundles of irradiated fuel are removed and 2 bundles of fresh fuel are inserted using two fueling machines connected to each end of a channel
- The fueling machine has a movable Class 1 pressure vessel that connects to the new and irradiated fuel ports and fuel channels in sequence to move the fuel



Description of CANDU Fuel

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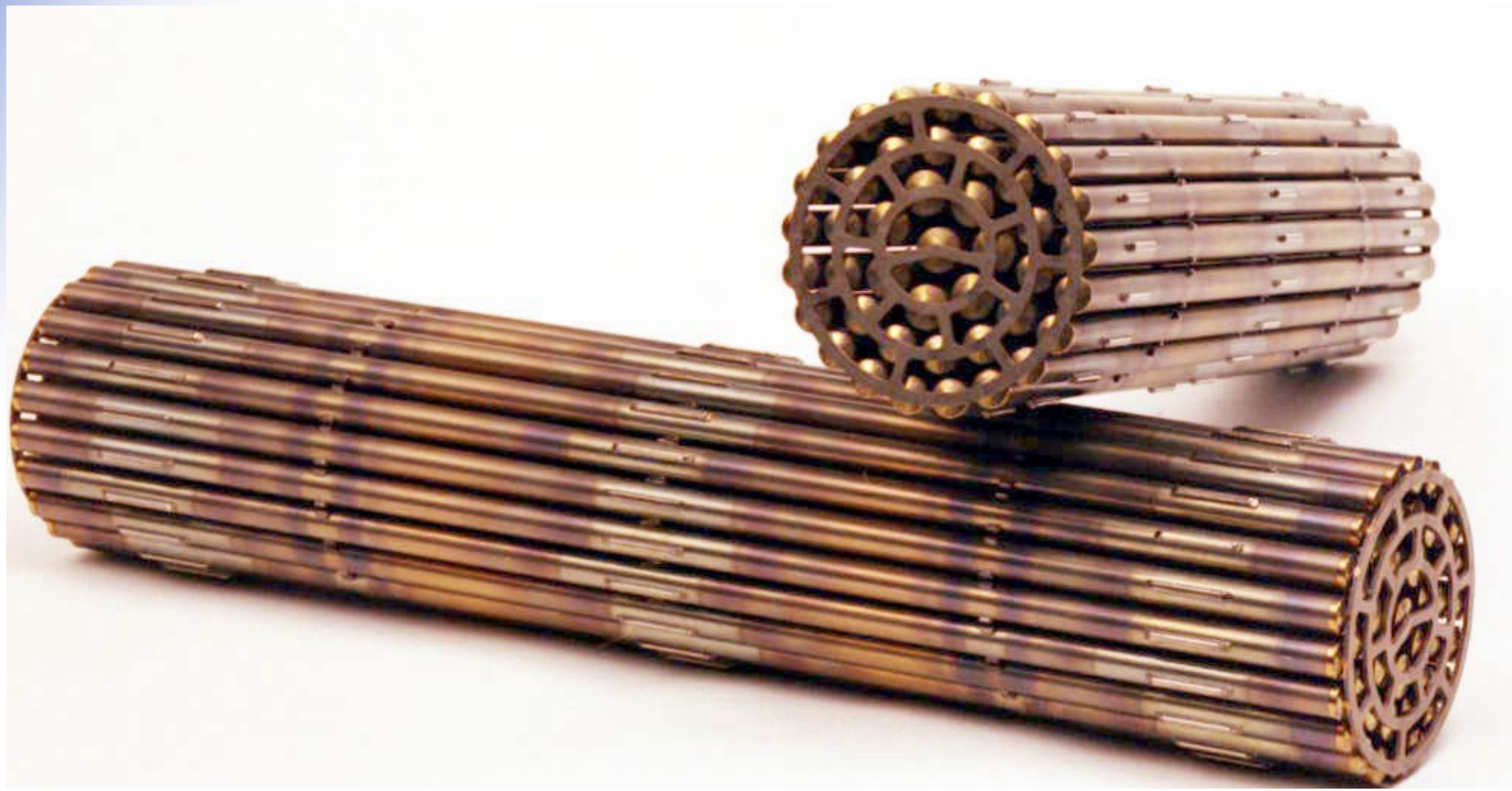
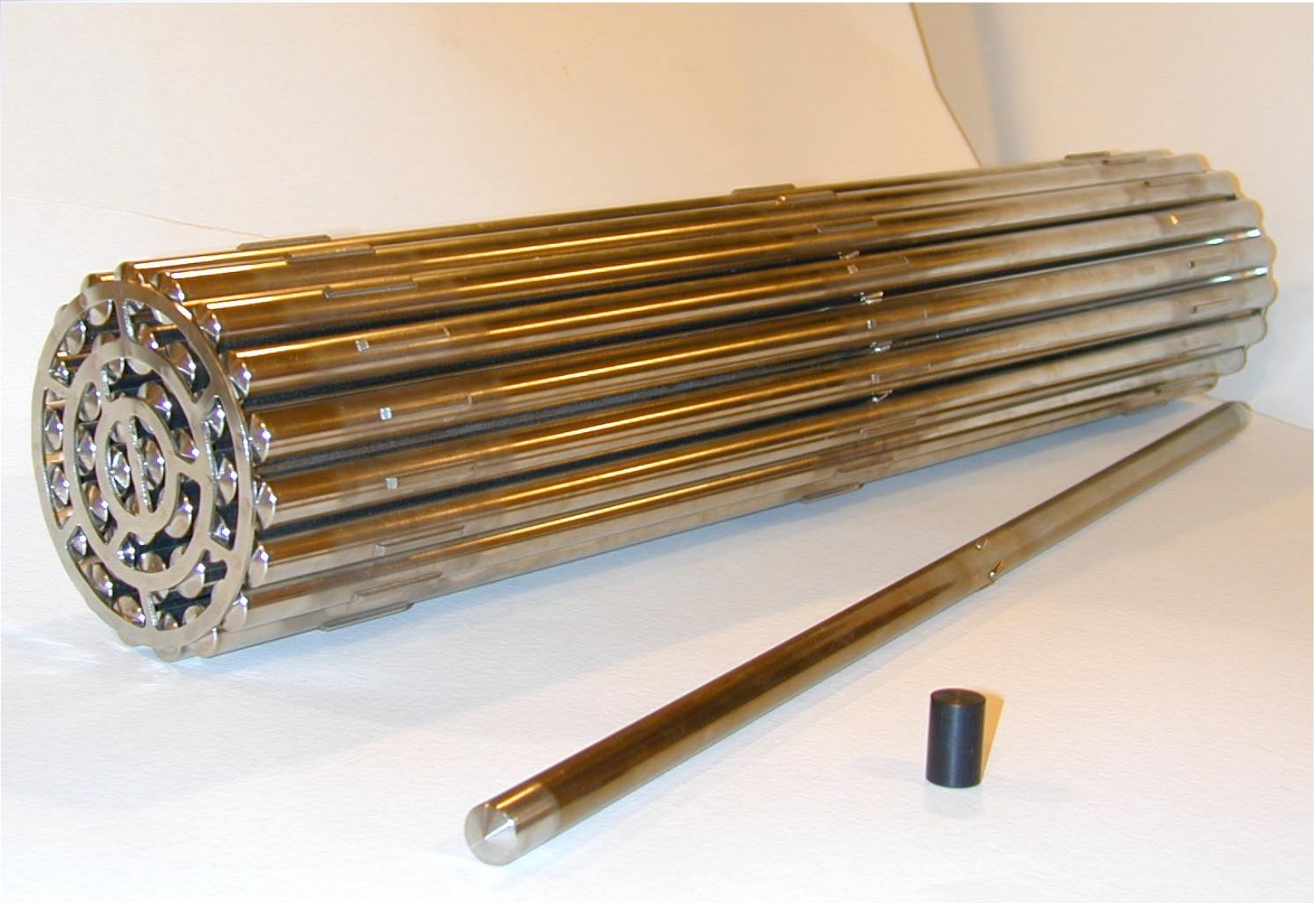
Features of CANDU Fuel

- Small
  - 50 cm (20”) length, 10 cm (4”) dia
- Lightweight
  - ~24 kg (50 lb) / bundle
- Simple in design
  - 7 separate components (8 in CANFLEX)
- Easy to manufacture
  - all countries having CANDU reactors manufacture their fuel
- Excellent performance
  - defect rate ~ 2 defects per million elements
  - ~2 million bundles irradiated

CANFLEX Bundle



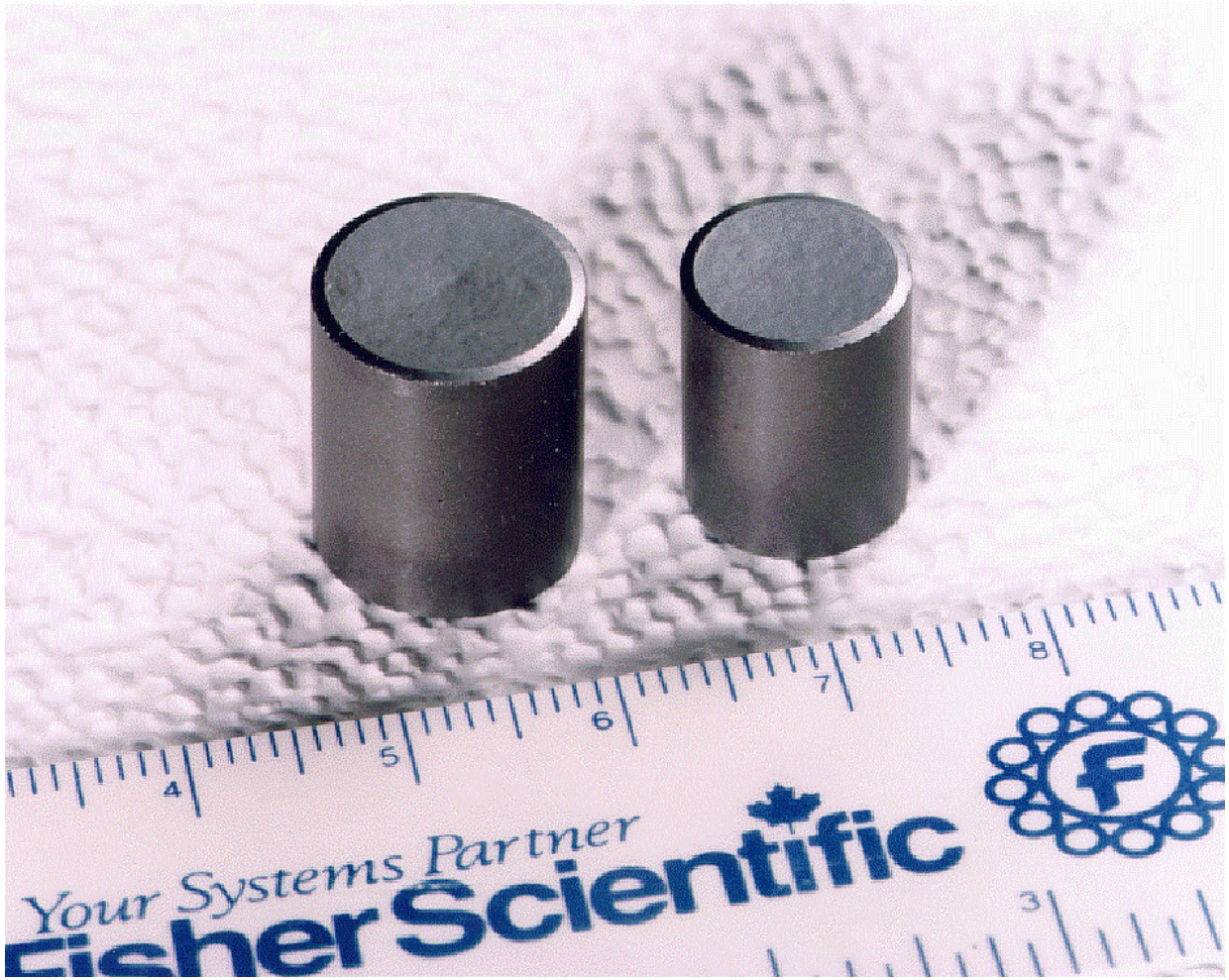
CANDU 6 37-element Fuel



UO_2 Powder

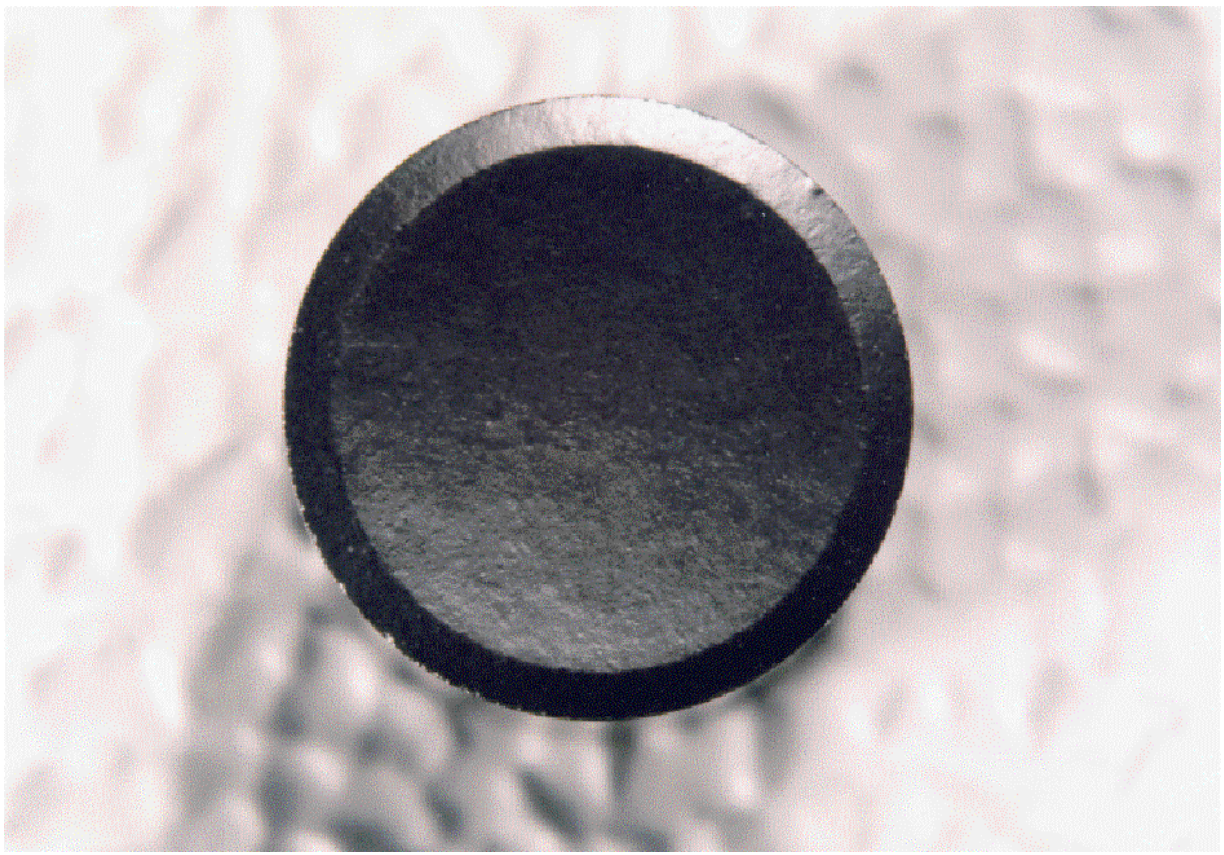
UO_2 Pellets

- UO_2, high density (for dimensional stability)



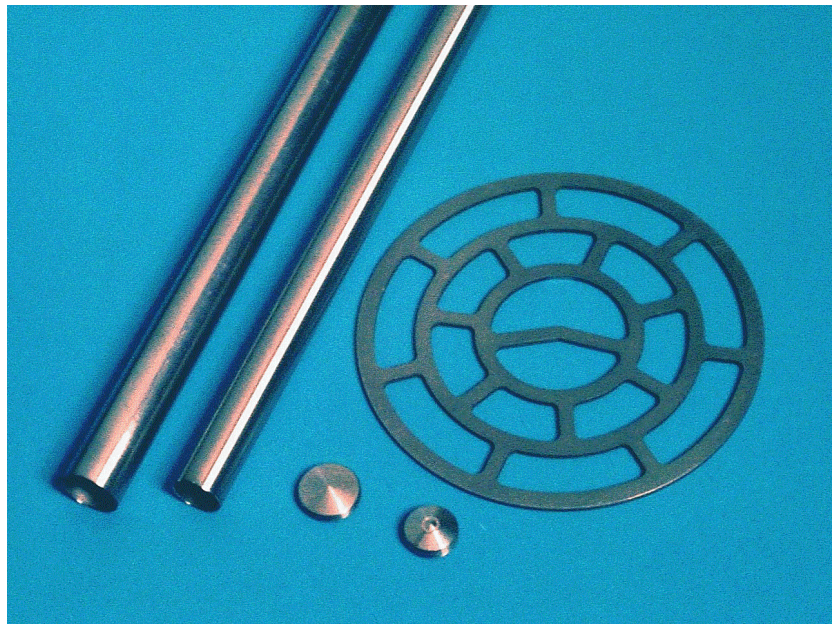
UO_2 Pellets

- Chamfers and end-dishes (reduce inter-pellet stresses on clad, volume for fission gas)



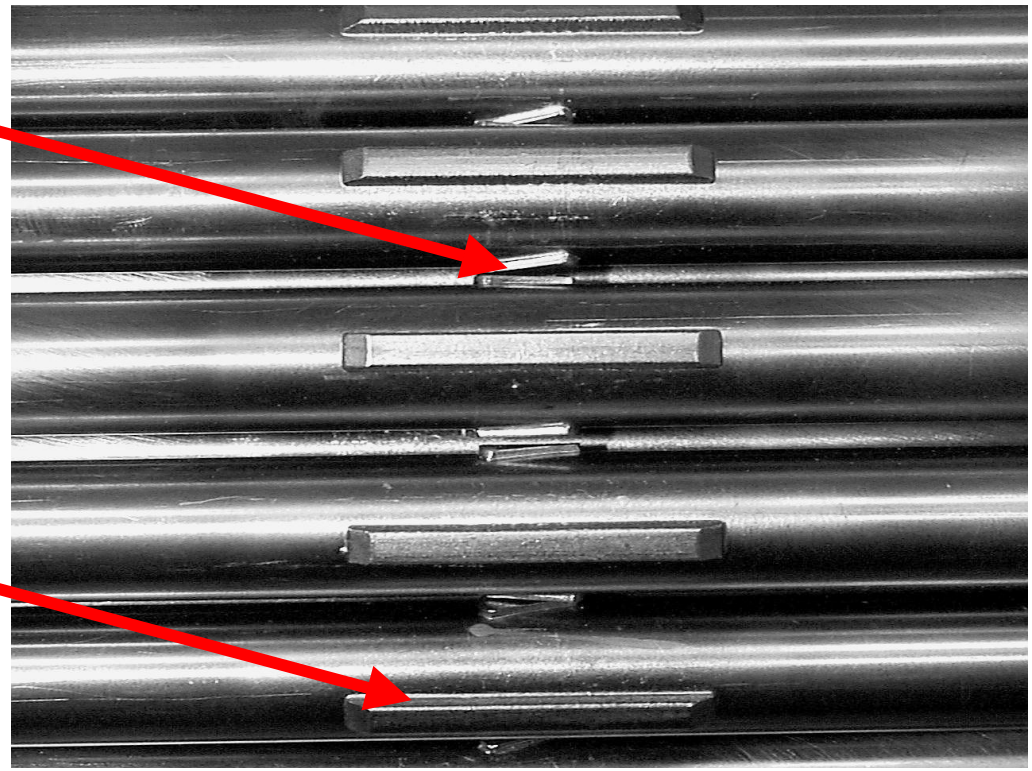
Clad, CANLUB, Endcaps, Endplates

- Clad
  - thin, collapsible (0.4 mm, 0.016”)
  - excellent heat transfer to coolant
  - low neutron absorption, Zr-4
- CANLUB
  - graphite coating applied to inside of clad provides protection against power ramp failures
- Endcaps
  - seal the fuel element
  - thin to reduce neutron absorption, good heat transfer
  - profiled to interact with fuel channel and fuel handling components
- Endplates
  - thin to minimize neutron absorption
  - flexible to accommodate fuel element differential expansion
  - strong to provide structural support and element separation



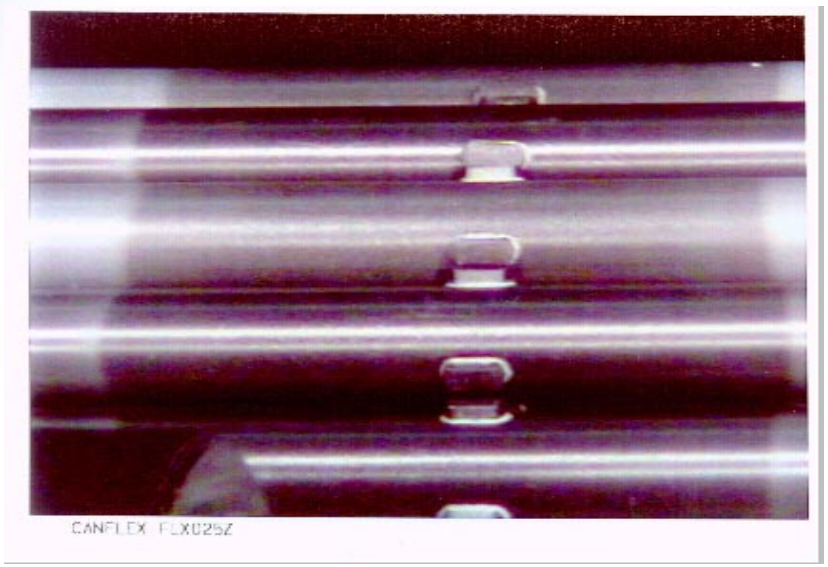
Spacers, Bearing Pads

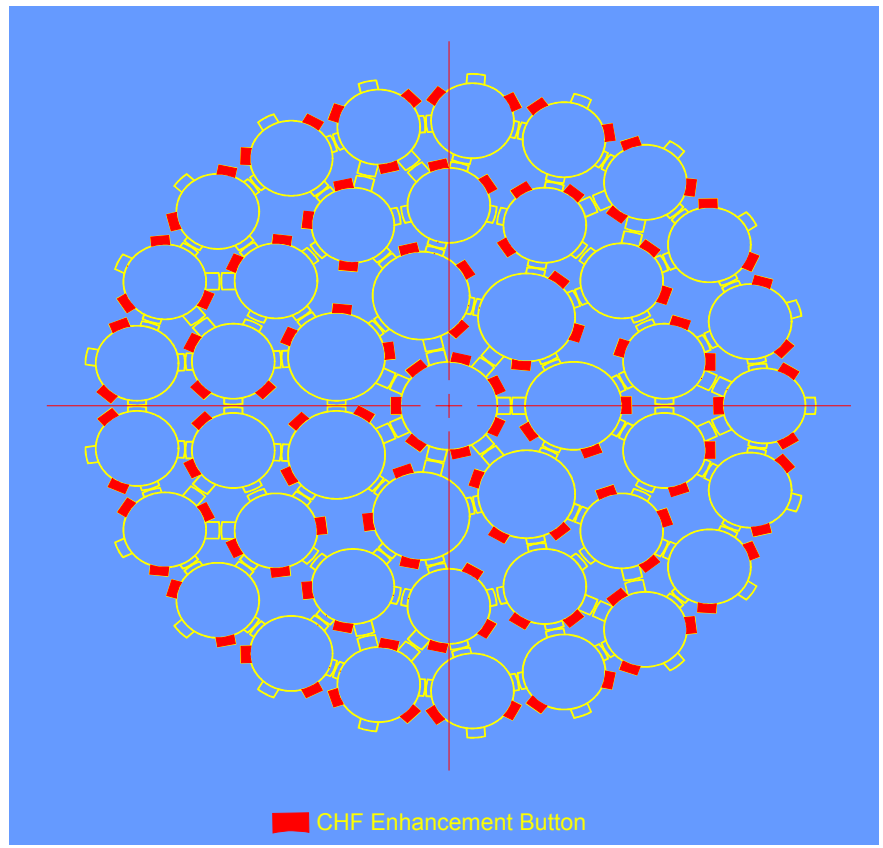
- Inter-element spacers
  - provide element separation at the bundle midplane
- Bearing pads
  - provide element-to-pressure tube separation



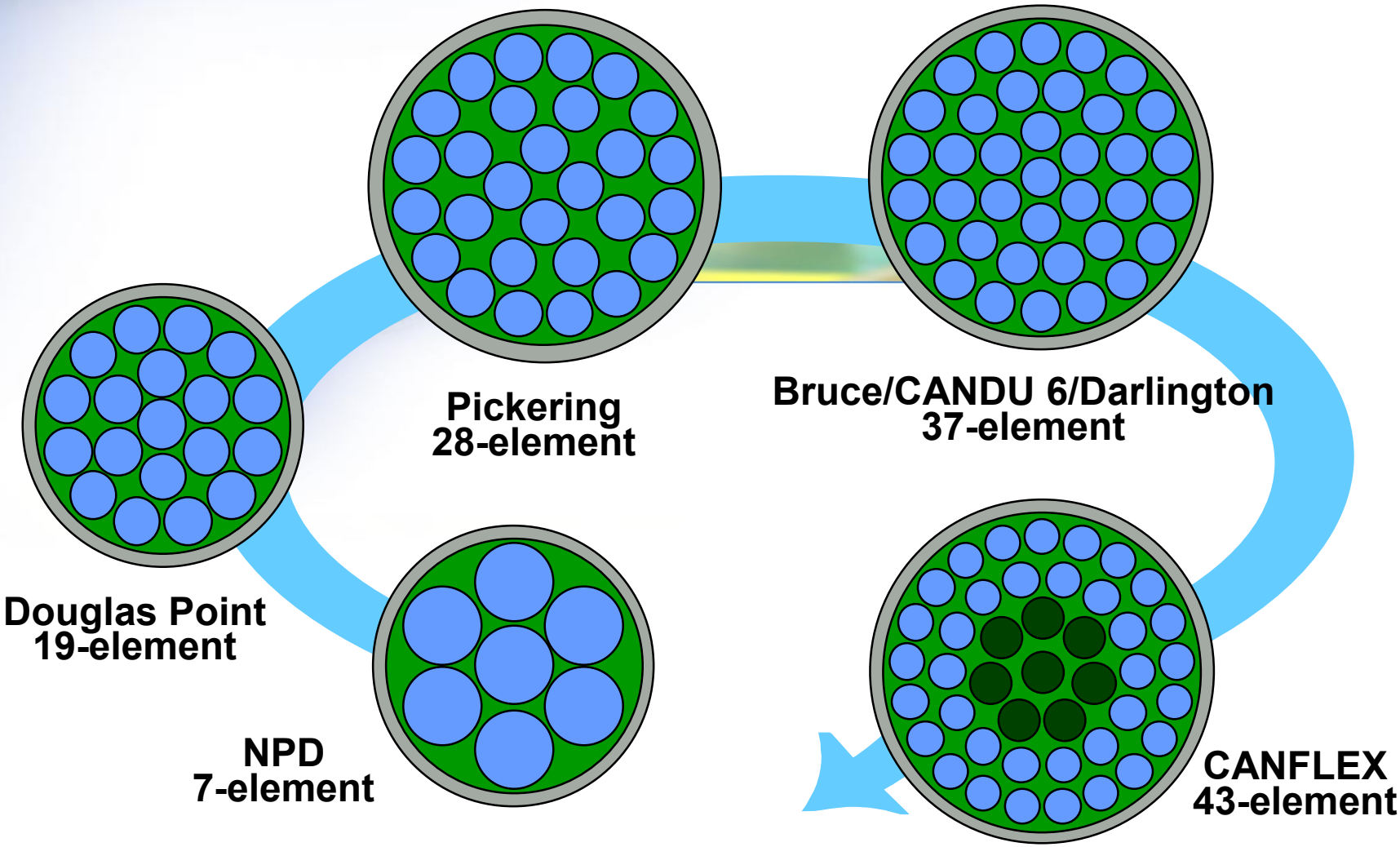
CHF-Enhancing Buttons (CANFLEX)

- Appendages are attached on the 1/4 and 3/4 bundle planes

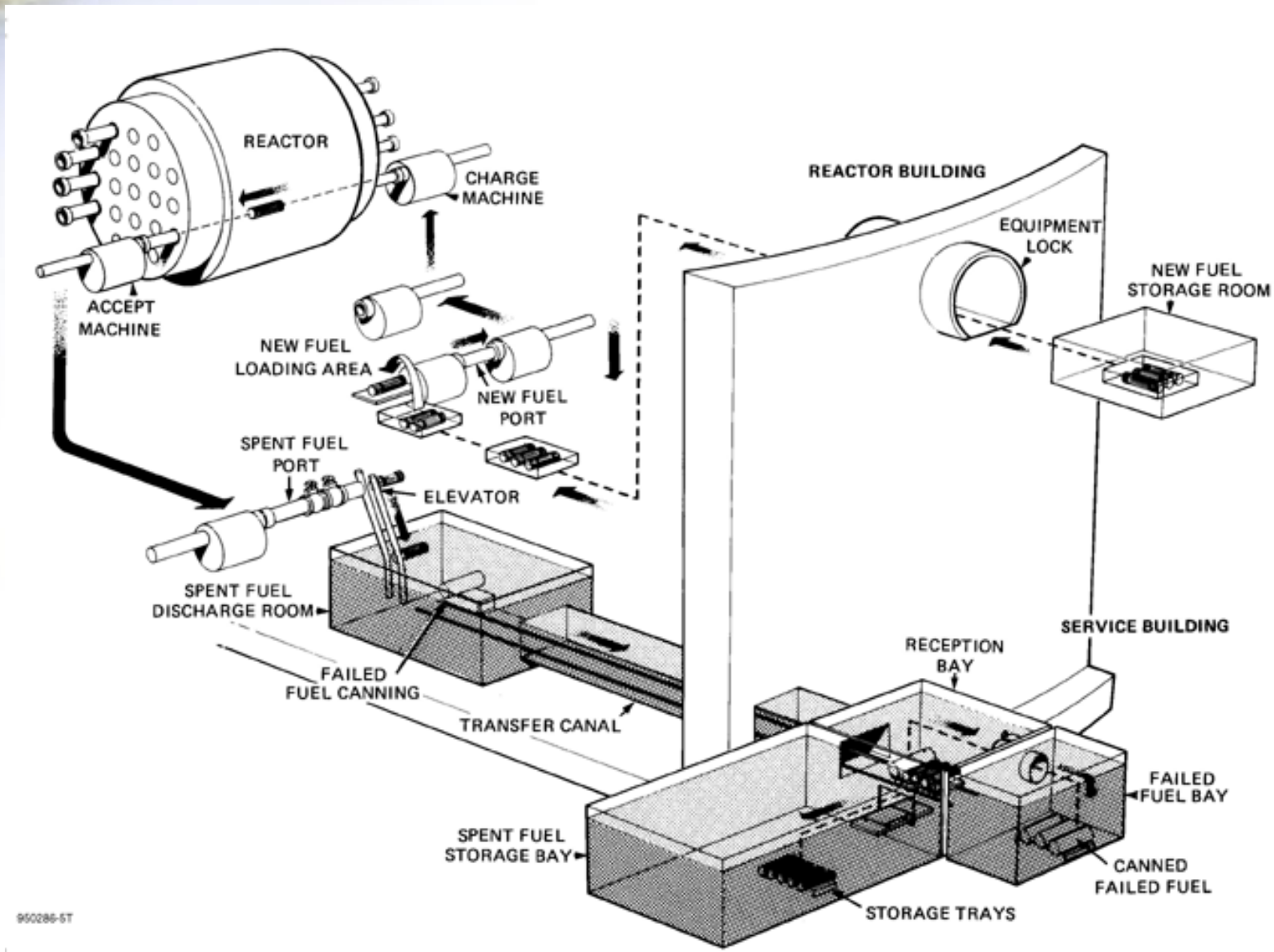




Evolution of CANDU Fuel

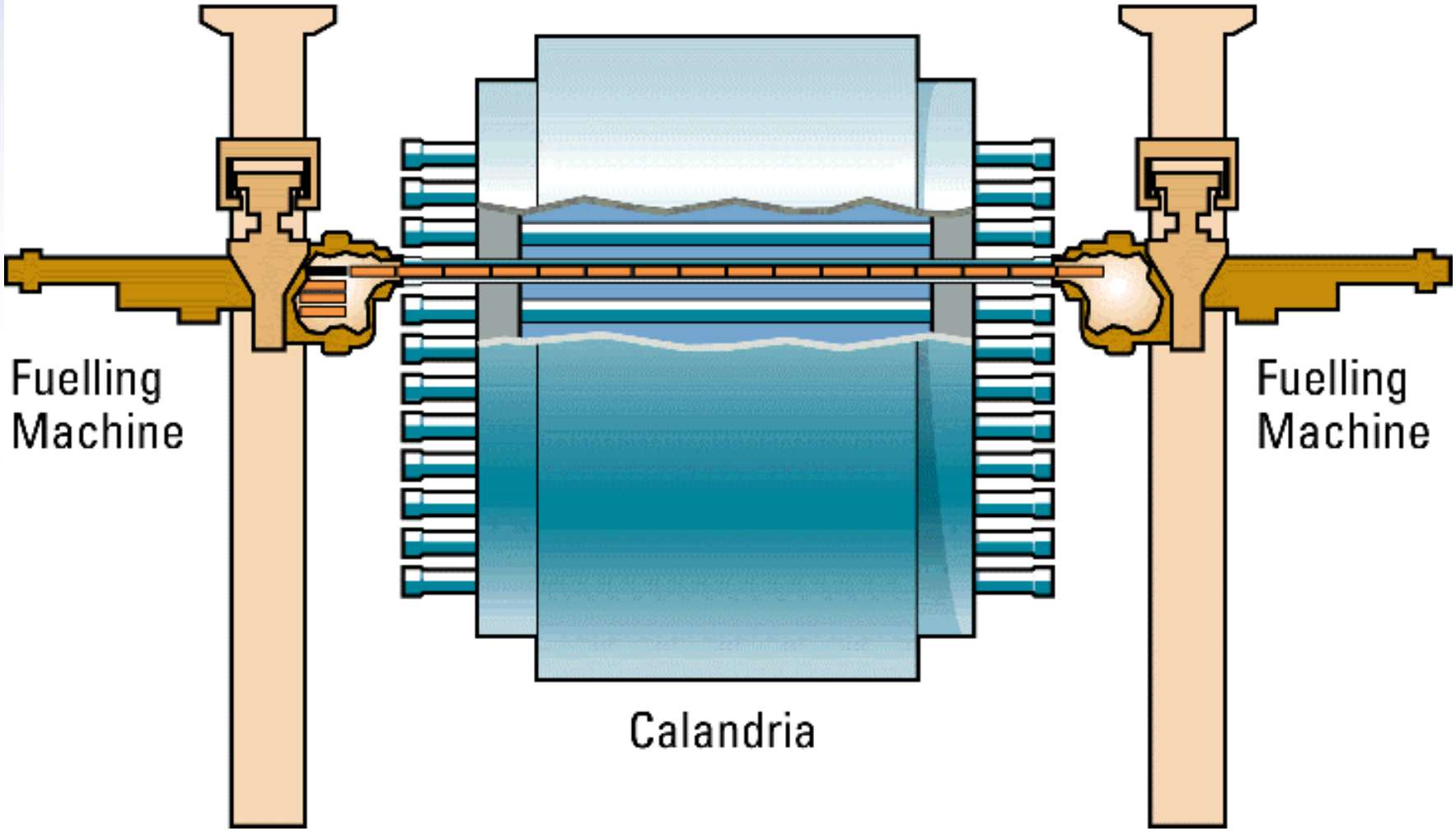


CANDU Fuel Duty Cycle

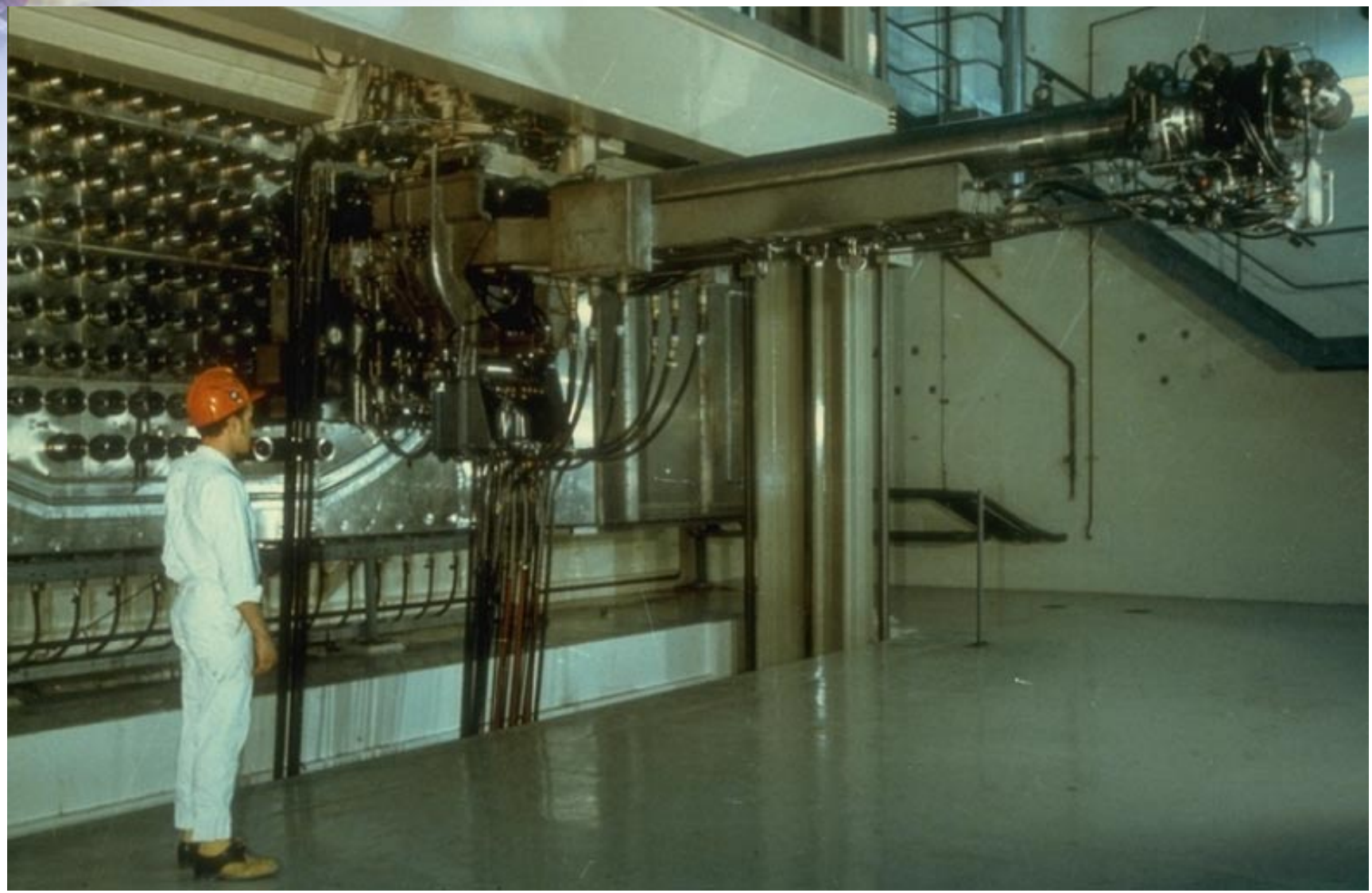


Loading CANFLEX at Point Lepreau in New Fuel Room

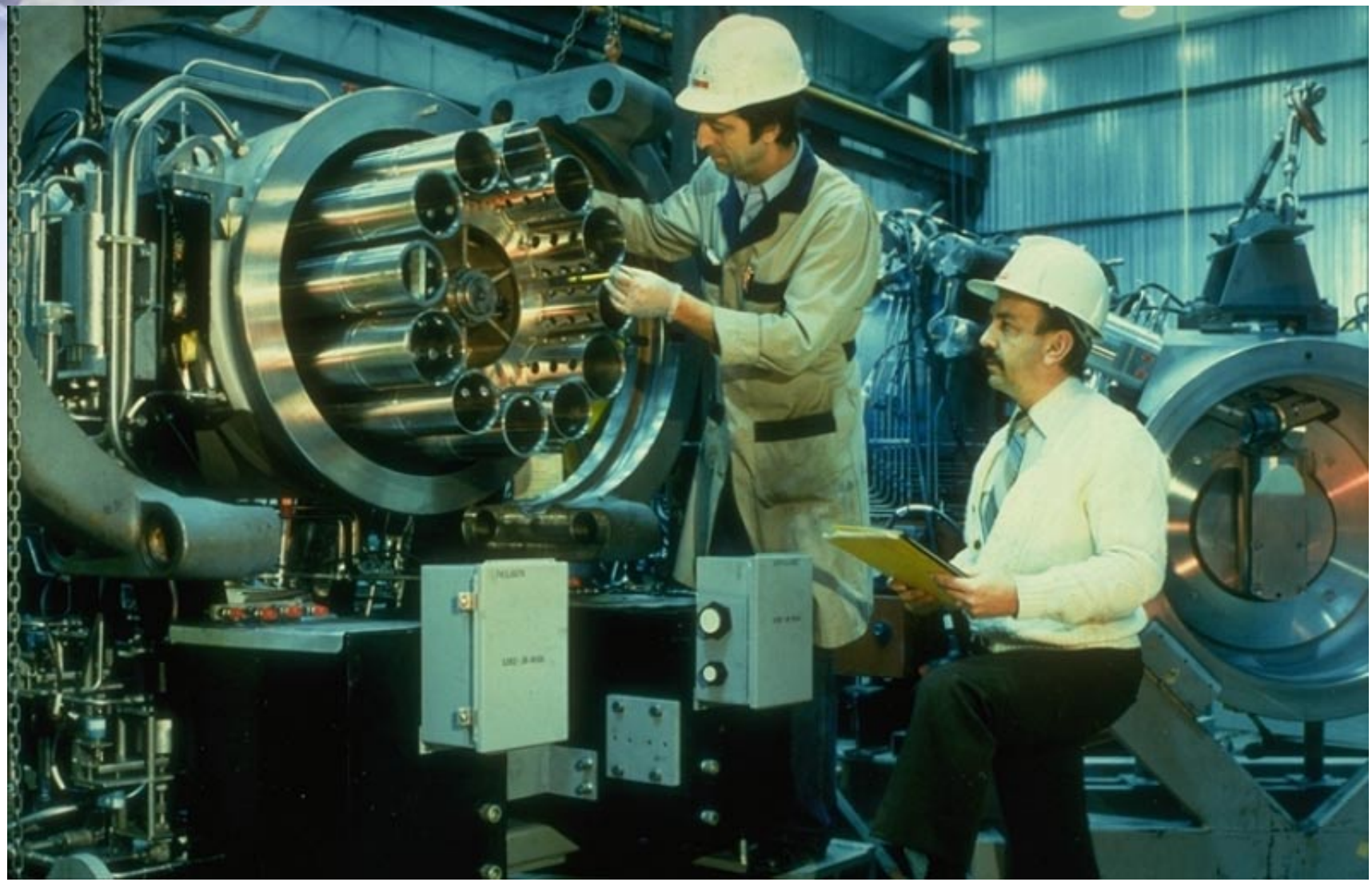
On-power Refueling



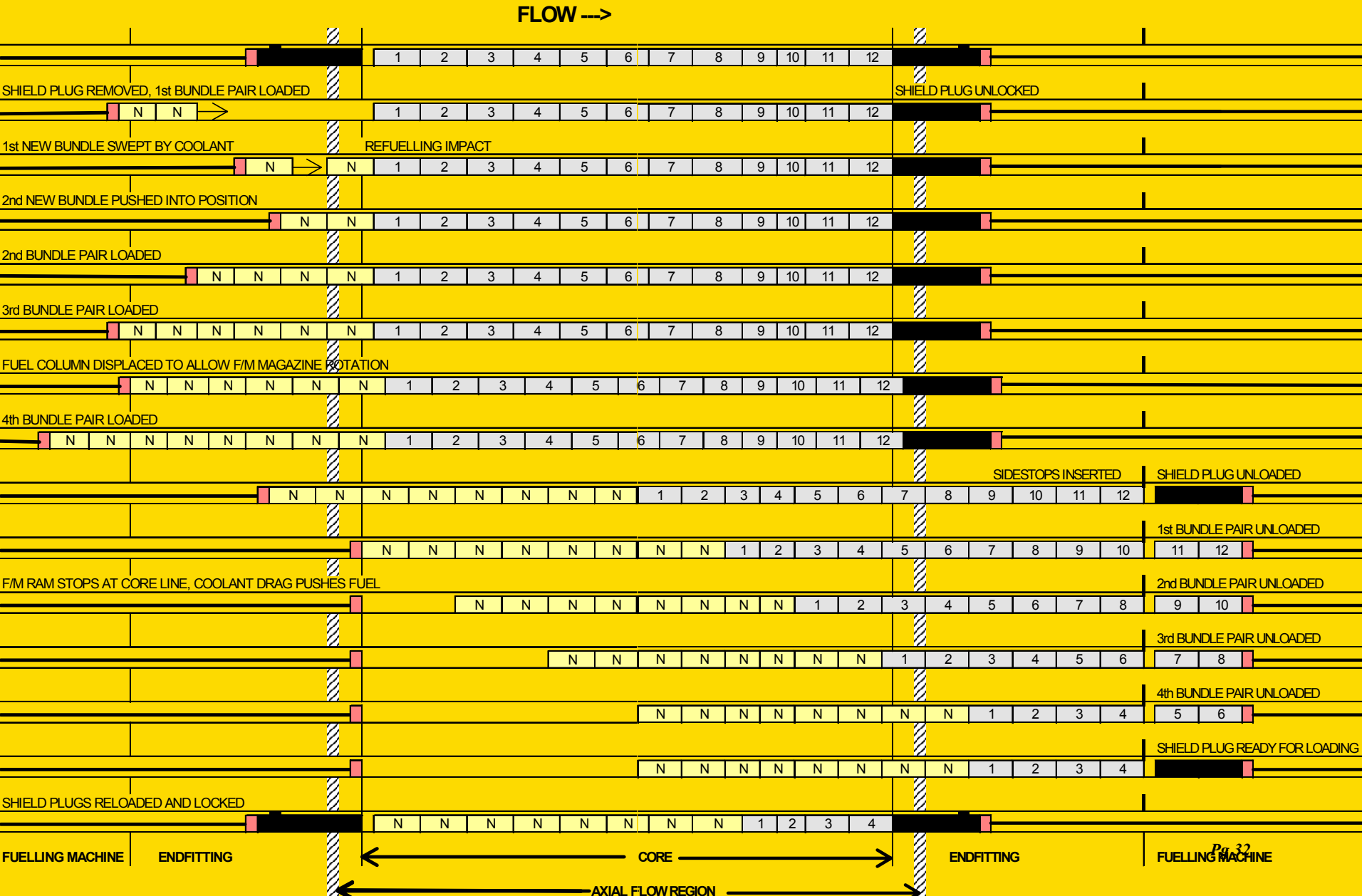
Pickering Fueling Machine



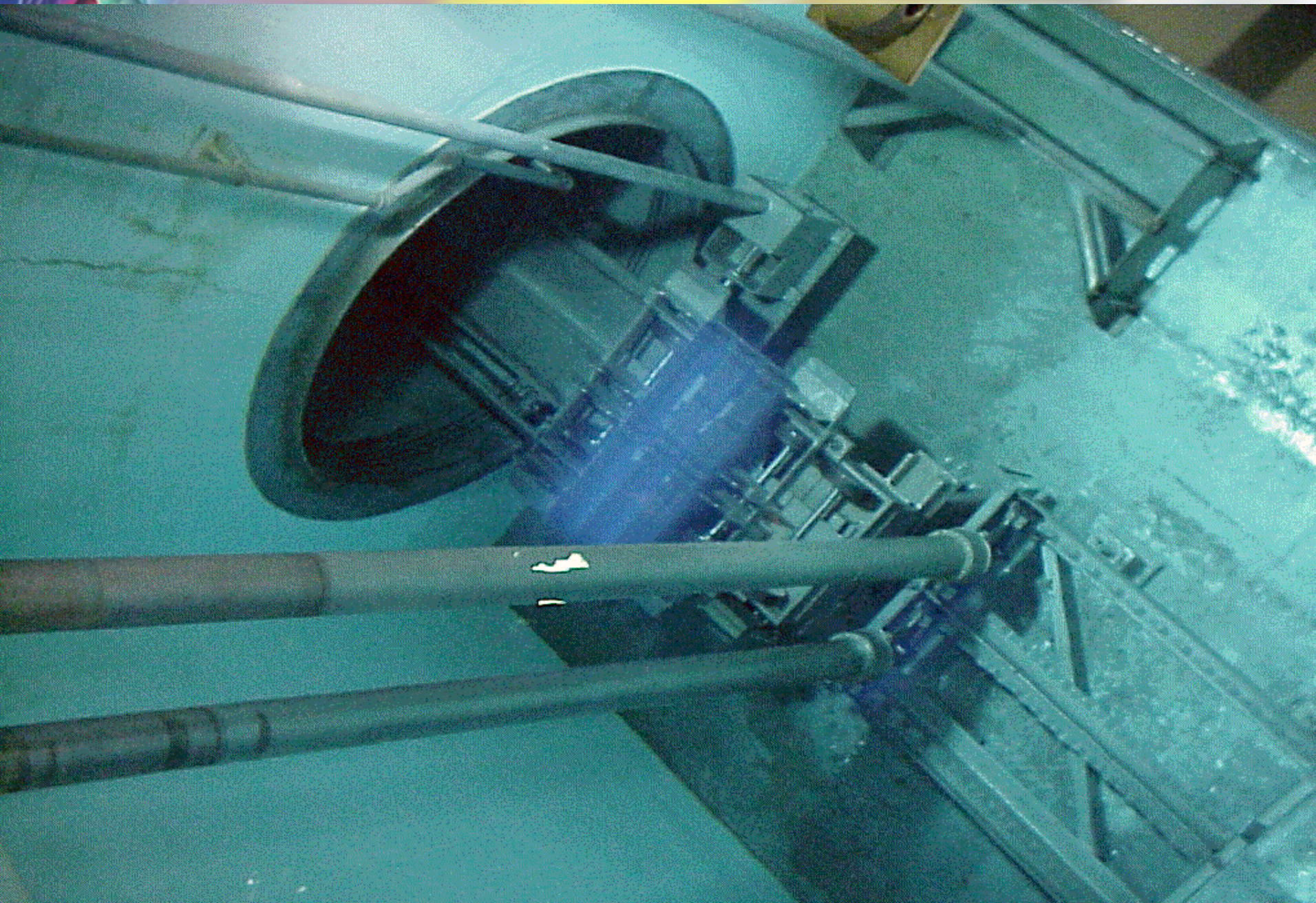
Fueling Machine Magazine



CANDU 6 Fuelling Scheme



Spent Fuel in Transfer Canal



Spent Fuel in Reception Bay

ACR Fuel Design

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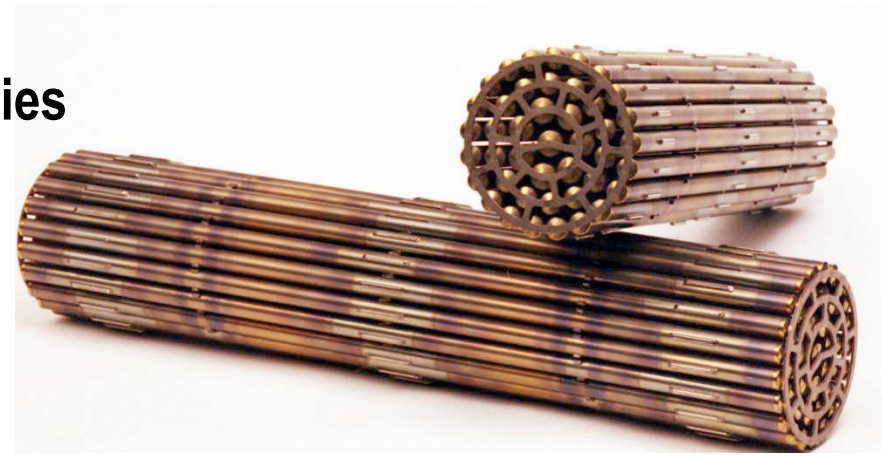
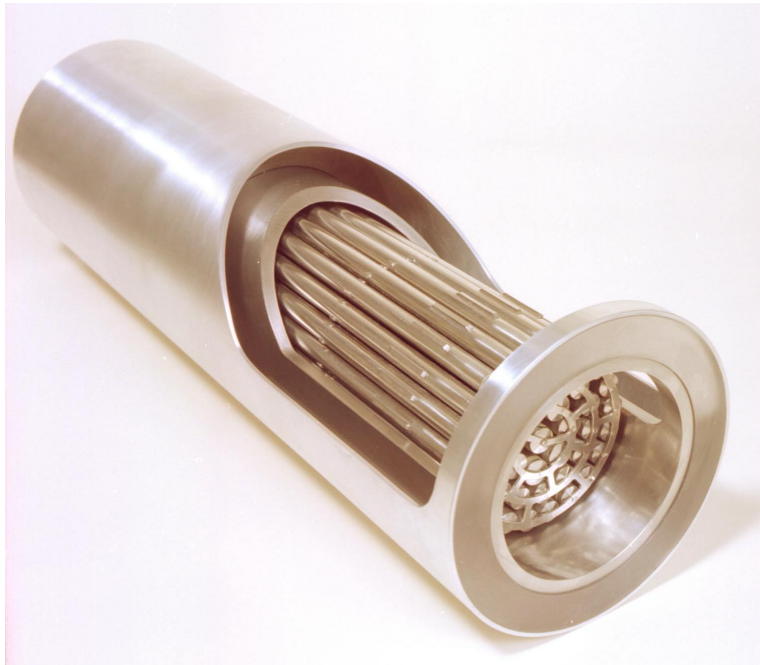
ACR Fuel Design

- Evolutionary extension of current fuel
  - extensive experience base on underlying technologies
- Based on 3 underlying technologies
  - CANFLEX Mk-4 geometry
  - low void reactivity fuel
  - extended burnup
- Key design features
  - 2.1% U^{235} in outer 42 elements
  - 7.5% Dy in nat. UO_2 in central element
  - 21 MWd/kg burnup



ACR Negative Coolant Void Reactivity

- Project requirement
  - negative CVR under all applicable design and operating conditions, accounting for calculation bias and uncertainties
- Negative void reactivity with light water coolant results from
  - tight lattice pitch
  - larger gap between pressure tube and calandria tube
  - Dy-doped central element
- Central poisoned element gives great flexibility in tailoring void reactivity coefficient
  - enrichment and Dy content can be independently varied to give desired burnup & void reactivity



Differences from CANFLEX Mk-4

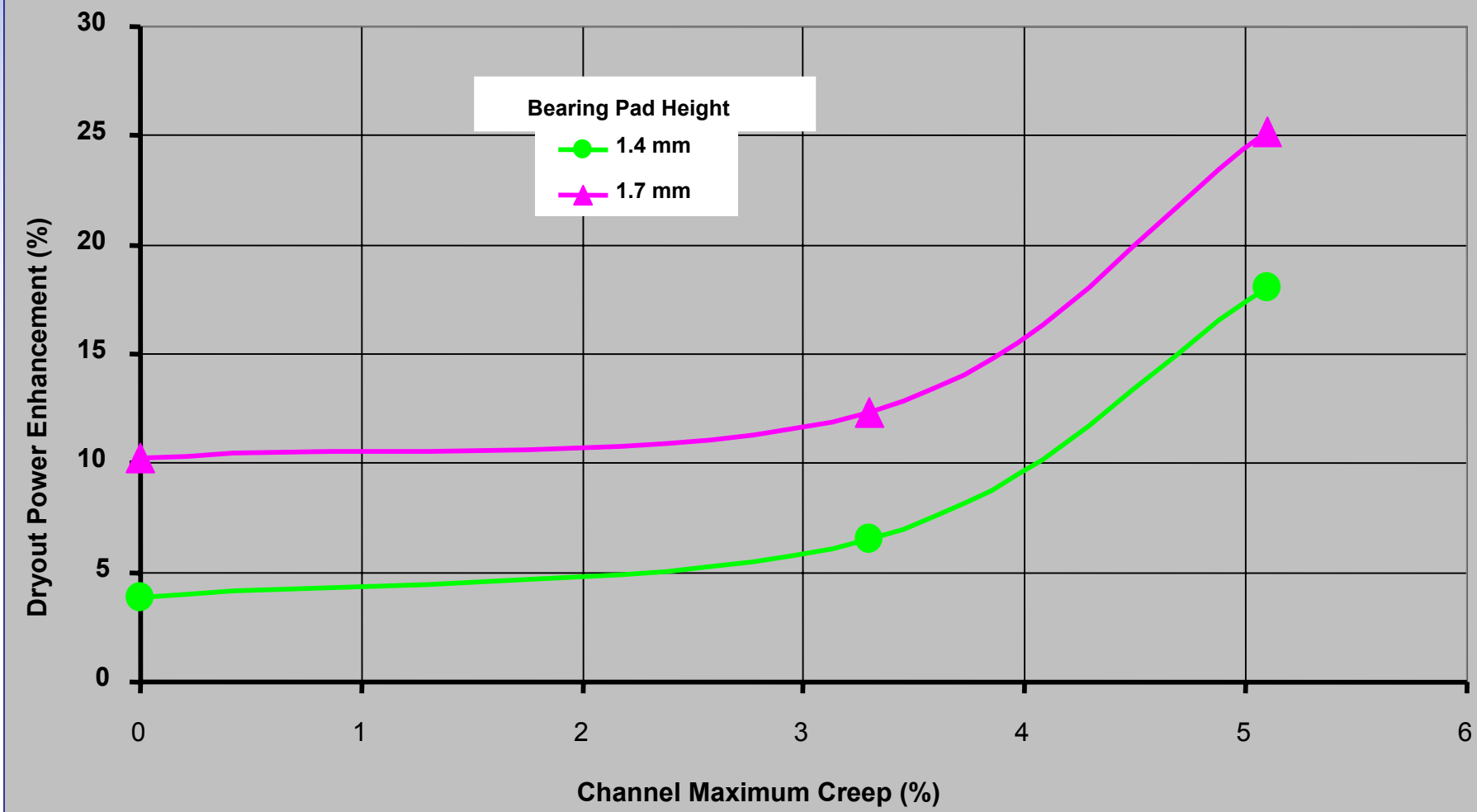
- Optimized pellet design
  - pellet design based on CRL irradiation experience
  - larger chamfers, deeper dishes, shorter pellets
    - reduces clad ridging
    - more internal void for accommodating FGR at higher burnup in smaller elements
  - length/diameter ratio, pellet shape
- Slightly thicker clad
  - to reduce potential of longitudinal ridging with higher coolant pressures and temperatures
  - reduces in-reactor element bow
- Slightly higher bearing pads
  - to increase CHF

Protected-Proprietary

Effect of Pellet Design on Performance

Protected-Proprietary

Effect of Bearing Pad Height on Dryout Power in CANFLEX NU



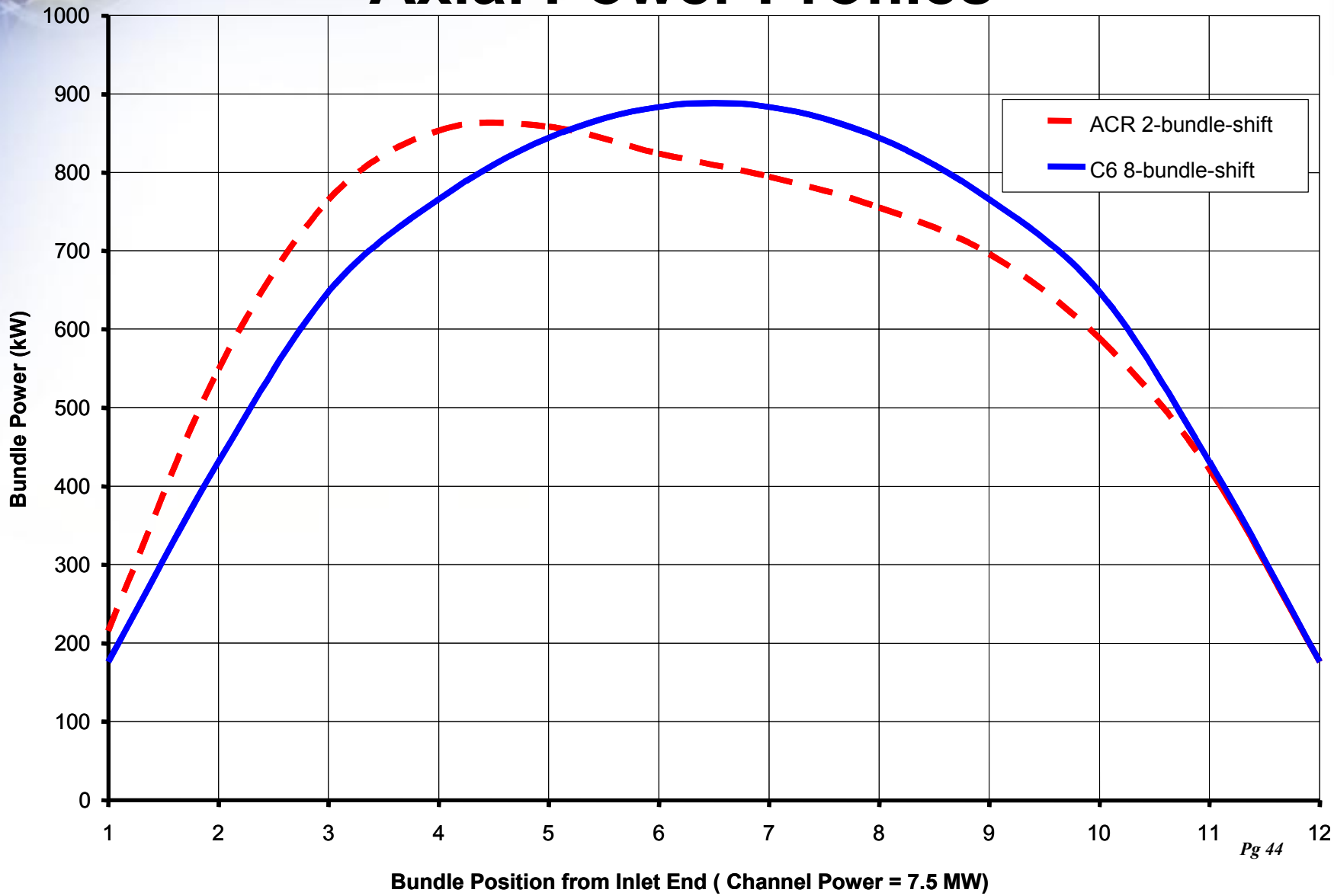
Differences from CANFLEX Mk-4 (con't)

- Endplate alignment change
  - "flipped endplates" reduce initial element bow
- "Low stress" endclosure welds
- Bruce style "square profile" endcaps
  - to interface with new fuel bundle separators (using Bruce and CANDU 6 technologies)
  - provides axial support to fuel column during on-power refueling

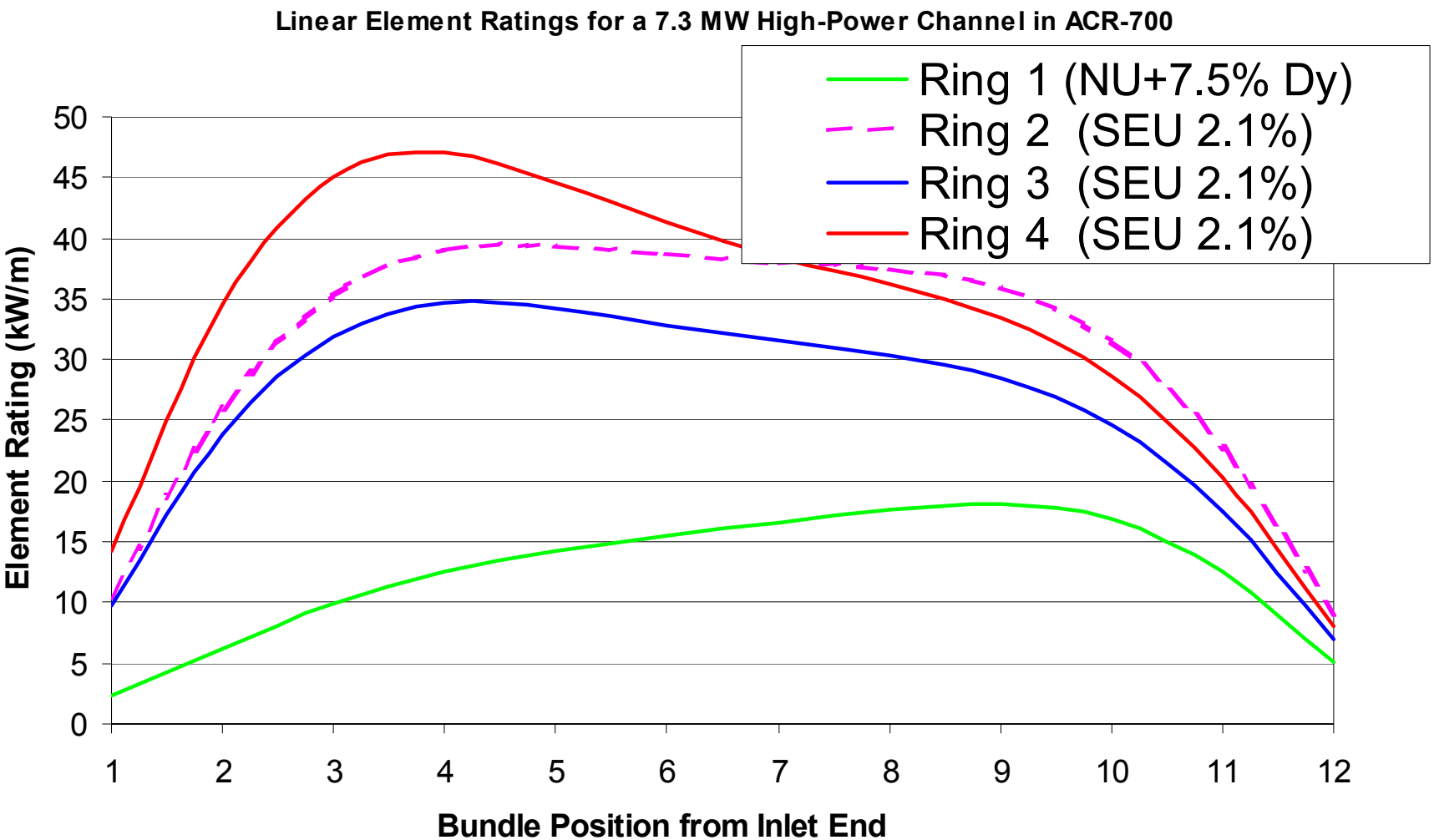
ACR Reference Fueling Scheme

- 2-bundle shift, bi-directional fueling
  - 2.8 channel visits per full power day
  - 5.6 fuel bundles per full power day
- Power peaks at inlet end of channel (bundle 4) and decreases along length of channel
- Excellent axial power distribution for
  - fuel performance
    - only fuel with low burnup experiences power ramp
  - thermal hydraulics
    - higher CHF than with either cosine-shaped or outlet-skewed axial power profile
    - at dryout location, bundle is well balanced for CHF (dryout occurs in inner ring of fuel)

Comparison of ACR and CANDU 6 Axial Power Profiles



ACR Power Envelope



Summary: ACR Fuel Design

- ACR fuel uses CANFLEX geometry
  - enriched uranium in outer elements
  - dysprosium mixed with NU in central element
- Other relatively minor changes to CANFLEX Mk 4 to accommodate ACR requirements

Experience Base for ACR Fuel

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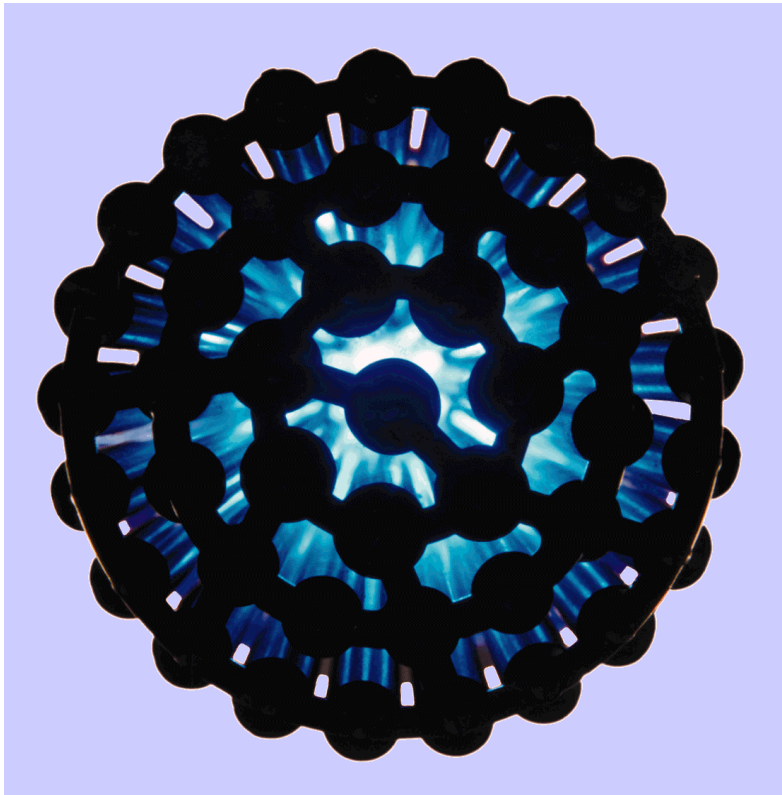
ACR Technology Base

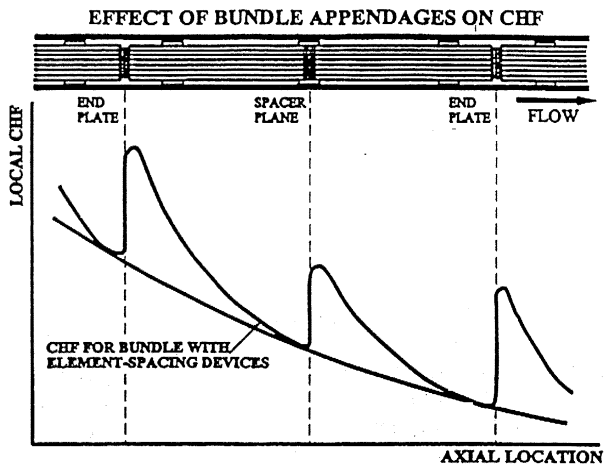
- CANFLEX
  - development and qualification
- Enriched fuel / extended burnup experience
- Low Void Reactivity Fuel (LVRF)
  - dysprosium-doped fuel
- Generic advanced fuel development in AECL

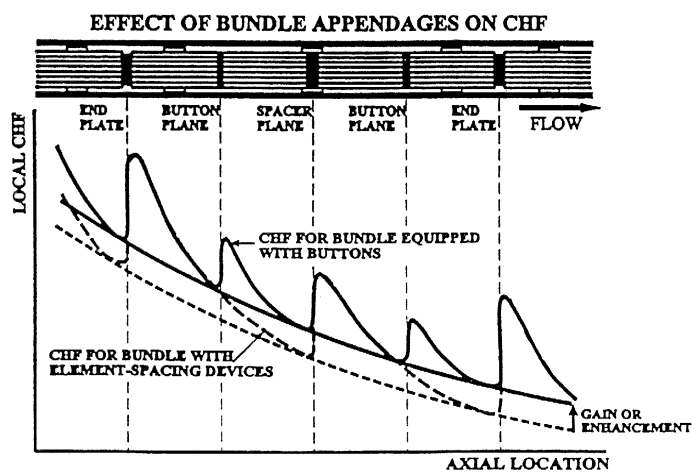
CANFLEX Fuel

- 43 elements, 2 sizes
  - 8 inner elements 13.5 mm (0.53”) in diameter
  - 35 outer elements 11.5 mm (0.45”) in diameter
- ~20% lower peak rating than for 37-element fuel (for same bundle power)
  - facilitates achievement of higher burnup
- CHF-enhancing buttons
  - increase coolant turbulence
  - higher operating margins



Effect of Appendages on CHF





Summary of CANFLEX NU Qualification

- Design requirements documented in Design Requirements, Design Verification Plan
- Tests and analysis confirmed that CANFLEX met all requirements
  - strength
  - impact and cross-flow
  - fueling machine compatibility, endurance
  - sliding wear
  - fuel performance (NRU irradiations)
  - CHF thermal hydraulic
- Design qualification program documented in Fuel Design Manual
- Demonstration irradiation (DI) in Point Lepreau 1998 to 2000
  - supplemented formal fuel qualification
  - 2 channels, 24 bundles
  - irradiation of 24 bundles currently taking place in Wolsong 1
- Ready for commercial implementation in CANDU 6 reactor

Documentation for Qualification Tests

- Qualification testing for CANDU fuel follows a set of AECL procedures that meet the requirements of CAN3-N286.2, "Design Quality Assurance for Nuclear Power Plants"

- Test Specifications (includes acceptance criteria)
  - Component Verification Specifications (CVS)
- Test procedures
  - prepared to ensure tests meet the requirements of the CVS
- Test Reports
  - Component Verification Reports (CVR)

Qualification Test Comparison (37/43 fuel designs)

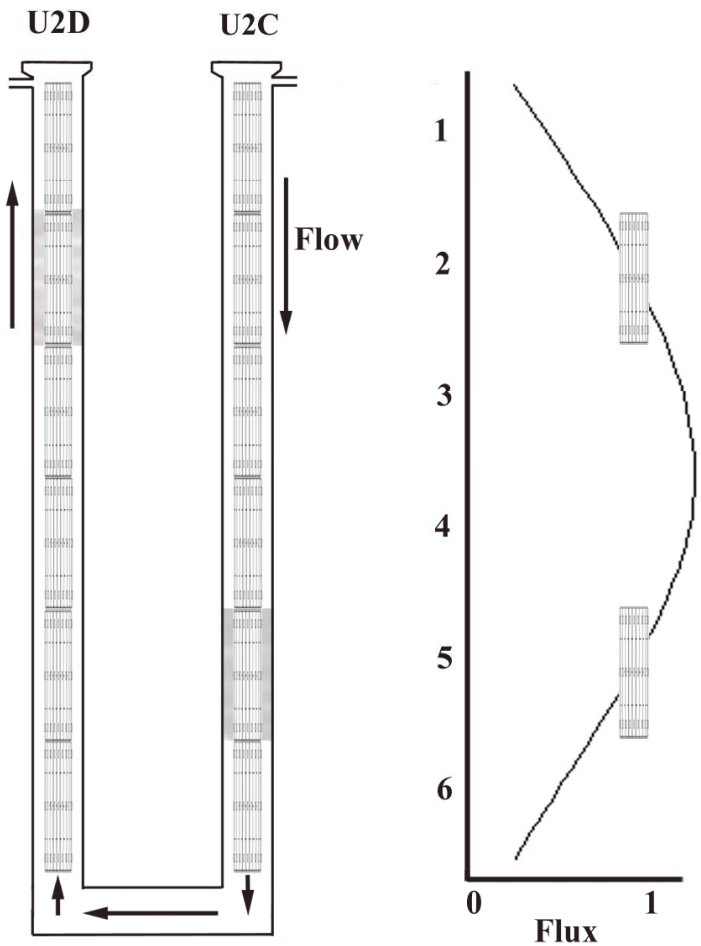
	CANDU 6 37-element	CANDU 6 CANFLEX NU
In-Reactor		
High Power	1	2
Power Ramp	1	2
High Burnup	0	1
Out-Reactor		
•Pressure Drop (full channel)	1	1
•Endurance	2	1
•Strength	1	1
•Refueling Impact	2	1
•Sliding Wear	1	0 (assessment only)
•Cross Flow	0	2
•Bundle Rotation Test (DP)	1 (high pressure)	1 (low pressure)
•Seismic	1	0 (assessment only)

NRU Tests

- Objective to demonstrate acceptable irradiation performance of CANFLEX UO_2 fuel elements for both NU and SEU fuel cycles
- 8 CANFLEX UO_2 bundles irradiated in NRU
  - overall performance similar to 37-el bundle
  - wide range of powers and burnups
- CANFLEX-NU requirements bounded
  - high power (typical and bounding powers)
  - power ramp (refueling power ramps simulated)
  - burnup (normal and high burnup)

NRU Fuel Irradiation Loops

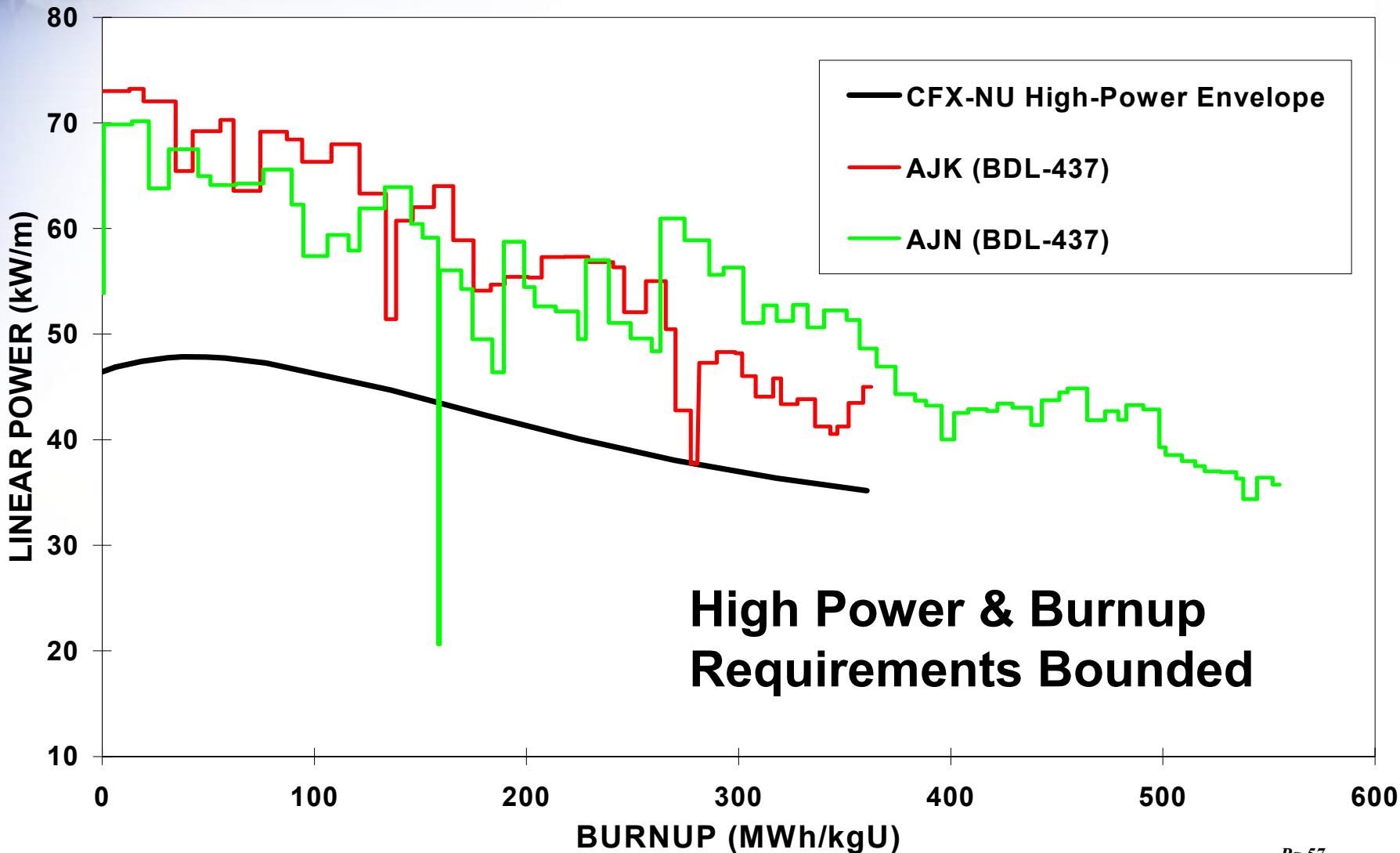
- Irradiation in 6-bundle fuel strings (full scale) in U1/U2 loops of the NRU reactor in Chalk River
- 3 test sections available (can accommodate 18 CANDU fuel bundles)
- Central element removed from bundles for NRU tie-rod
- Typical CANDU operating conditions
  - 260-305 ^{o}C
  - 9.5-10.9 MPa
  - similar chemistry
- Movement of bundle in string can produce power ramp



CANFLEX Irradiations

TEST	BUNDLE	POWER (kW/m)	BURNUP (MWd/kgU)
BDL-437	AJJ	35-69	12.9
	AJK	29-73	15.5
	AJL	47-83	3.8
	AJM	27-69	22
	AJN	29-70	23
BDL-440	AKT	20-59	7.3
BDL-443	AKV	19-49	18.3
	AKW	18-45	13.0+

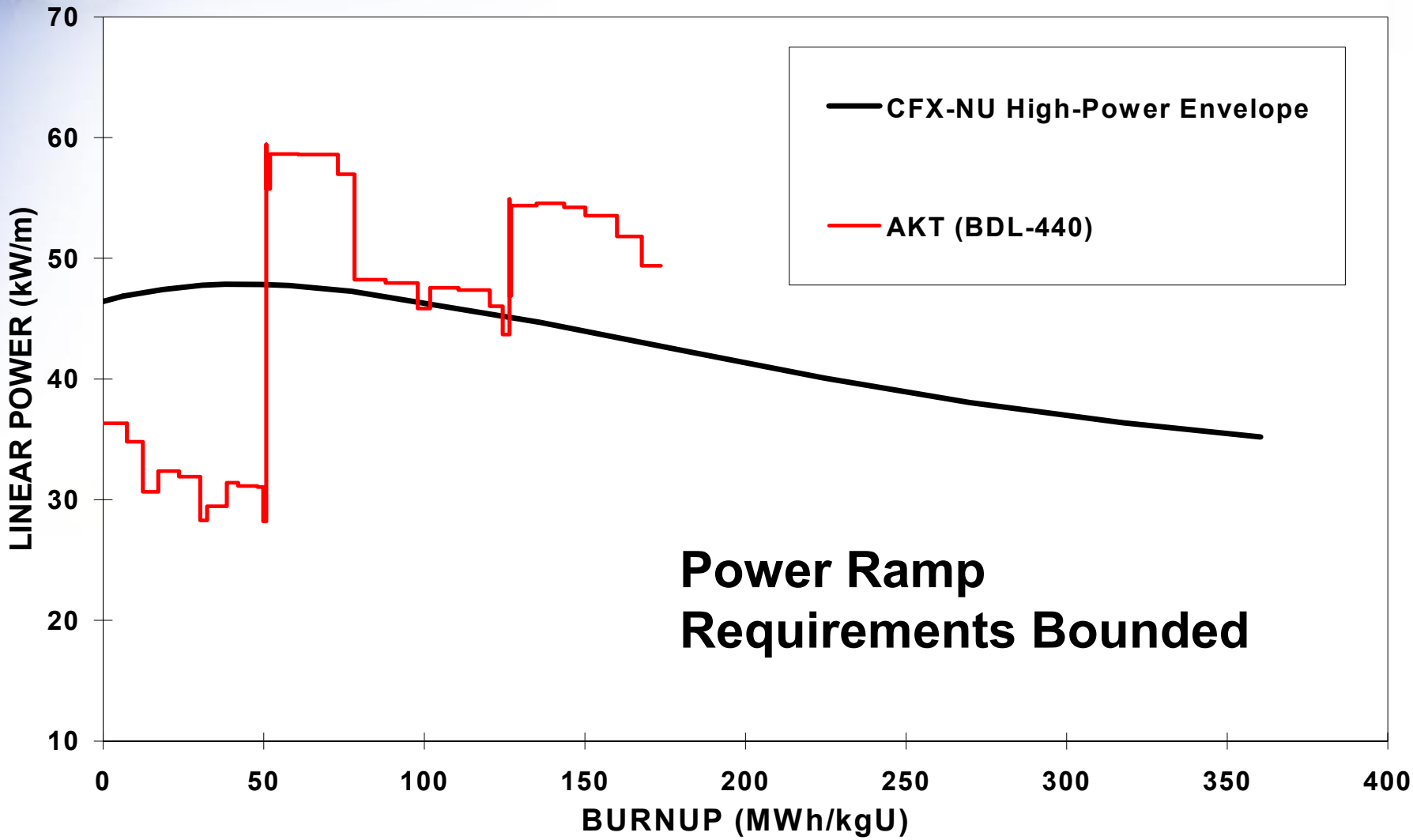
Summary of NRU High Power Tests



SEU Elements – Bundle AJN

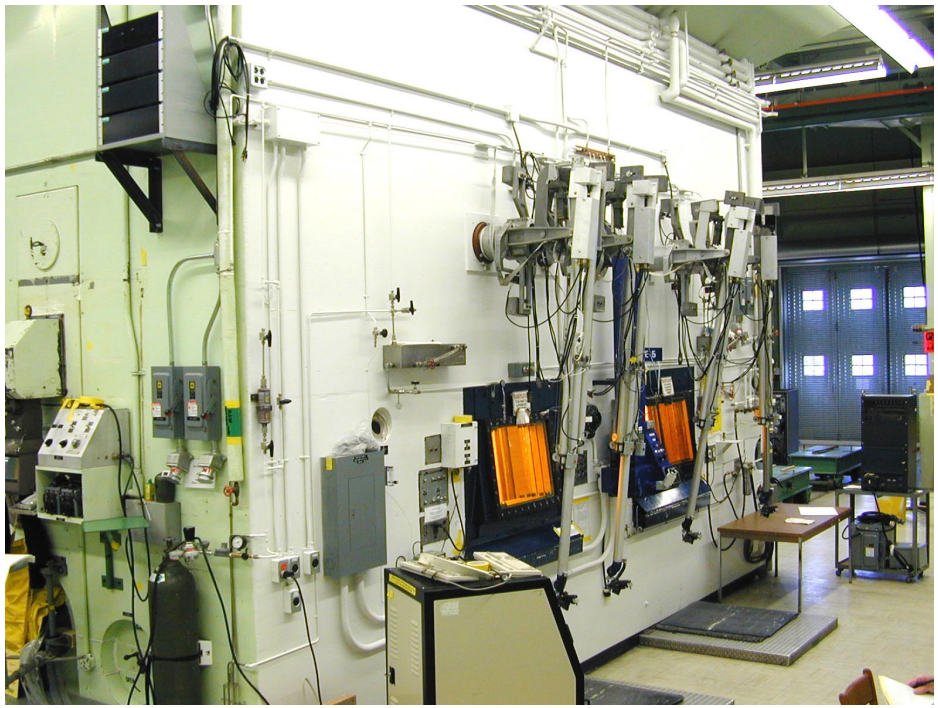
	Max Power (kW/m)	Burnup (MWd/kg)	Sheath Strain (%)	FGR (%)	Condition
Outers	70	34	1.0	25	Intact
Inters	46	17	0.0	3	Intact
Inners	49	14	-0.2	8	Intact

Summary of NRU Power Ramp Tests



PIE Measurements

- Visual examinations
- Fuel dimensioning
- Gamma scanning
- Gas-puncture analysis
- H/D analysis
- Burnup measurement
- Clad metallography
- Pellet ceramography



Out-Reactor Qualification Tests

- Thermal hydraulic / pressure drop*
- Strength*
- Fueling machine compatibility*
- Refueling impact*
- Endurance*
- Cross flow*
- Sliding wear
- Seismic

* 6 out-reactor tests performed for CANFLEX

Strength Tests

- Objective: CANFLEX fuel can successfully withstand axial drag loads when downstream bundle is supported either by both side stops (normal operation) or by single side stop (abnormal operation)
- Tests showed no significant distortion
  - fuel element length change and/or endplate profile change
  - bent tube gauge test
  - bearing pad wear, marking of the fuel element endcaps

Fueling Machine Side-Stops for CANFLEX Strength Tests

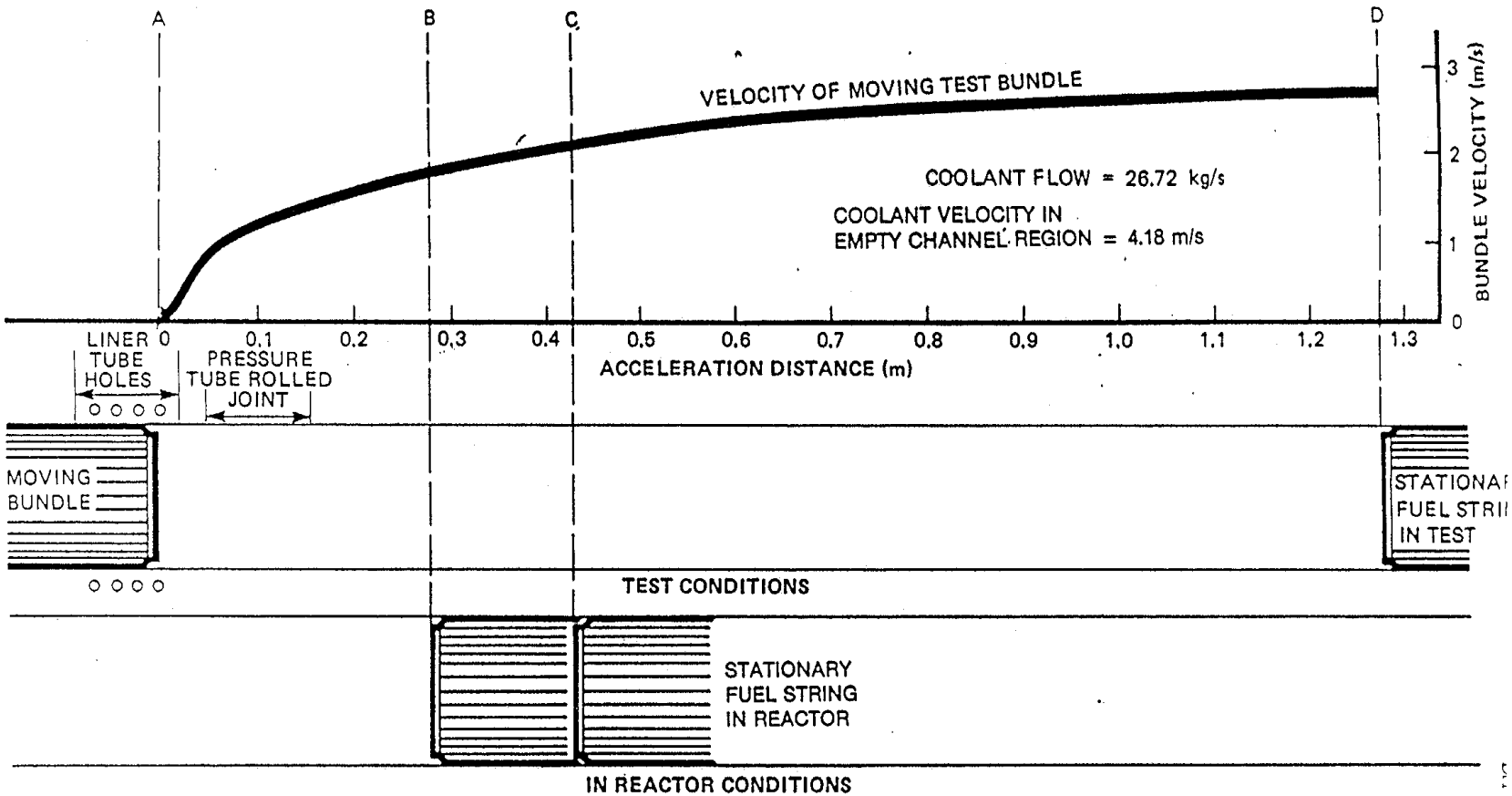
Fueling Machine Compatibility Test

- Objective: CANFLEX fuel bundles are compatible with CANDU 6 fueling machine and with grappling tool
- Tests performed in Wolsong fueling machine and full scale test fuel channel under typical reactor operating conditions
- 10 bench tests of the CANDU 6 fueling machine grappling tool were performed, with two different bundle orientations, on two separate bundles, for a total of 40 tests

Refueling Impact Test

- Objective: fuel can successfully withstand the normal refueling impacts
  - up to 0.5 m acceleration distance, 30 kg/s channel flow
- Tests showed
  - no significant distortion or damage to the fuel bundle endplate, or to the fuel elements
  - waviness of the endplates within the bundle specification
  - bundles passed through the kinked tube gauge test
  - no visible damage to the pressure tube from impacts of the fuel bundle
    - depth of scratches less than the design allowance for pressure tube wear

Refueling Impacts (CANDU 6)



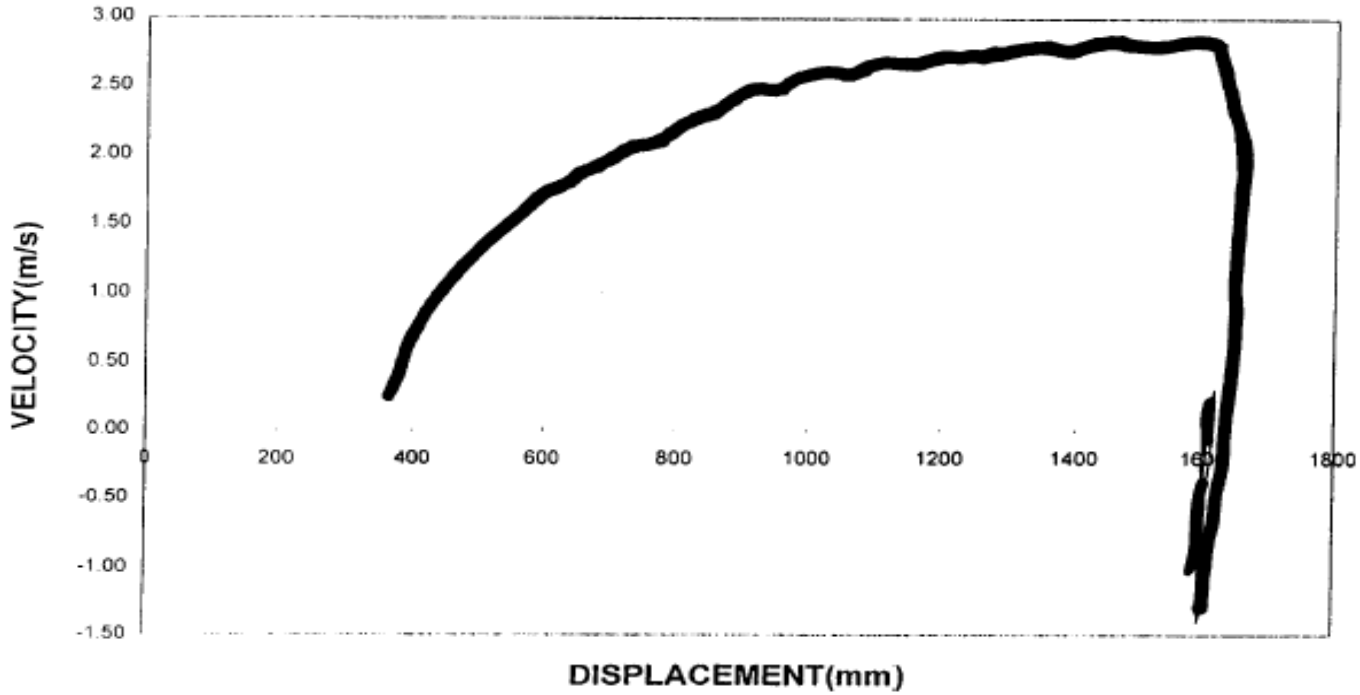
A – POSITION OF MOVING BUNDLE BEFORE ACCELERATION

B – POSITION OF FUEL STRING IN-REACTOR BEFORE PRESSURE TUBE AXIAL CREEP

C – POSITION OF FUEL STRING IN-REACTOR AFTER PRESSURE TUBE AXIAL CREEP

D – POSITION OF FUEL STRING IN OUT-REACTOR TEST

Refueling Impacts (CANFLEX)



Velocity versus Displacement of the Moving Bundle
(Measured by Vel./Disp. Transducer)

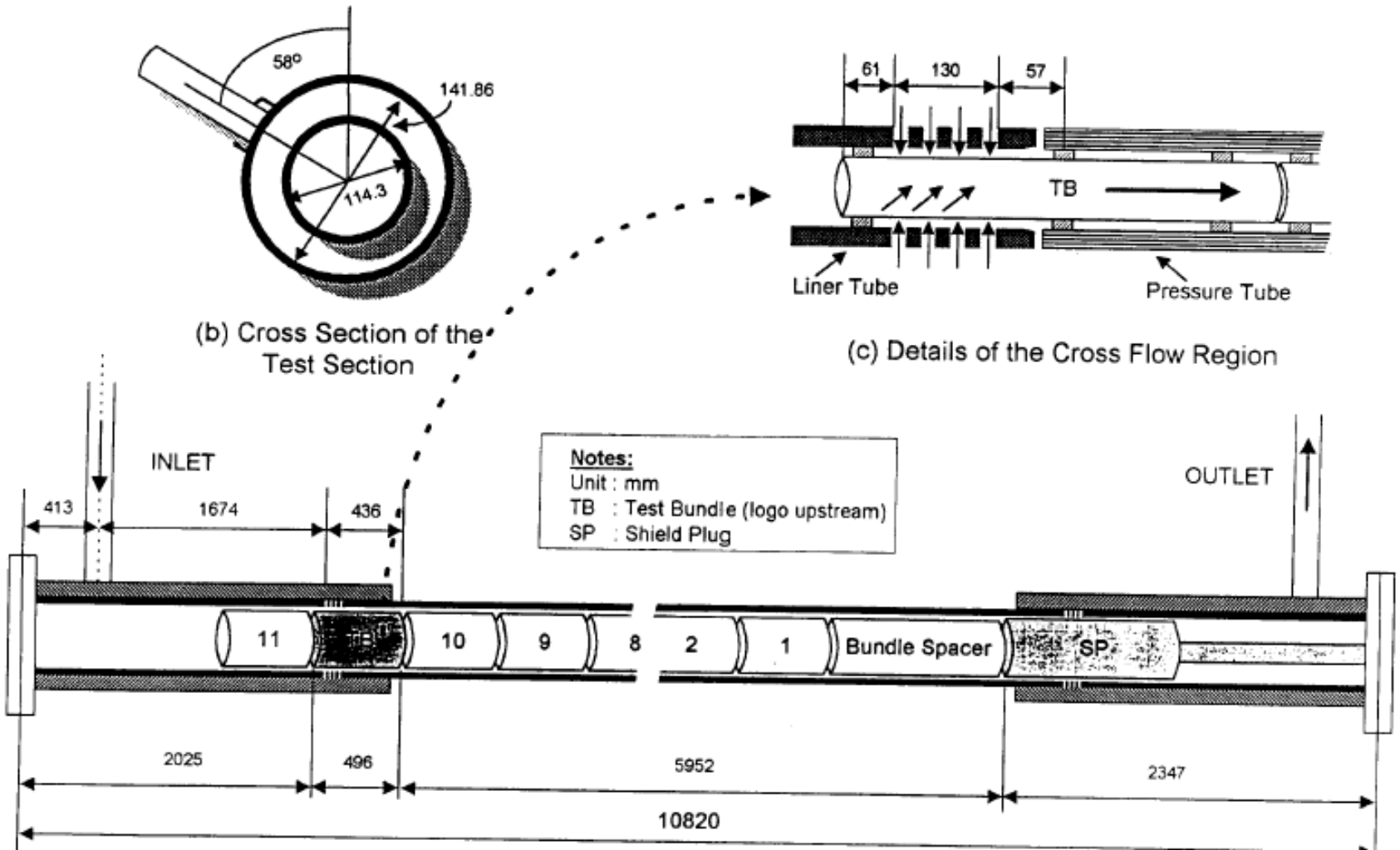
Mechanical Fretting Endurance Test

- Objective: fretting of the pressure tube and CANFLEX fuel bundles are acceptably low under in-reactor flow, temperature and pressure conditions, for representative bundle dwell periods
- Tests showed
  - material loss due to fretting of the inter-element spacers and of the pressure tube within the wear allowances

Cross-flow Endurance Test

- Objective: CANFLEX fuel can successfully survive up to 4 hours in the cross-flow region of the liner of a CANDU 6 channel
  - holes in the liners of the fuel channel inlet and outlet end fittings are locations where coolant enters and exits causing high coolant velocities in the radial direction (normal refueling results in bundles being in cross flow for only a few minutes)
- Tests showed
  - bundle meets all dimensional requirements & was free of failures of the endplate-to-endcap welds, and free of cracks in the endplates
  - inter-element spacers maintained minimum spacing between elements of 1 mm
  - bundle passed the bent tube gauge test
  - there was no spacer interlocking

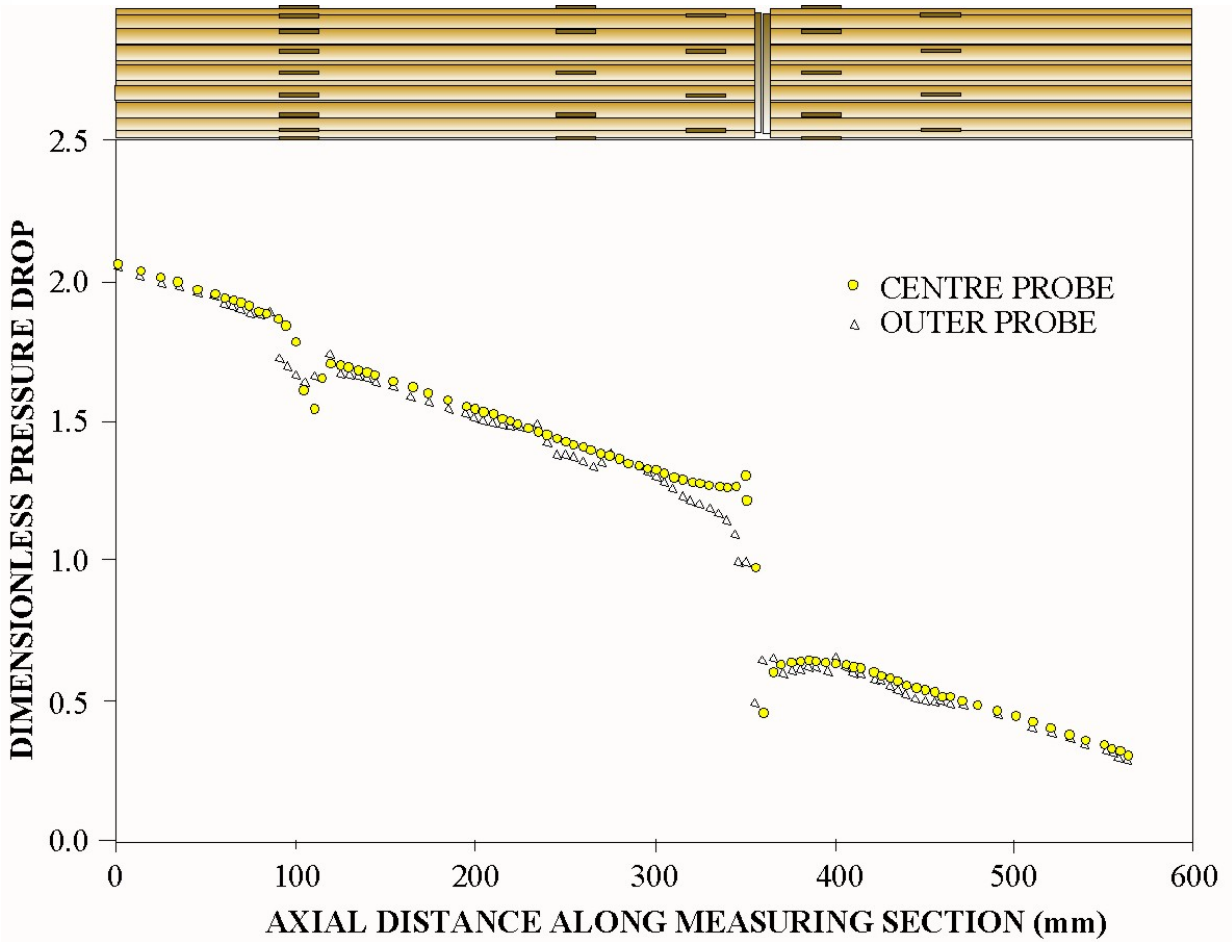
Test Bundle Location in Cross-flow Region



Thermal Hydraulic Parameters

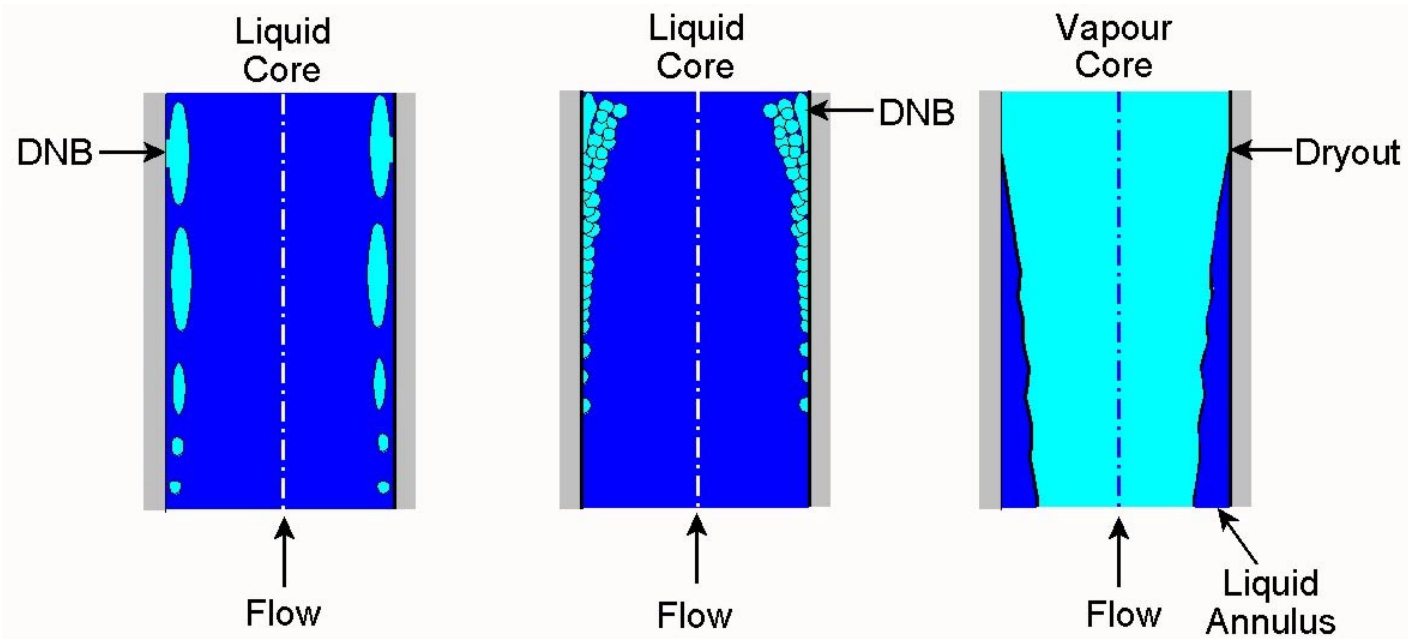
- Fuel string pressure drop
  - establishes channel flow based on pump characteristics
- Critical Heat Flux (CHF)
  - determines trip set-points for
    - Neutron Overpower Protection (NOP) system (loss-of-regulation accident)
    - process trip parameters for other accidents (such as loss-of-flow)
  - a determinant in setting reactor power, operating margins
- Post-dryout (PDO) behavior
  - establishes behavior in operation beyond dryout
    - heat transfer, and drypatch stability and spreading

Axial Pressure Profiles Along Fuel Bundles



Critical Heat Flux

- Phenomenon corresponding to point where continuous liquid contact cannot be maintained at the heated surface
- CHF mechanism corresponds to dryout at CANDU (& ACR) conditions of interest (high flows and high qualities)

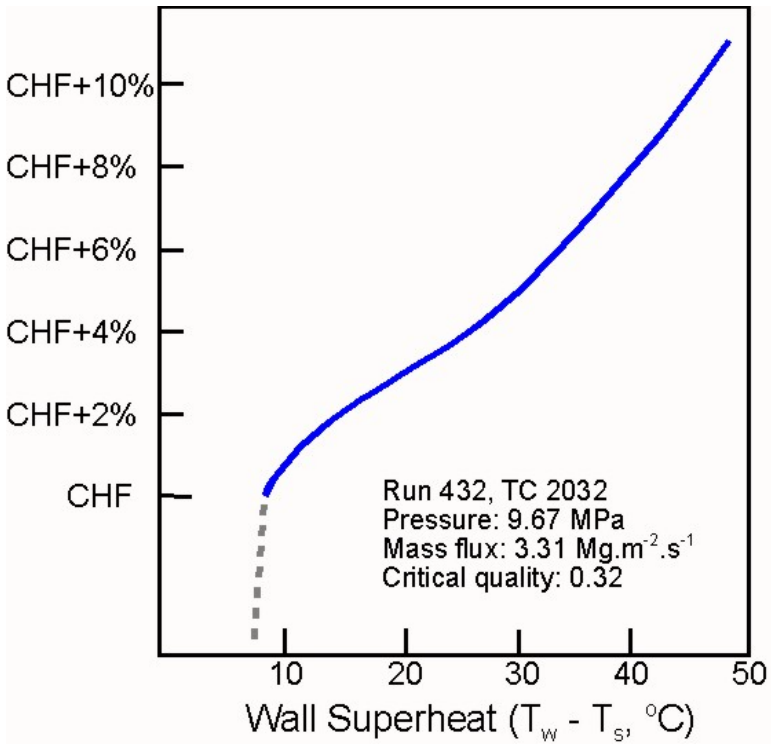


CHF in PWR

CHF in CANDU

Dryout Characteristics

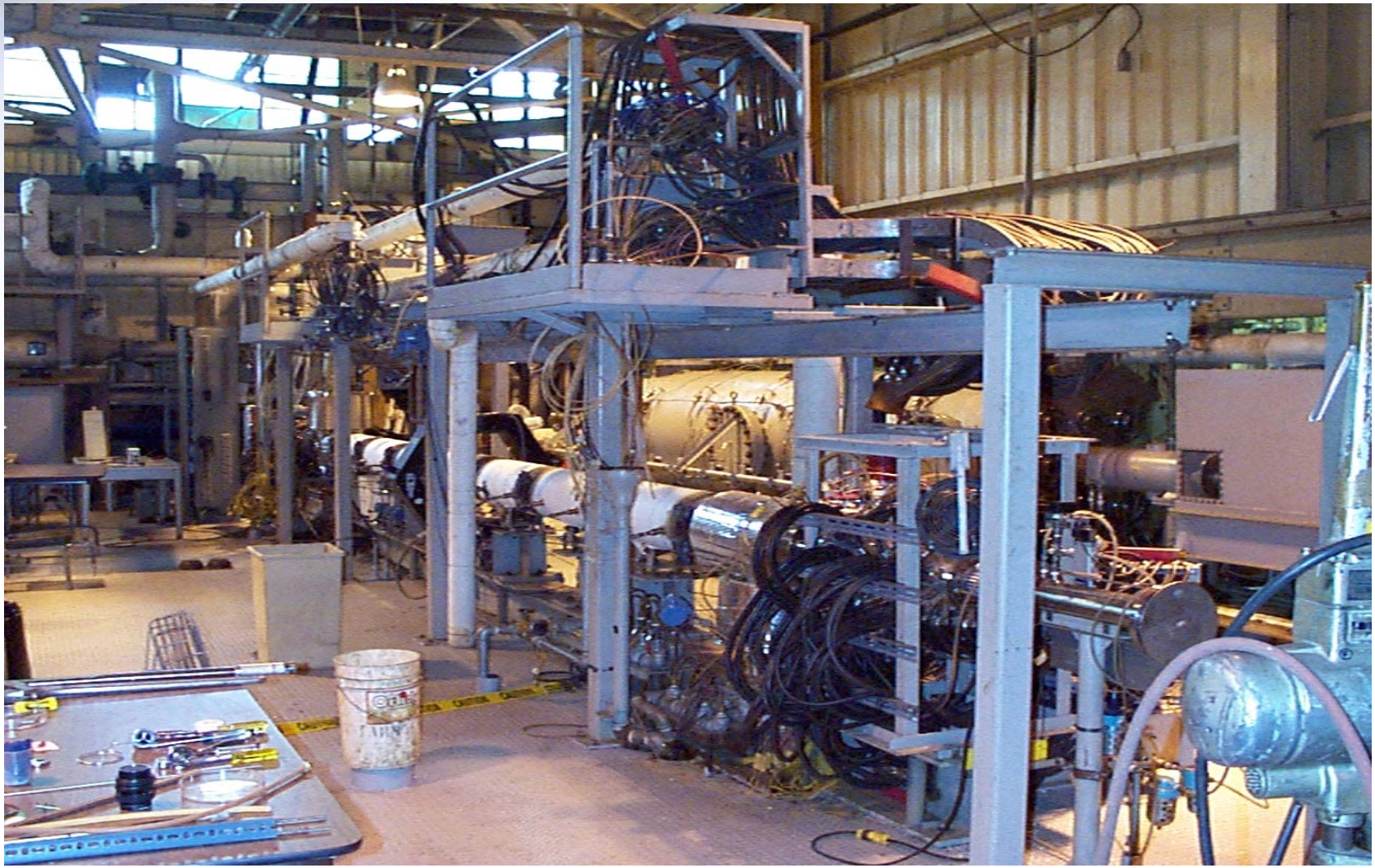
- Onset of intermittent dryout criteria is used
  - only a single point of the fuel clad may encounter dryout while the remainder of the bundle remains wet
- Clad temperature increases only slightly beyond dryout
- No severe consequence to fuel or clad integrity



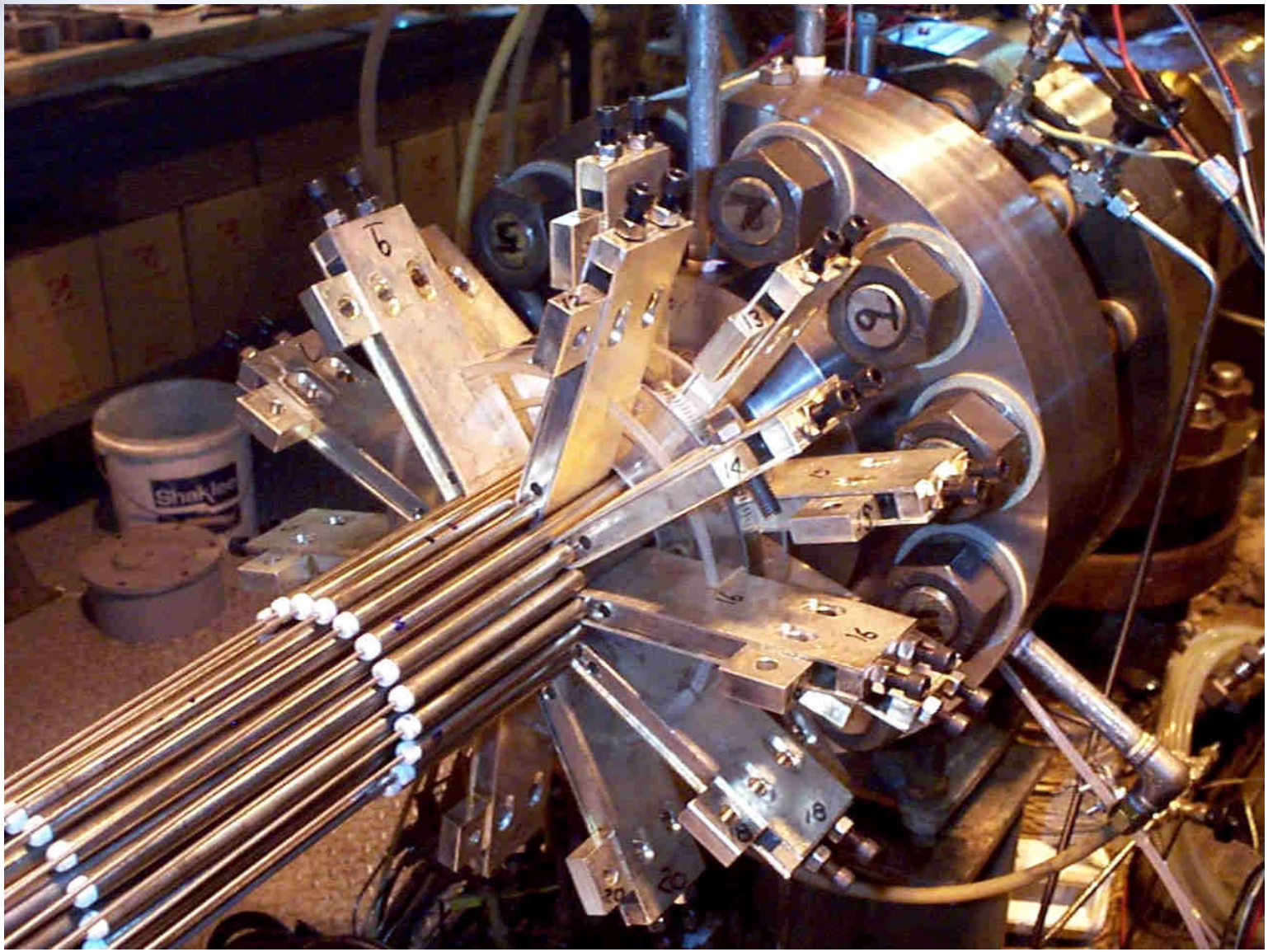
Water CHF Program

- Objectives:
  - produce thermal hydraulic data required for licensing CANFLEX fuel in CANDU reactors
  - secure regulatory approval of the enhanced CHF performance over the 37-element fuel
- CHF testing of CANFLEX Mk 4 in 0%, 3.3% and 5.1% crept channels
- CHF correlations prepared for the NUCIRC and CATHENA codes
- Bundle dryout power increase up to 17%, and CCP enhancement up to 8% demonstrated
- Report implemented for use in licensing

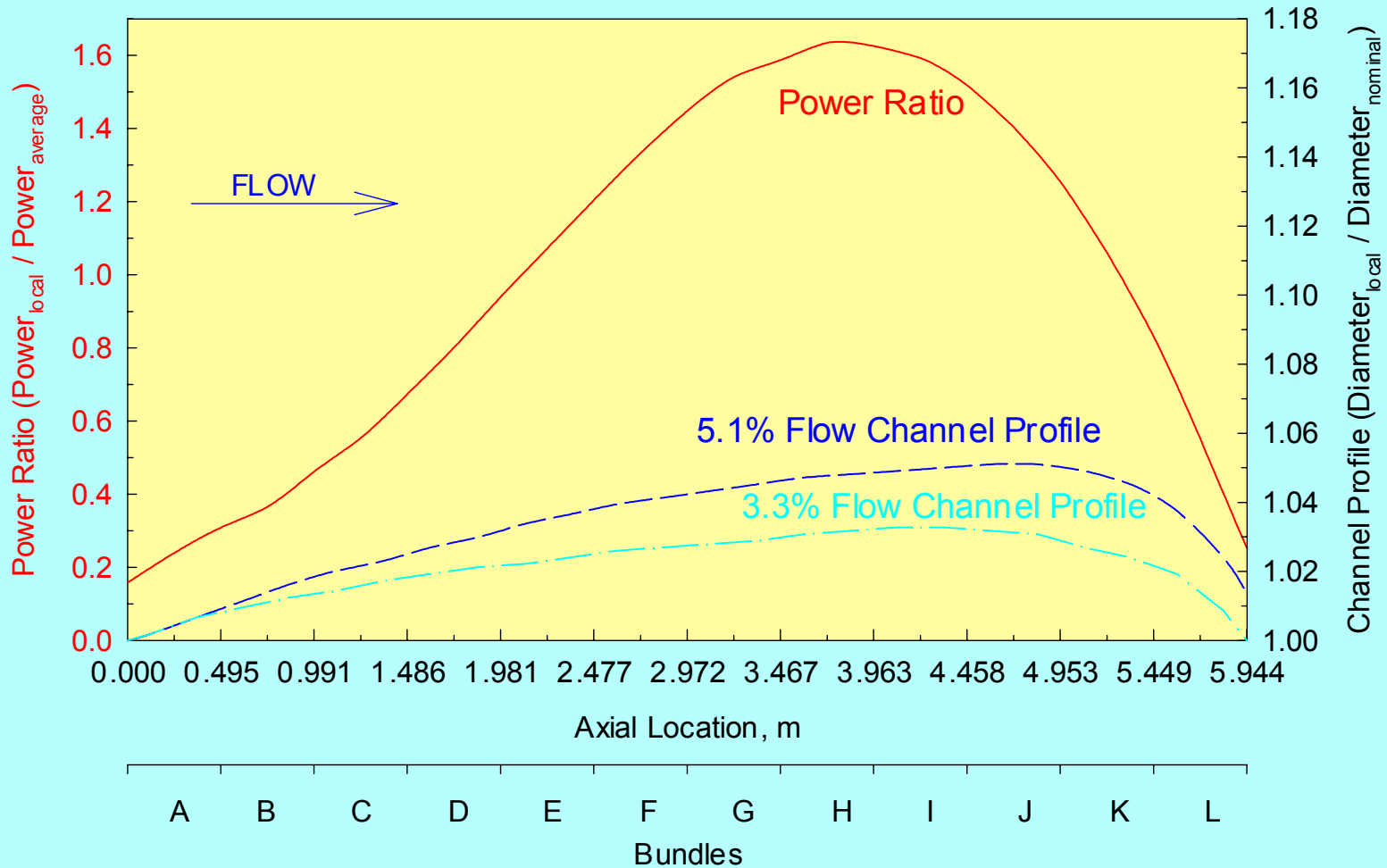
Water CHF Test Station at Stern Labs



Power Connection for Water CHF Test



Thermal Hydraulic Testing



CANFLEX Mk 4 NU Water Test Matrix

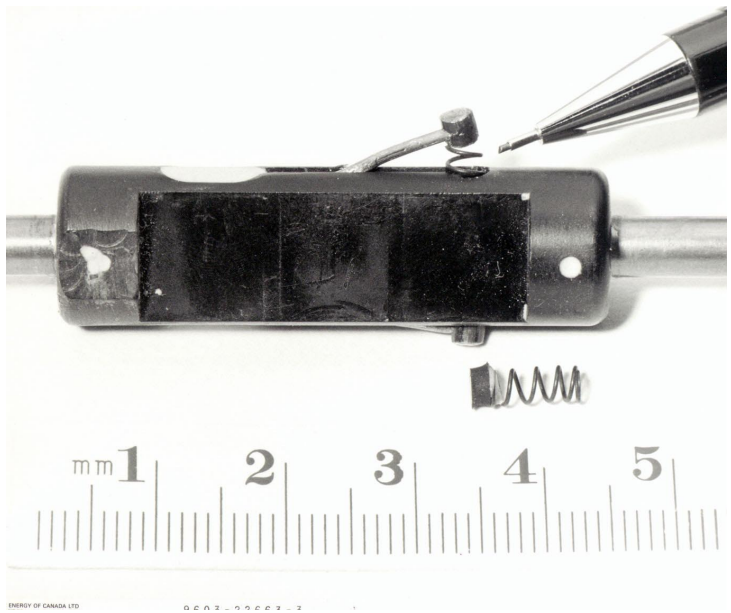
SL CANFLEX water CHF test matrix.

	Uncrept flow tube									3.3 % crept flowtube							5.1% crept flowtube								
Flow kg/s	7	10	14	17	19	21	23	25	27	7	10	14	17	19	21	23	10	14	17	19	21	23	25	27	29
6 MPa																									
265°C		x	x,P	x							x	x	x	x			x	Px	x	x					
245		x	x	x	x						x	x	x	x			x	x	x	x					
225		x	x	x	x					x	x	x	x	x			xx	x	x	x					
212																									
200		x	x	x						x	x	x	x	x			x	x	x	x					
7.5 MPa																									
265°C		x		x		x					x		x		x										
9 MPa																									
284°C			x	x	x	x						x		x		x			xxx		x	x			
273			x		x						x	x	xx	x	x	x		xxx	xxx	xx	xxx	xx	xxxx		
263		xx	xxx	xxx	xxx	xxx	xx				x	x	x	x	x			xP	xxP	x	x	x			
253		x	xxx	xx	x	x					x	x	x	xx	x	x		xx	xx	xx	xx	x	xxxx		
240		x	x	x	x						x	x	x	xx	x			xxx	xxx	xxx	xx	xx			
228			xxx	x	x	x					x		xx		x			x		x		x			
10 MPa																									
265°C		x		x		x					x		x		x										
10.5 MPa																									
265°C		x	x	x	x	x					x		x		xx			xxx	x	xx			x		
11 MPa																									
290°C	x	x	x	xx		xxx	xx					x	x		x		x		x		x				
280		x	x	x	x	x	xx	x			x	x	x	x	x	x		xx	xxx	xx	x	x	xxx	x	
268		xx	x	xx	x	xxx	xx	xx	x		x	x	xx	x	xx	x	Px	Px	Px	xxx	x	x	xxx	x	x
255		x	x	x	xxx	xxx	xx	x			x	x	x	x	x	x		x	xx	x	x	xxx	xxx	x	
249																									
243		xxxx	xxx	xxx	xxx	xx	xxx				x	x	x	x	x	x	xx	x	xxx	xx	x	x	x		

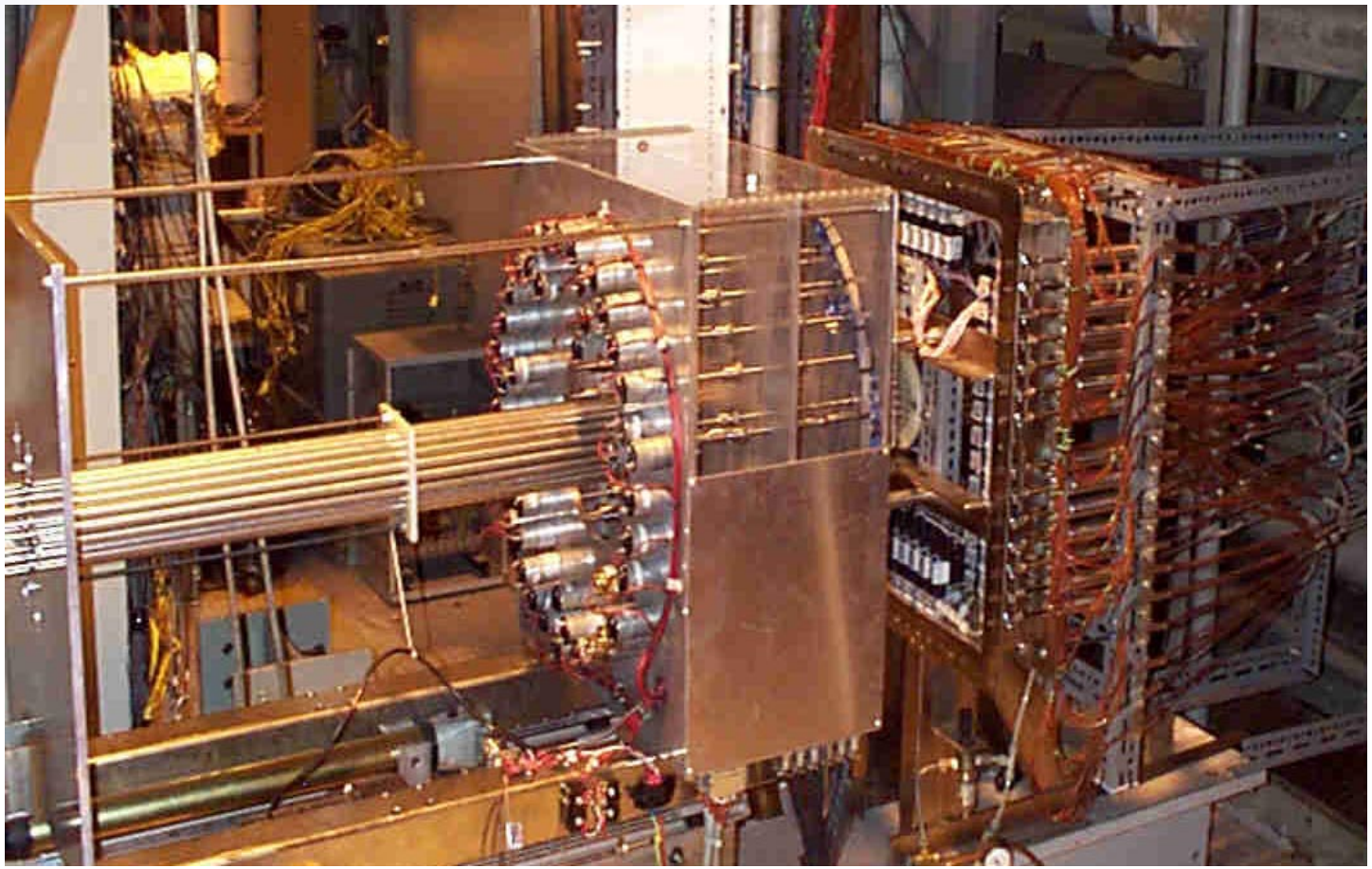
x CHF runs completed. Blue background are 37 element CHF tests.

Sliding Thermocouples

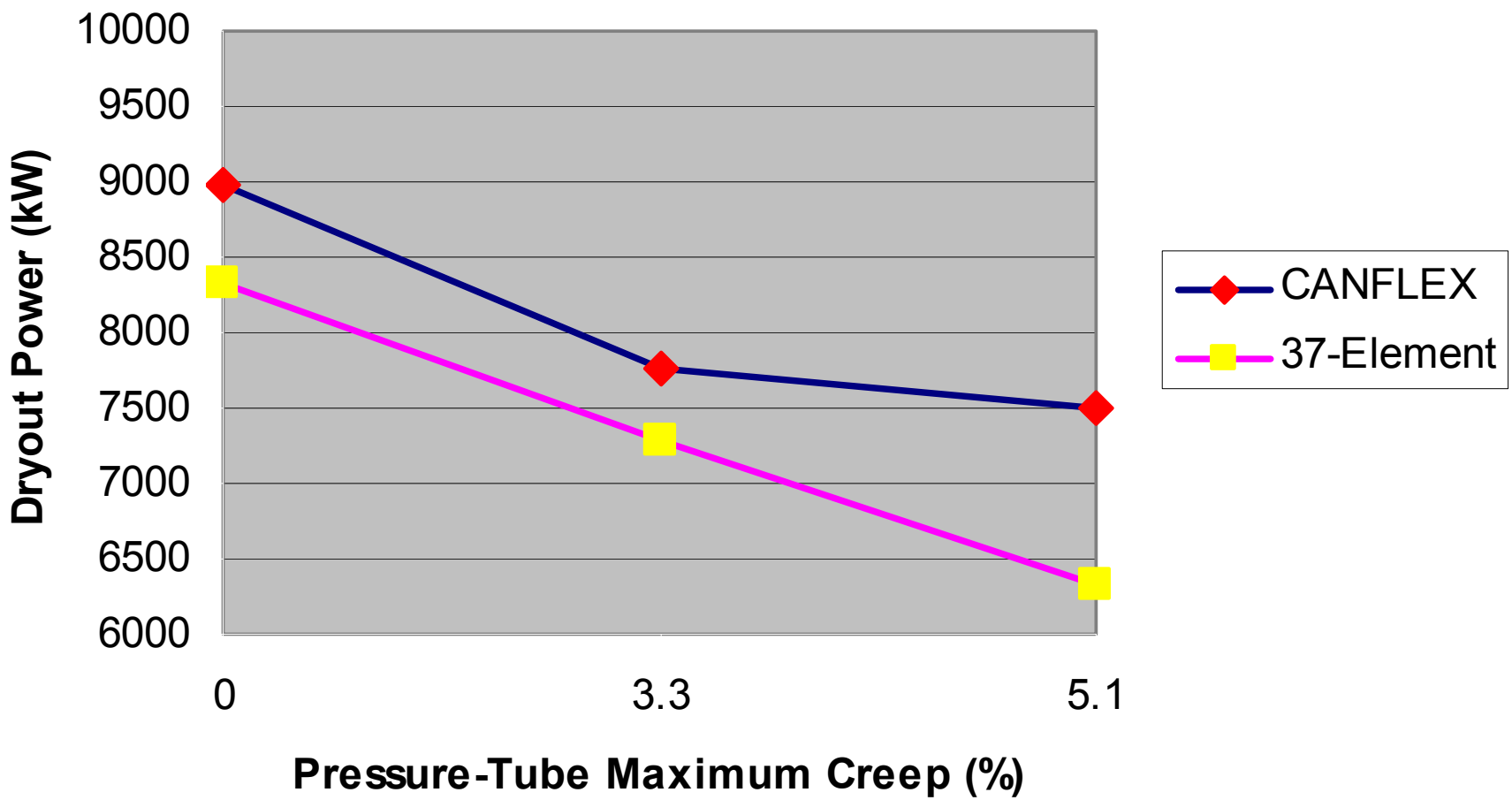
- Sliding thermocouple assemblies for dryout detection and fuel clad temperature measurements
  - cover almost the entire fuel clad area
  - detects initial and subsequent dryout locations
  - allows 3-D representation of clad temperature



Sliding Thermocouple Drive Unit



CANFLEX Thermal Hydraulic Performance

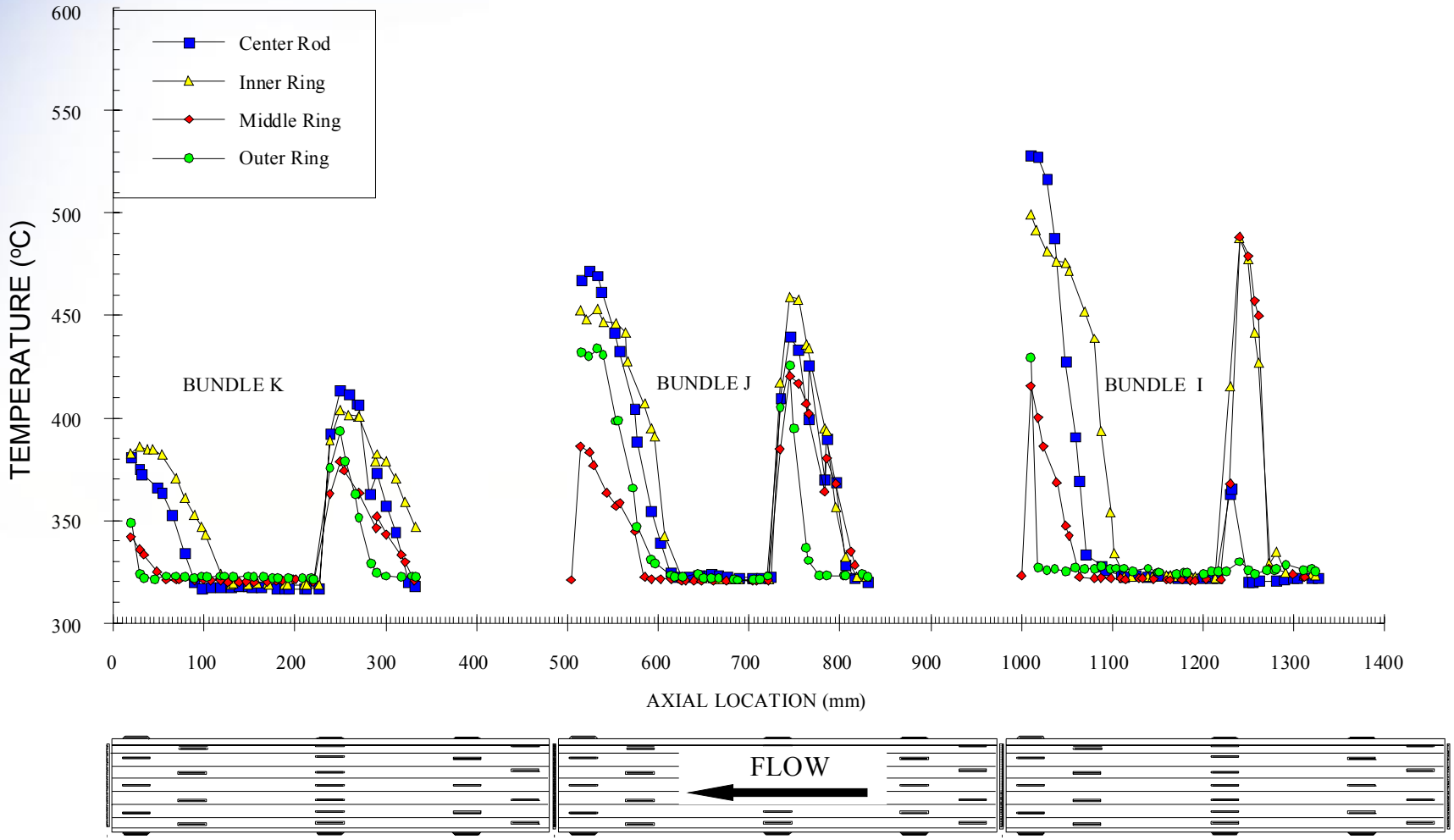


Post Dryout (PDO)

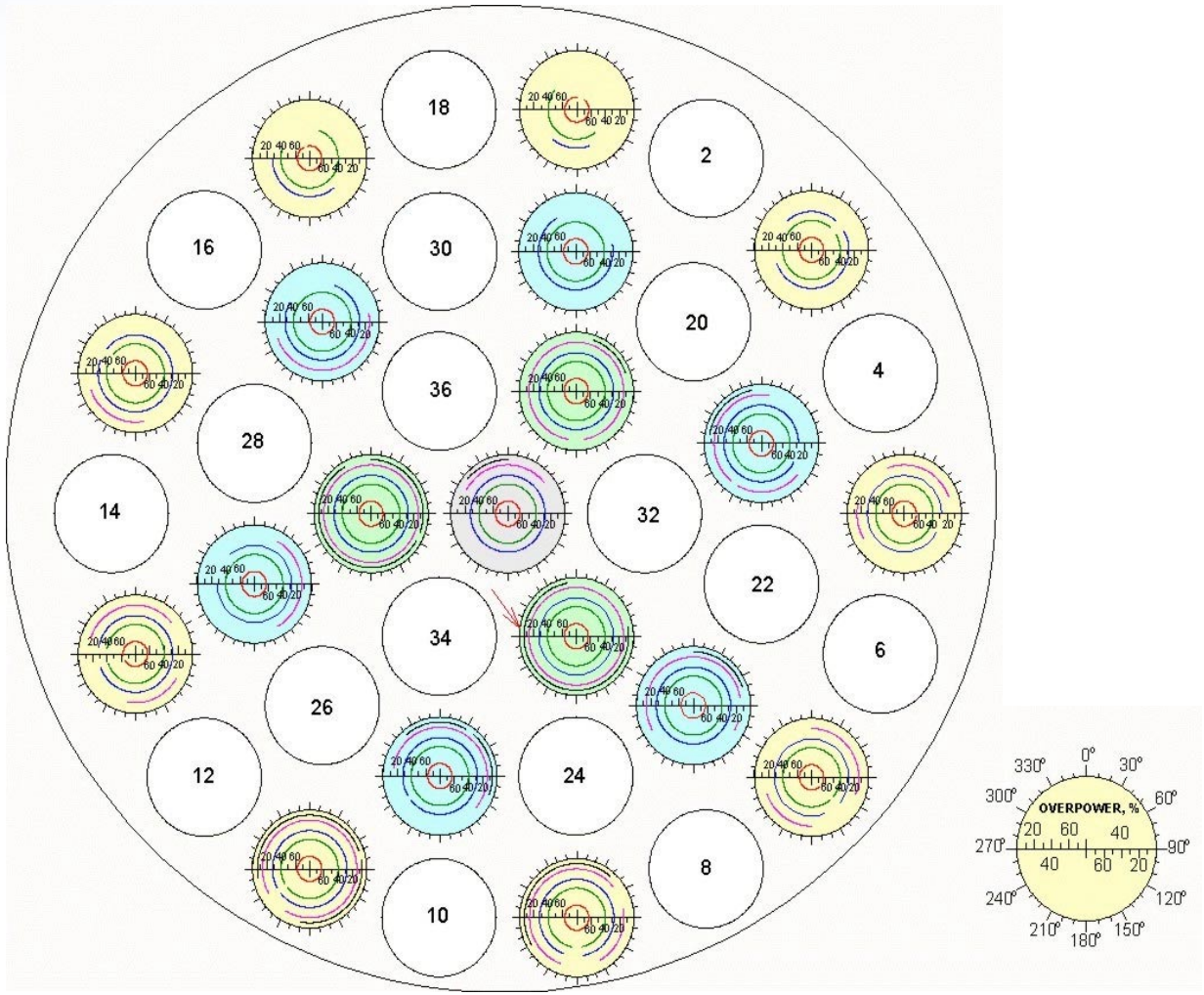
- Heat-transfer regime where the heated surface is cooled by the vapor flow

- At PDO conditions
  - fuel clad temperature rise is gradual and controllable
  - maximum clad temperature is predictable
  - drypatches are stable and propagate gradually with change in flow conditions

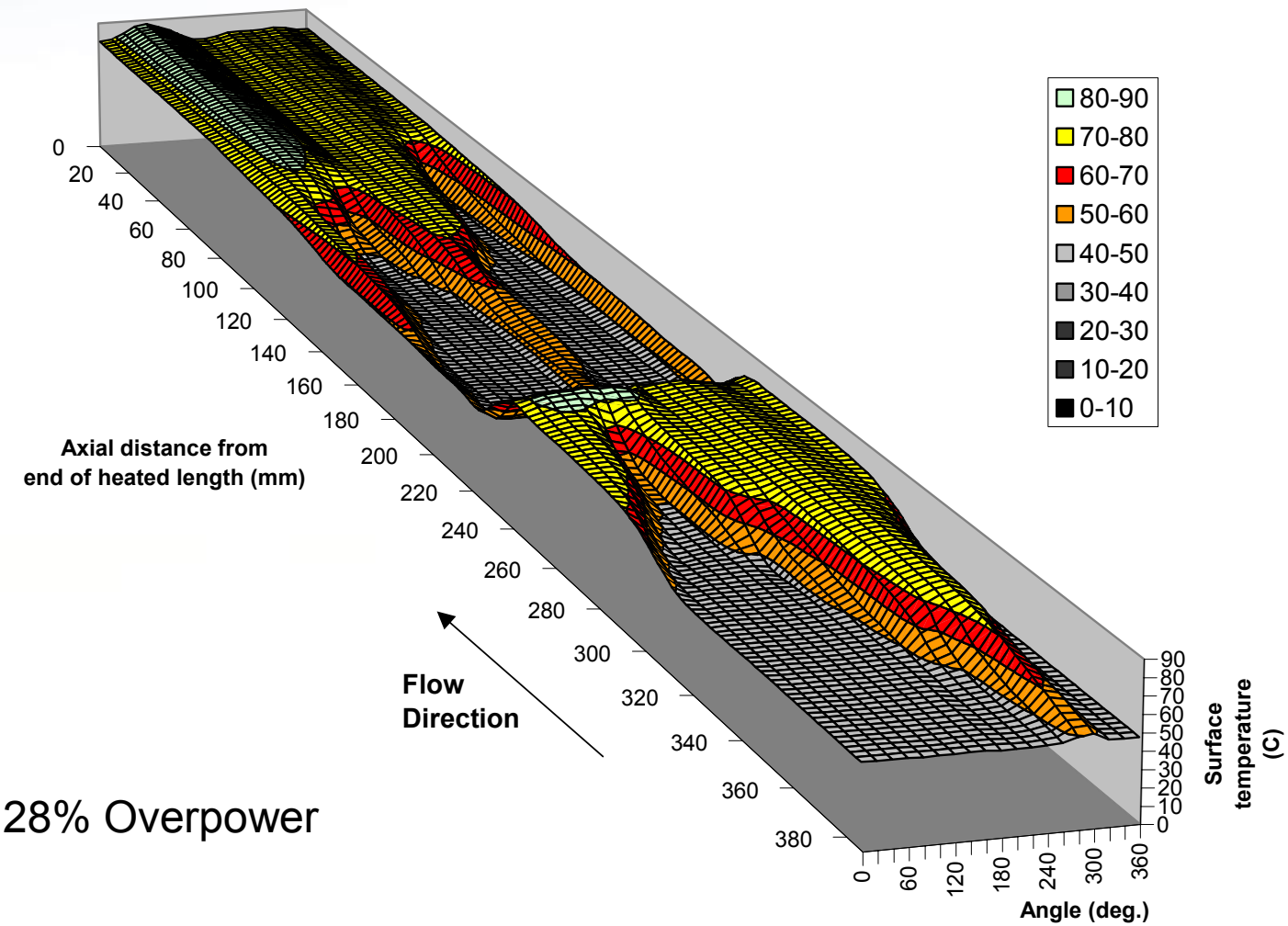
PDO Temperature Distribution in 37-element Bundles



Circumferential Drypatch Map



PDO Clad Temperature Profile



Element 35, 28% Overpower

CANFLEX Demonstration Irradiation (DI)

- In 2 channels in the Point Lepreau Generating Station (PLGS)
  - a high-power and low-power, instrumented channel
- All on-power refueling with CANFLEX was normal
- 24 discharged bundles were inspected visually and in normal condition for irradiated fuel
- Two bundles were examined in the hot cells at Chalk River and all evidence showed excellent fuel performance
- As a result of DI minor changes were made to the CANFLEX design drawing to tightening dimensions on appendages

High Power Channel S08 Fueling Scheme

Prior to first CANFLEX fuelling

1	2	3	4	5	6	7	8	9	10	11	12

After first 8 bundle CANFLEX fuelling (Sept 98)

1	2	3	4	5	6	7	8	9	10	11	12

After second 8 bundle CANFLEX fuelling (Aug 99)

1	2	3	4	5	6	7	8	9	10	11	12

First 4 CANFLEX bundles into bay

After first 8 bundle 37-element bundle fuelling (Feb 00)

1	2	3	4	5	6	7	8	9	10	11	12

8 CANFLEX bundles into bay

After second 8 bundle 37-element fuelling (Aug 00)

1	2	3	4	5	6	7	8	9	10	11	12

Last 4 CANFLEX bundles into bay

#	37-element bundle
#	1st fuelling of 8 CANFLEX bundles
#	2nd fuelling of 8 CANFLEX bundles

A total of 16 CANFLEX bundles fuelled into the high power channel

Low Power Channel Q20 Fueling Scheme

Prior to first CANFLEX fuelling

1	2	3	4	5	6	7	8	9	10	11	12

After first 8 bundle CANFLEX fuelling (Sept 98)

1	2	3	4	5	6	7	8	9	10	11	12

After first 8 bundle 37-element bundle fuelling (Mar 99)

1	2	3	4	5	6	7	8	9	10	11	12

First 4 CANFLEX bundles into bay

After second 8 bundle 37-element fuelling (Jan 00)

1	2	3	4	5	6	7	8	9	10	11	12

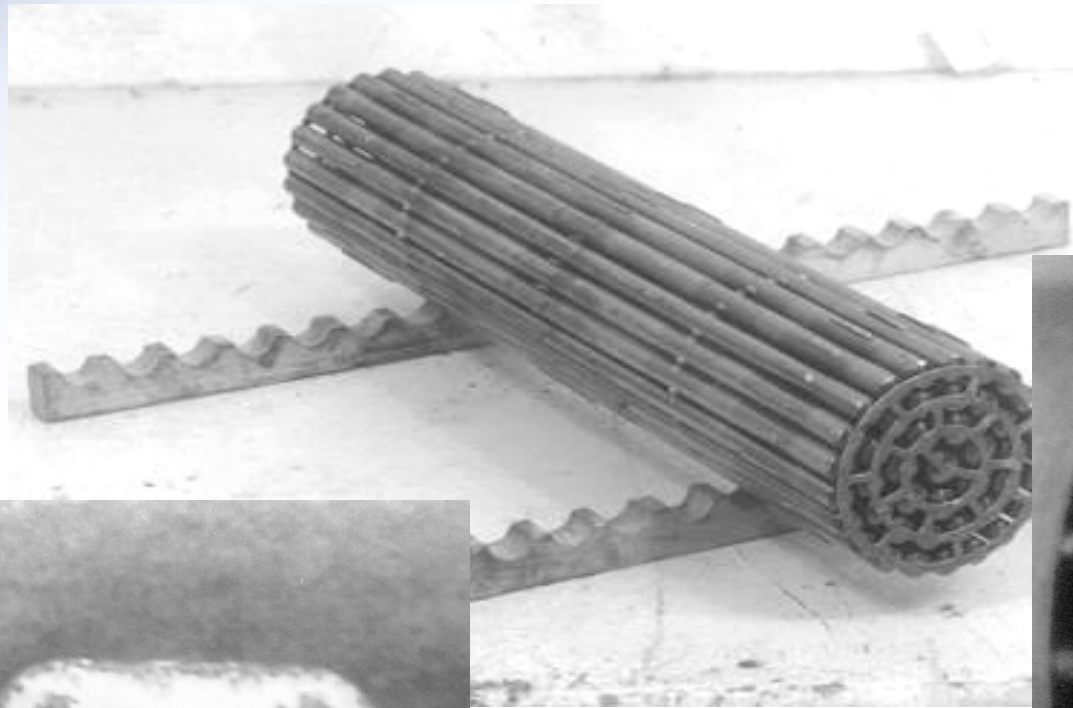
Last 4 CANFLEX bundles into bay

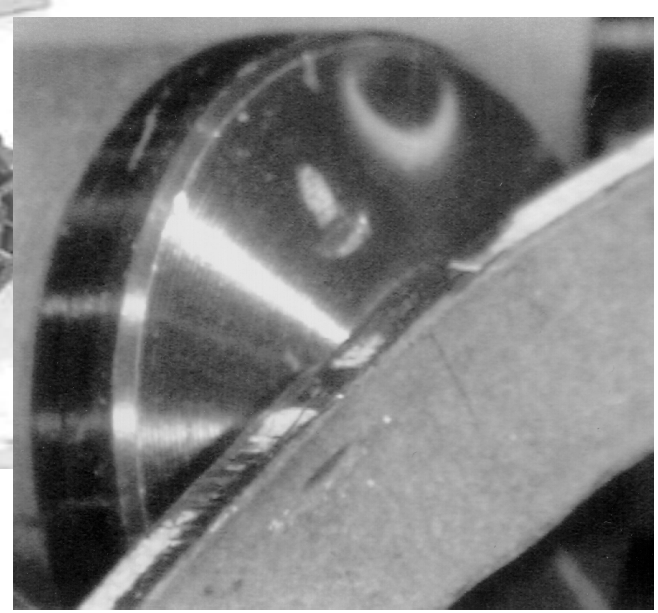
#	37-element bundle
#	1st fuelling of 8 CANFLEX bundles

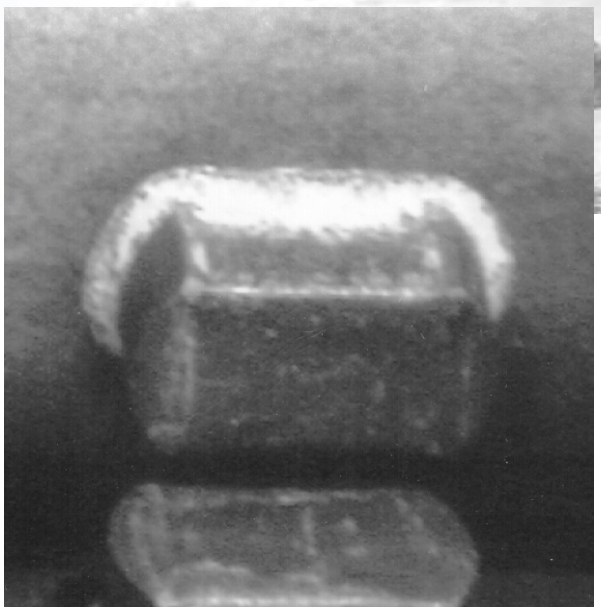
A total of 8 CANFLEX bundles fuelled into the low power channel

Loading CANFLEX at PLGS Fuel Room

PIE of CANFLEX Bundle from PLGS

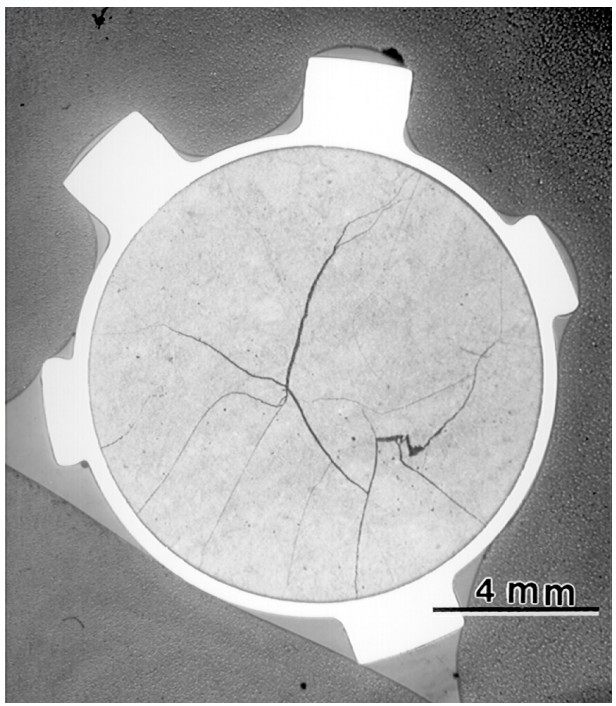




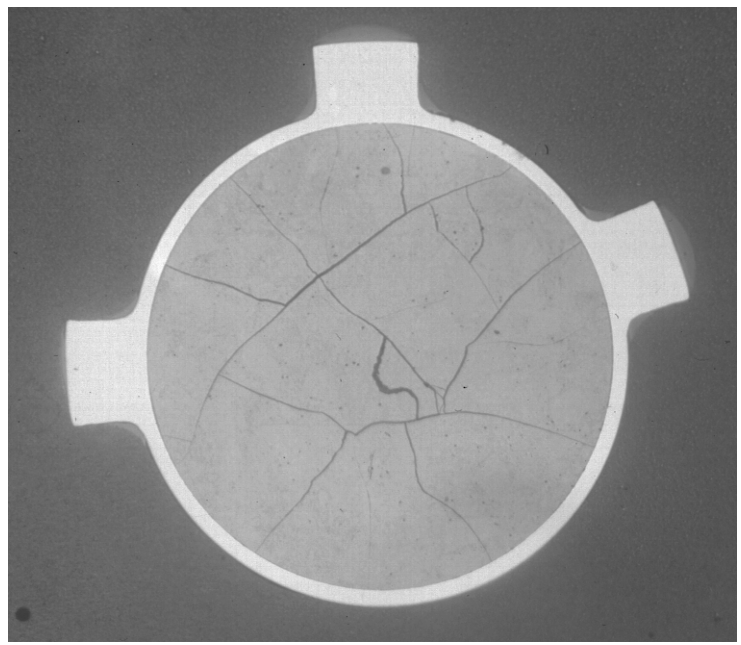


Element Fuel Microstructure Profile

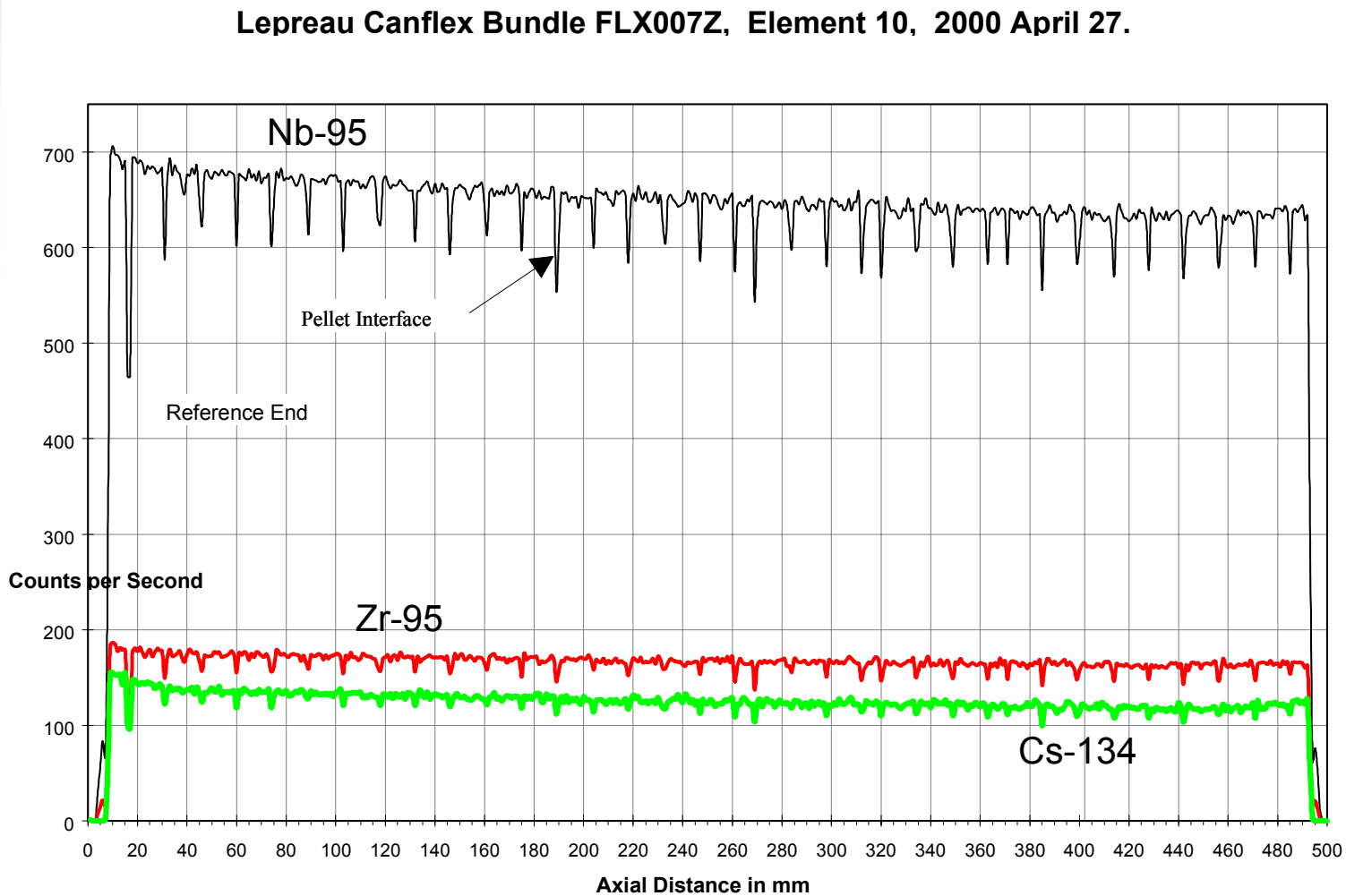
FLX0019Z



FLX007Z



Typical CANFLEX Gamma Scan Results



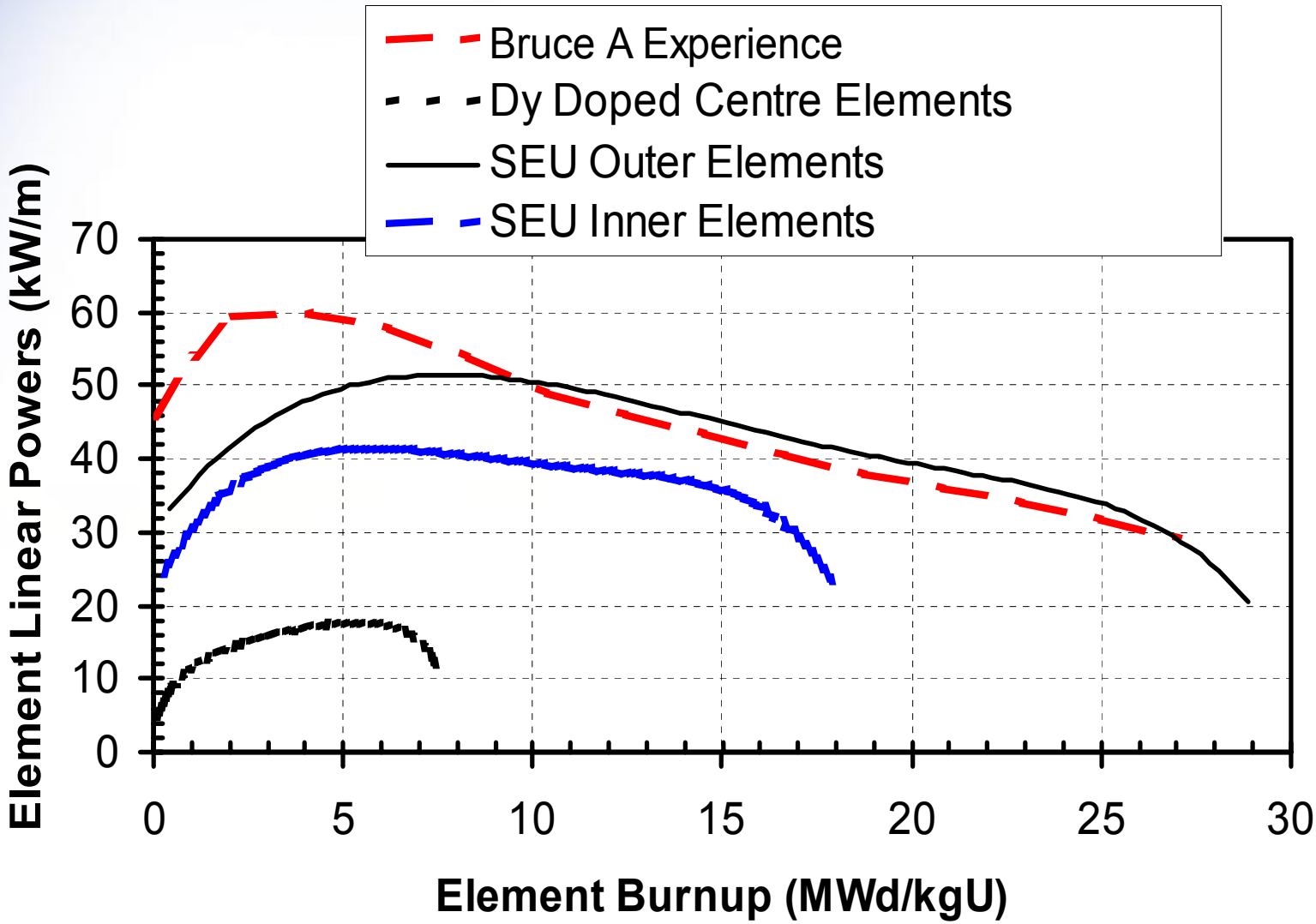
Summary for CANFLEX Mk 4 NU

- Fuel design verification process completed in accordance with CAN/CSA-N286.2 to meet the interface requirements of existing CANDU 6 stations
- CANFLEX is ready for full commercial implementation
- Conversion to full core of CANFLEX NU in Canada will require regulatory approval
  - proponent to demonstrate that change in fuel does not compromise safe operation of reactor, based on existing safety report and supporting documentation
  - must also consider transition between an all-37-element-bundle core, and an all-CANFLEX core
  - AECL will work with CANDU utilities in Canada to establish the licensing program requirements and the various roles and responsibilities

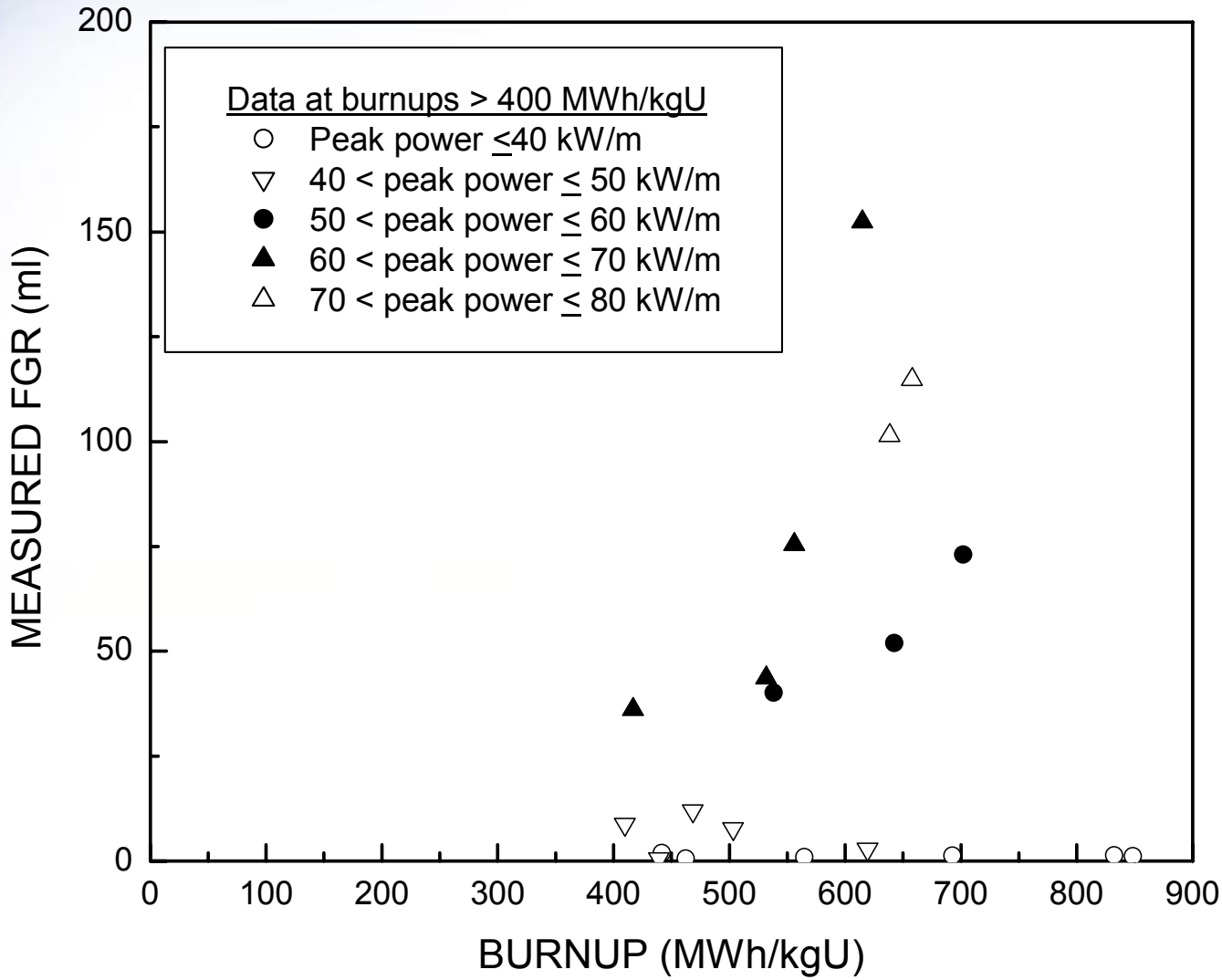
Extended Burnup Irradiation Experience

- Power reactor experience
  - >230 37-element bundles achieved burnups > 17 MWd/kg in Bruce A
- Research reactor experience
  - >24 bundle and element irradiations in NRU > 17 MWd/kg
  - 15 irradiations with burnups greater than 21 MWd/kg
  - 10 of 24 irradiations also experienced power ramps
  - several irradiations ongoing
- Qualified irradiated fuel databases
  - 28, 37-element and CANFLEX
- Good confidence in ACR fuel performance based on our experience
  - ACR power envelope is below the high power envelope for which we have experience
  - ACR fuel pellet design is optimized for extended burnup, based on our experience base and assessments

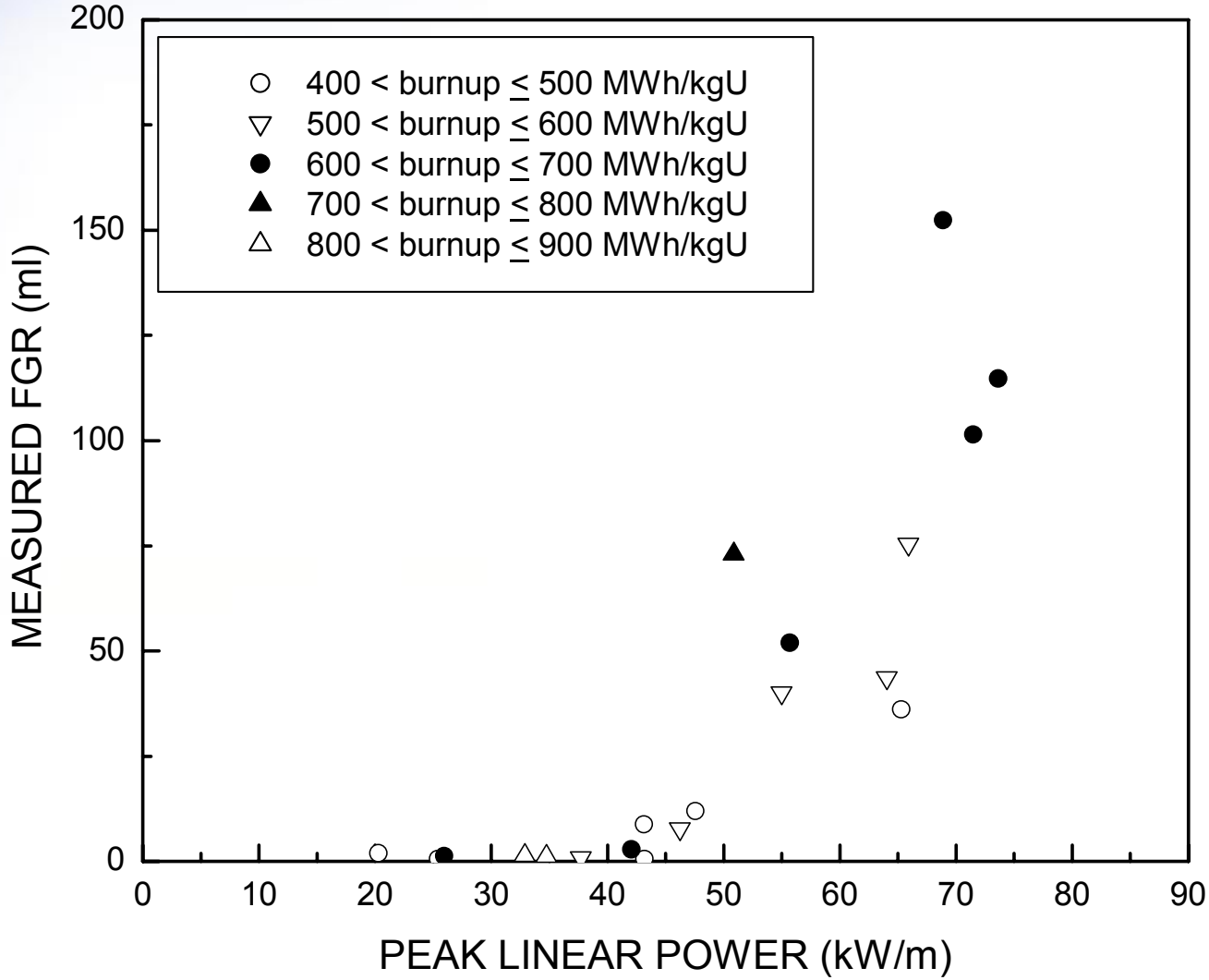
ACR Power Envelope vs. Bruce A Experience



Measured Fission Gas Release vs. Burnup

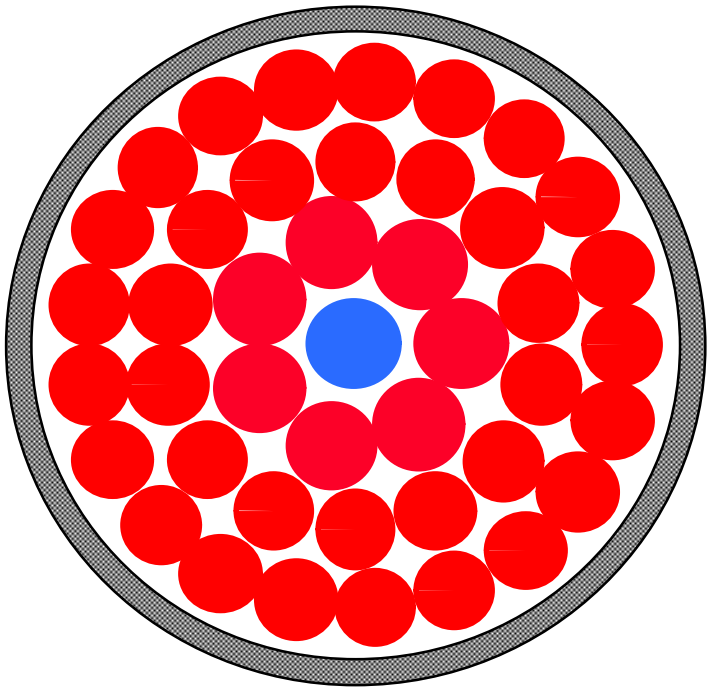


Measured Fission Gas Release vs. Element Rating



LVRF Concept

- CANFLEX (or 37-element bundle)
- Dy_2O_3 (neutron absorber) in central element(s), mixed with NU
- Enrichment in outer elements
- Dy content, and enrichment can be independently varied to give desired value of void reactivity reduction and burnup



Dy_2O_3 in NU

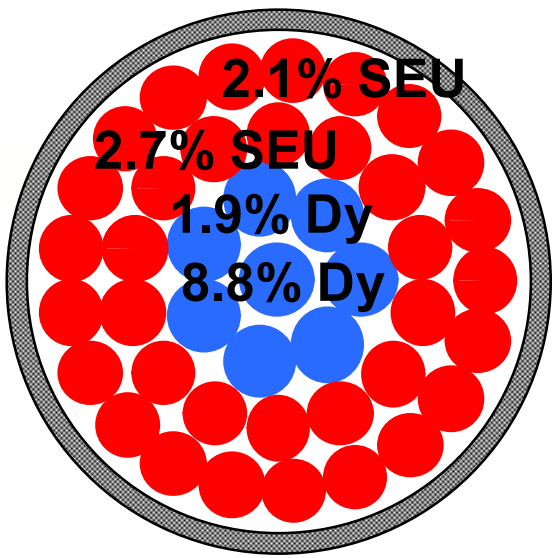
SEU/RU

Overview of AECL Studies

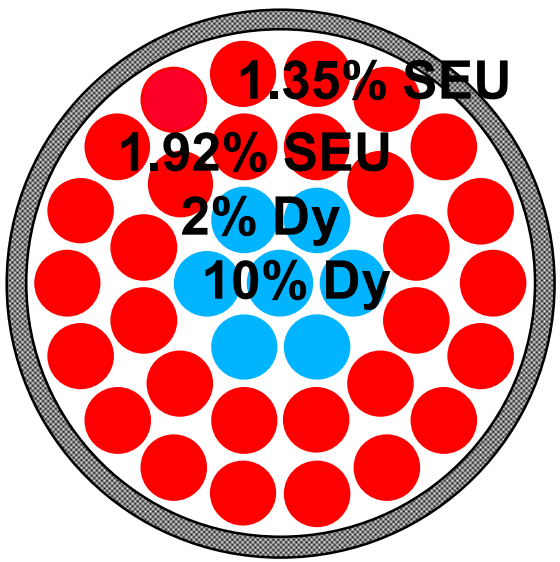
- Over past decade, AECL has undertaken many studies on fuel options for reducing void reactivity in CANDU
- Negative void reactivity fuel (NVRF)
  - 37-element NVRF, NU burnup
  - CANFLEX NVRF, 3x NU burnup
  - developed as "insurance" for international CANDU markets
  - extensive testing done
  - provides high confidence for ACR
  - MOX fuel for Pu-dispositioning application
- Low void reactivity fuel (LVRF)
  - developed as "insurance" for domestic markets
  - basis for current qualification program for Bruce Power

Negative Void Reactivity Fuel (NVRF)

- Considered limiting case for void reactivity reduction in current reactors
- Bundle designs chosen had negative void reactivity at mid-burnup for current reactors
- Dy mixed with DU in central elements, graded enrichment
- 37-element: NU discharge burnup; CANFLEX: 3x NU burnup



CANFLEX NVRF



37-element NVRF

Overview of Generic NVRF Testing

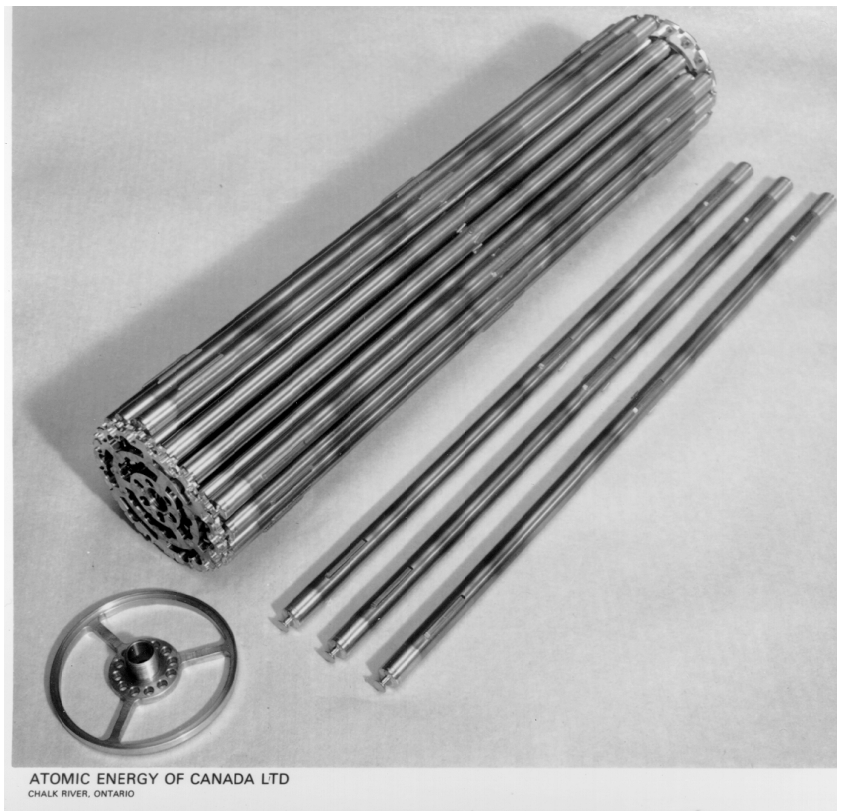
- Dy_2O_3 -UO_2 Pellet Fabrication
  - generic
- Bundle Fabrication
- Irradiation Testing and PIE
  - Dy demountable elements
  - prototype bundles
- Reactor Physics
  - ZED-2 measurements
    - void reactivity
    - fine structure
  - WIMS validation
- Thermalhydraulics
  - measurements, modelling
- High temperature interactions
  - interactions with Zircaloy
  - grain-boundary inventory

Dy-UO_2 Pellet Fabrication

- Investigated methods of blending Dy_2O_3 & UO_2
- Developed process capable of
  - high production rate in laboratory
  - scaleable to production levels
  - uniform (relatively) microstructure
- Produced demountable elements to evaluate Dy-poisoned fuel under CANDU conditions
  - Dy levels of 1, 2, 5, 10 and 15%, mixed with either depleted or NU (to look at effect of power level under irradiation)

Dy Elements Irradiation Testing and Pie

- Demountable element bundle irradiation in NRU
- 21 elements fabricated and irradiated (bundle holds 18 at a time)
- Discharged over a range of burnups
- PIE summary:
  - no changes in microstructure as a result of the irradiation
  - low fission gas release, typical of that in UO_2 under similar power histories



Summary of $(Dy,U)O_2$ Experience

Protected-Proprietary

DME-214

DME-214

BDL-436

Thermal/Physical Property Measurements of $(U,Dy)O_2$

Thermal/Physical Property Measurements of $(U,Dy)O_2$

Thermal/Physical Property Measurements of $(U,Dy)O_2$

Thermal/Physical Property Measurements of $(U,Dy)O_2$

High Temperature Interactions

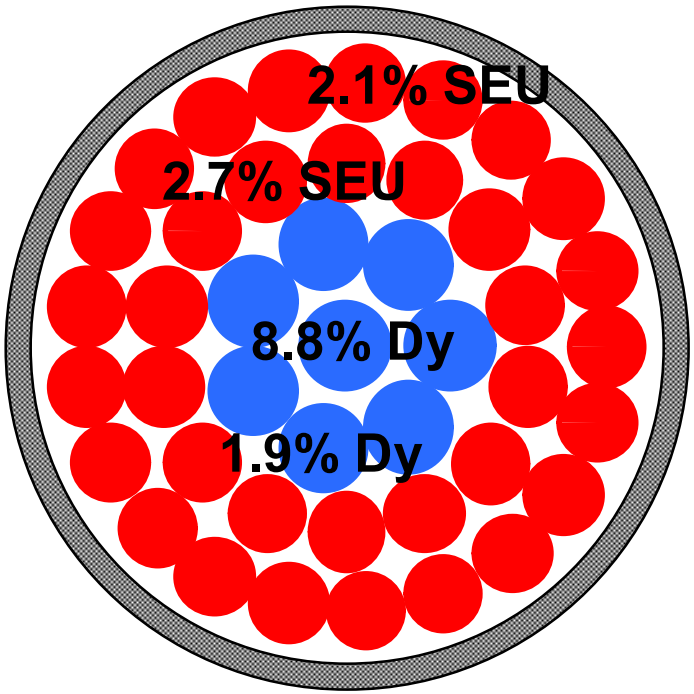
- Interactions with Zircaloy
  - UO_2 pellets containing 2% and 10% Dy_2O_3 were crushed and sieved to produce powder of size less than 45 mm
  - mixed with zirconia powder
  - submitted to DTA / TGA to 1500ºC, subsequent x-ray
  - no evidence of interaction
  - no evidence of liquid formation
- Grain Boundary Inventory
  - in accident analysis, the distribution of fission products within the element (free, grain boundary inventory, matrix) needs to be estimated
  - three elements (containing 2%, 5% Dy and 10% Dy) removed from demountable bundle after 150 full power days for assessment
  - average grain boundary inventory was consistent with prior measurements on low-powered UO_2 fuel

37-element NVRF: Bundle Fabrication, Irradiation and PIE

- Produced 35 bundles for ZED-2 physics tests
- Produced two prototype bundles for NRU irradiation
  - irradiated in positions 2 or 4 of the U1 and U2 loops of NRU between 1994 and 1996, to ~10 MWd/kg
- PIE Summary
  - overall fuel performance as expected based on NU experience
  - large flux gradient in NVRF causes larger burnup in outer elements than NU, but performance comparable to NU at these higher powers and burnups

CANFLEX Negative Void Reactivity Fuel

- 3x NU burnup
- Testing performed
  - ZED-2 measurements on single channel
  - CHF freon testing
  - 2 prototype bundles in NRU



Bruce Power CANFLEX LVRF

- AECL and Bruce Power currently qualifying CANFLEX-LVRF for specific application to Bruce Power
- Details commercially proprietary
  - session on Bruce Power CANFLEX LVRF in 2004 CNS conference
- Program will complement ACR fuel qualification
  - reactor physics substitution experiments in ZED-2 with 36 bundles (5 channels) and qualification of physics codes
  - thermal hydraulics measurements in freon
  - computer code validation (fuel, safety)

Summary: Testing Supporting LVRF

- Generic qualification for LVRF through
  - reactor physics, thermal hydraulics, fabrication development, NRU irradiation, and safety tests
- Small amount of additional work is being undertaken to qualify LVRF bundle for Bruce application
- Complementary to ACR fuel qualification

Generic Advanced Fuel Development Supporting ACR Fuel

- NRU irradiations at high power, and power ramp
  - of optimized internal element design
  - of advanced CANLUB coatings
- Improved SCC (power ramp) defect prediction capability
  - both for single power ramp, and for multiple power ramps
- Fuel chemistry studies
  - particularly at the fuel/clad interface
- Fundamental fuel properties (including Dy-doped fuel)
  - measurement of oxygen-to-uranium ratio; intrinsic fission product diffusion coefficients; thermal conductivity
- International collaboration
  - fundamental properties
  - IAEA FUMEX II
  - load-following test with Pitesti in Romania
  - CANFLEX SEU/RU collaboration with BNFL, KAERI, NASA (Argentina)

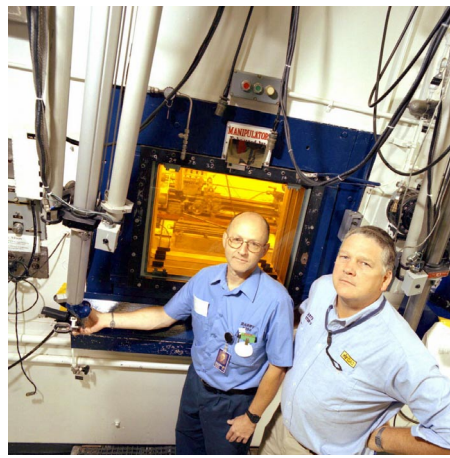
Facilities Supporting Fuel Development

Fuel Fab Lab

Thermal hydraulics Lab

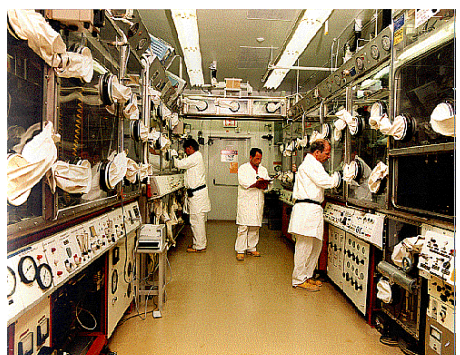
Surface Science Lab

Hot-cells

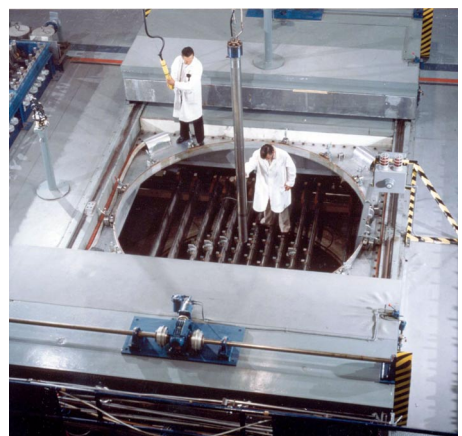


NRU

RFFL



ZED-2



Summary: Experience Base for ACR Fuel

- ACR fuel is based on 3 underlying, well established fuel technologies
  - CANFLEX
    - fully qualified, ready for commercial implementation
  - enriched uranium with extended burnup
    - extensive experience both in power reactors (Bruce A) and in NRU irradiation tests
  - Low Void Reactivity Fuel (LVRF)
    - generic qualification is applicable to ACR
    - Bruce LVRF fuel currently undergoing qualification

- AECL maintains a strong fuel development capability, encompassing fundamental studies, support for operating stations, and advanced fuels and fuel cycles
  - includes qualified staff, computer codes, and facilities

CANDU Fuel Design and Performance Codes

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Fuel Design and Performance Codes

- ELESTRES is the principal fuel design and performance code
  - will be submitted for formal certification to US NRC
- 6 other supplementary codes used for assisting design and for supporting tests for fuel qualification
  - add on-power effects to out-reactor tests
  - extrapolate to untested combinations
- These codes used for fuel design assessments for last 20 years
- Extensive validation and documentation over the years
  - experimental data
  - independent analytical results
- Rigorous configuration management
- Formally qualifying all fuel codes
  - ISO 9001-2000
  - CSA N286.7 Standard

Applications

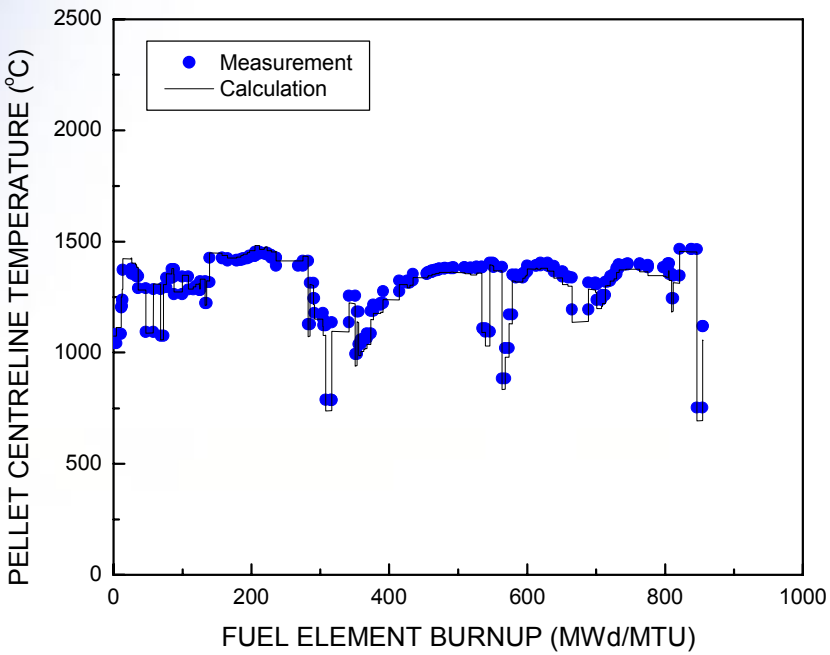
- Predict thermal & mechanical performance of fuel under normal operating conditions
  - analytical assessments of thermal integrity, structural integrity, and compatibility
- ACR fuel design, and in assisting qualification
- Predict operating values of key design parameters
  - under on-power conditions, e.g. internal gas pressure
  - account for important effects, e.g. irradiation embrittlement
- Confirm design margins
- Provide initial conditions for accident analysis codes
  - e.g. ELESTRES providing input to ELOCA

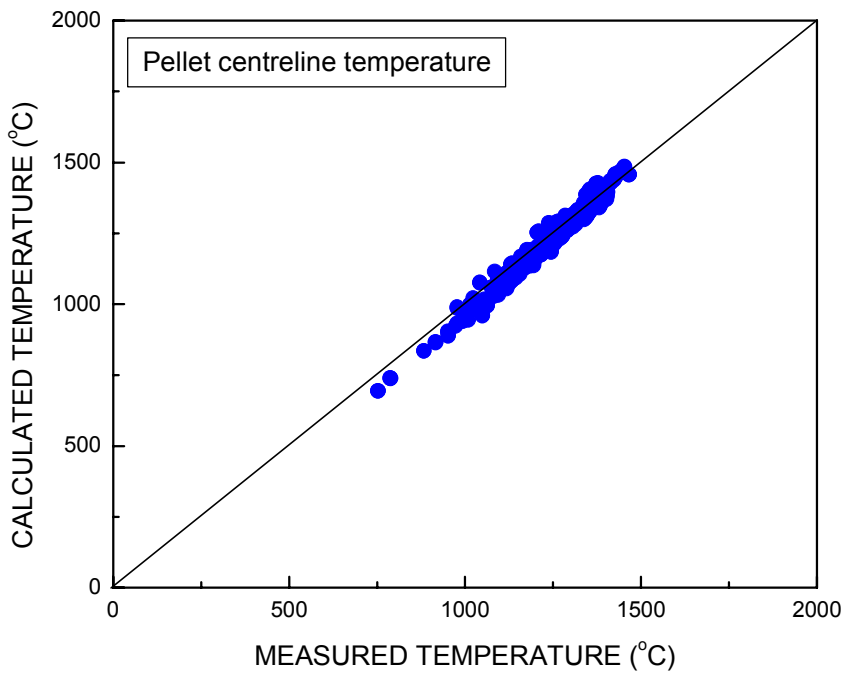
ELESTRES

- Used to calculate temperature, fission gas release, internal pressure, and clad strain including circumferential ridge
- Temperature
  - heat generation, heat conduction, flux depression, thermal conductivity, gap/contact, finite difference, 100 radial annuli
- Fission gas release
  - microstructural models, generation, diffusion, grain boundary sweep, bubble growth and interlinkage, release via tunnels
- Strain
  - densification, fission product swelling, thermal expansion, elasticity, plasticity, creep, cracking, radial and axial gaps, finite element, 2-dimensional
- Validation against ~ 130 irradiations

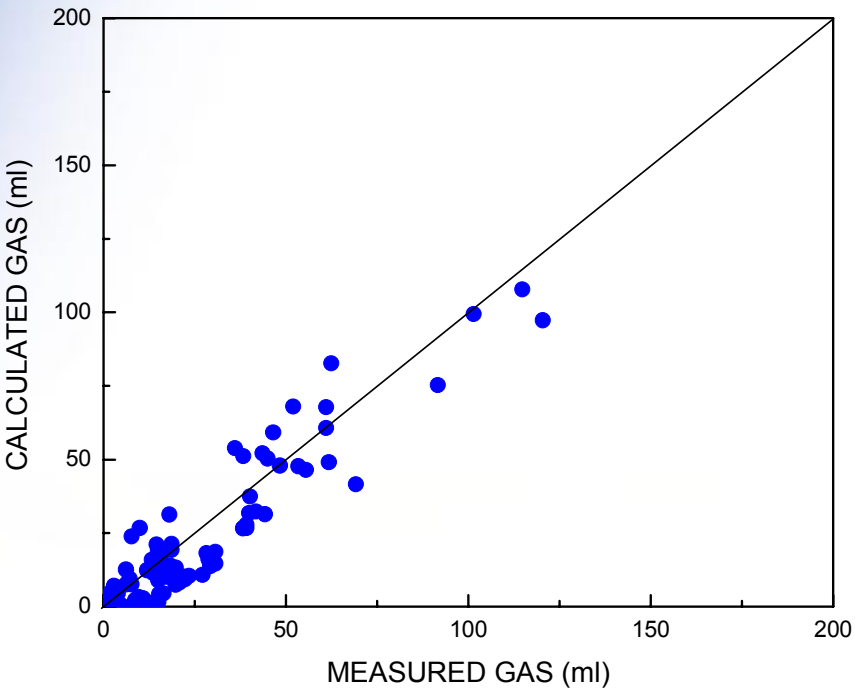
ELESTRES: Pellet Temperature

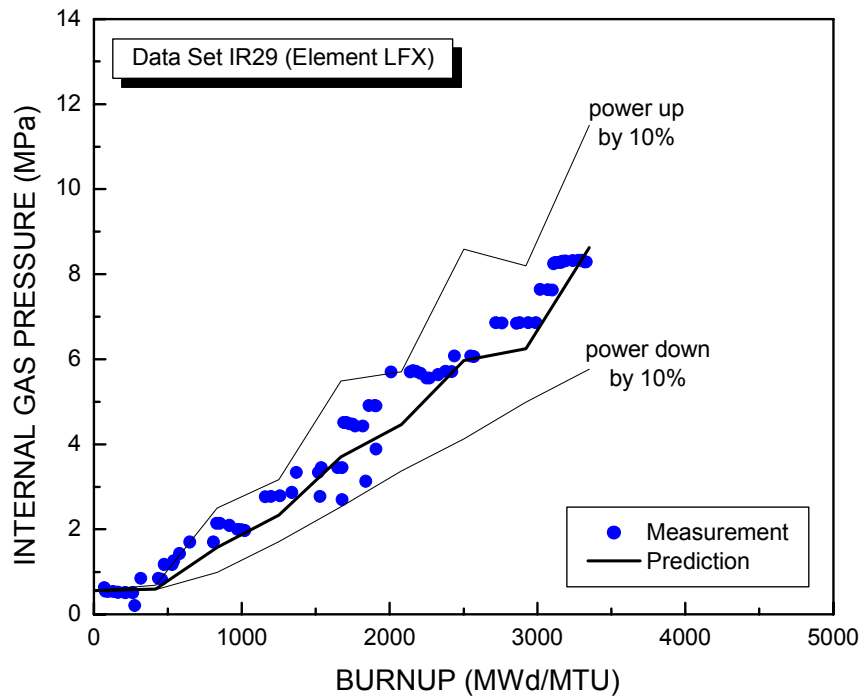
EXP-FIO-142



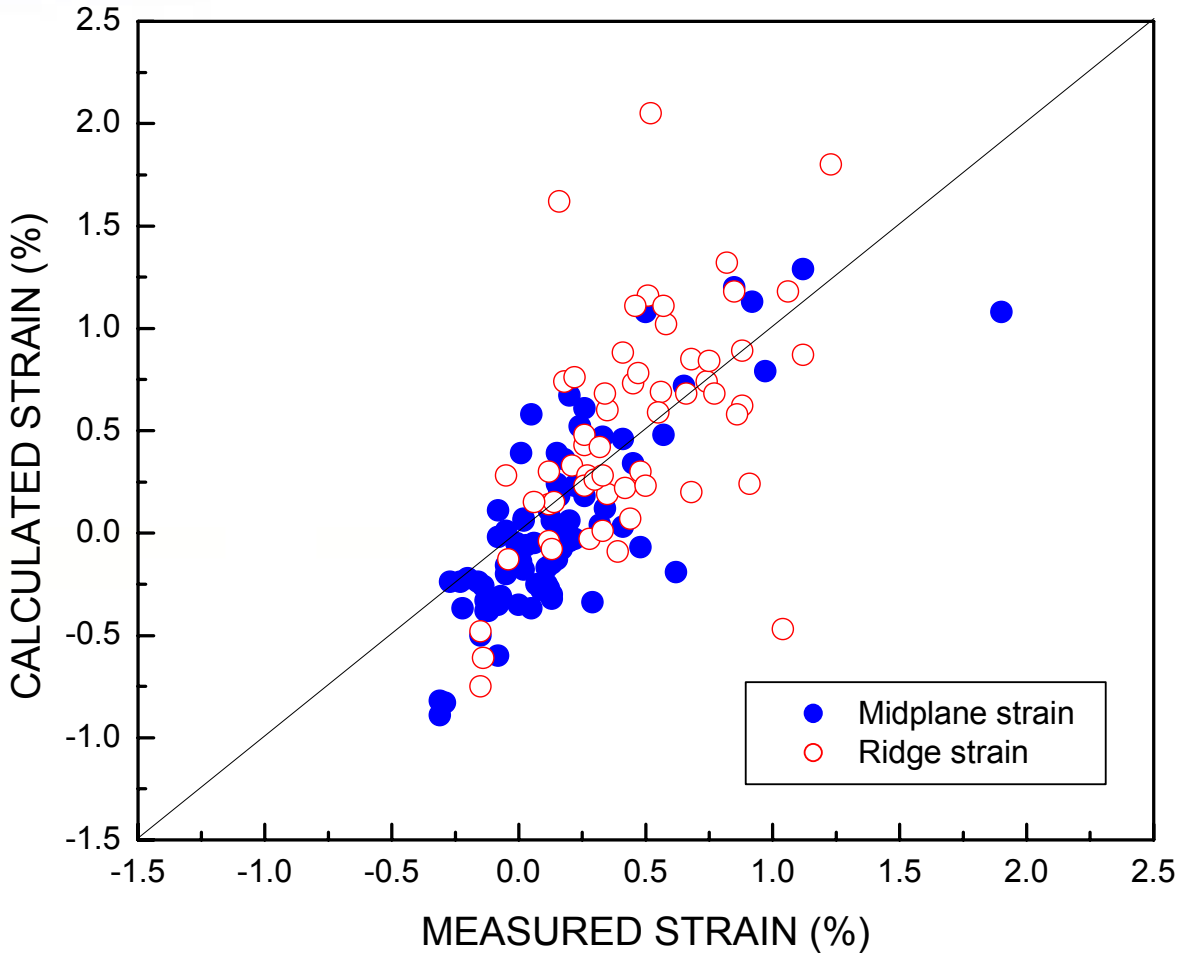


ELESTRES: Fission Gas





ELESTRES: Clad Strain

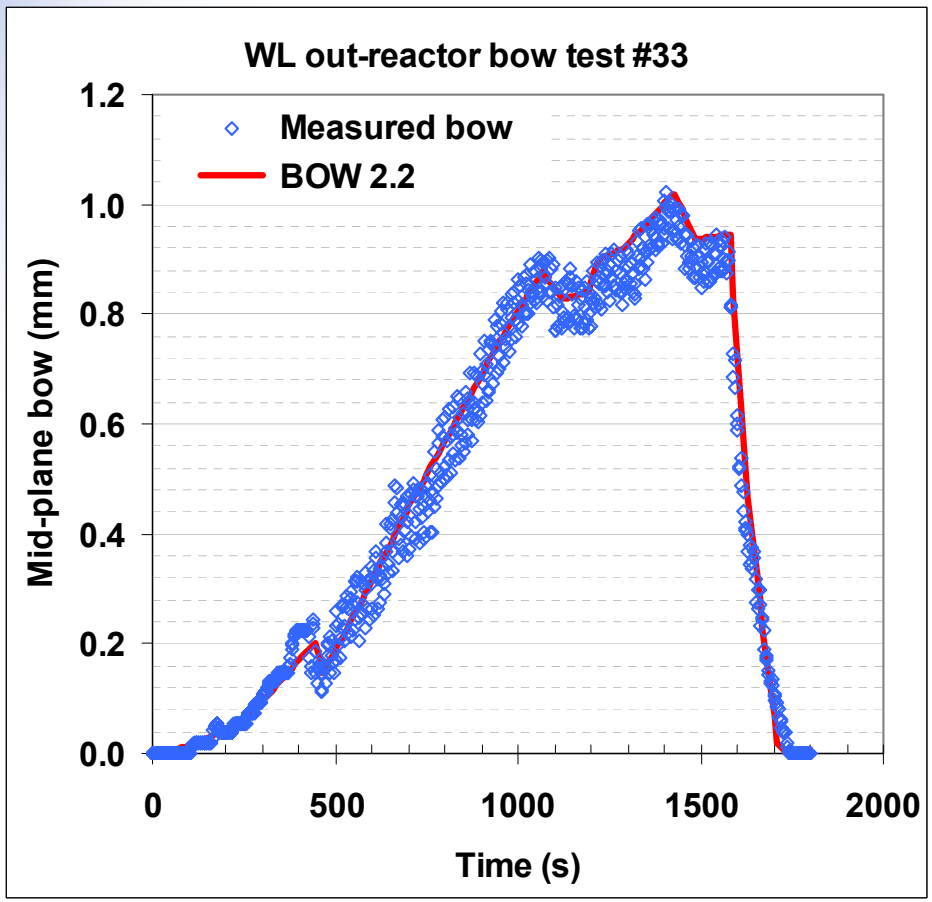


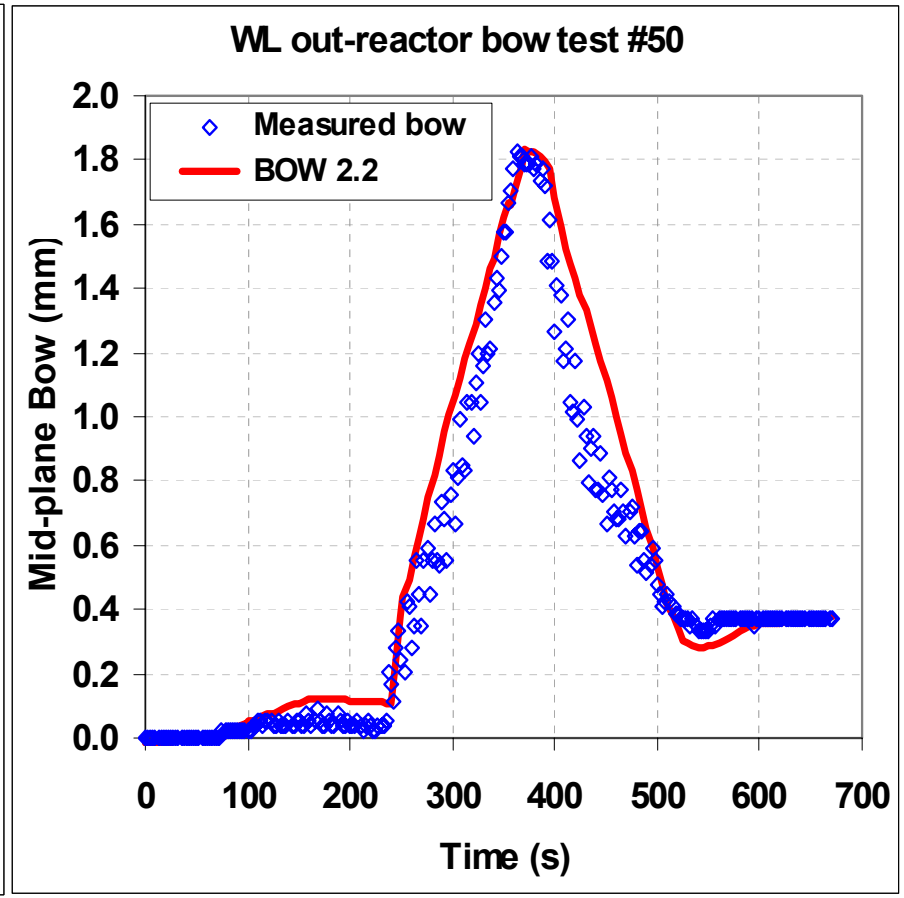
BOW

- Used to calculate deflections of fuel elements: bow, sag and droop
- Circumferential and axial temperature gradients in clad and pellets
  - neutron flux gradients, dry patches, non-uniform heat transfer coefficient, and coolant temperature
- Hydraulic drag, gravity, length differentials, creep
- Endplates, pellets (including cracking), appendages, neighboring fuel elements and pressure tubes
- As-fabricated bow, variations in material properties, etc.
- Finite element method
- Validation against 5 experimental measurements plus 46 independent analytical solutions (generally within 1%)

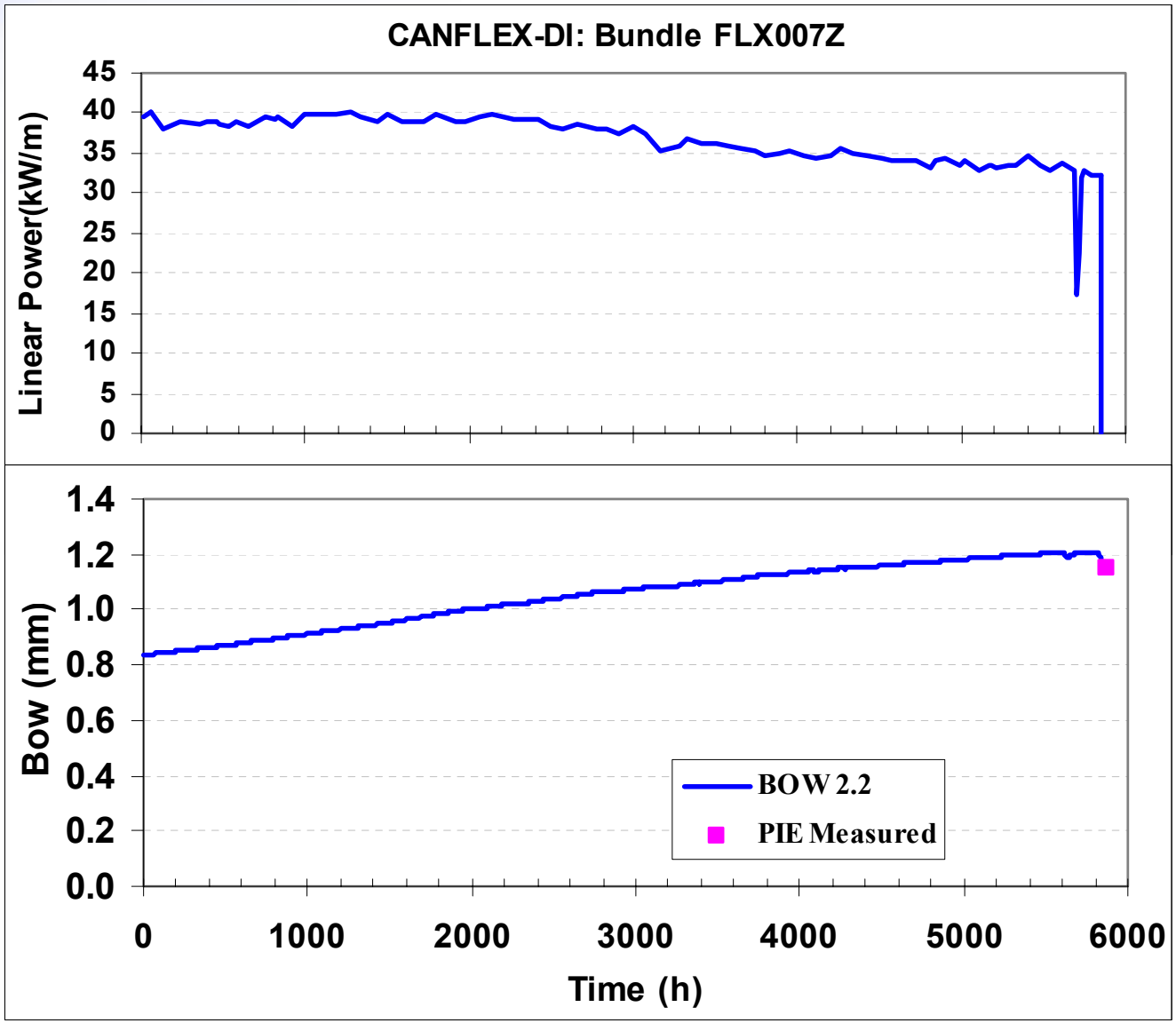
BOW vs. WL Tests

Eccentric heaters





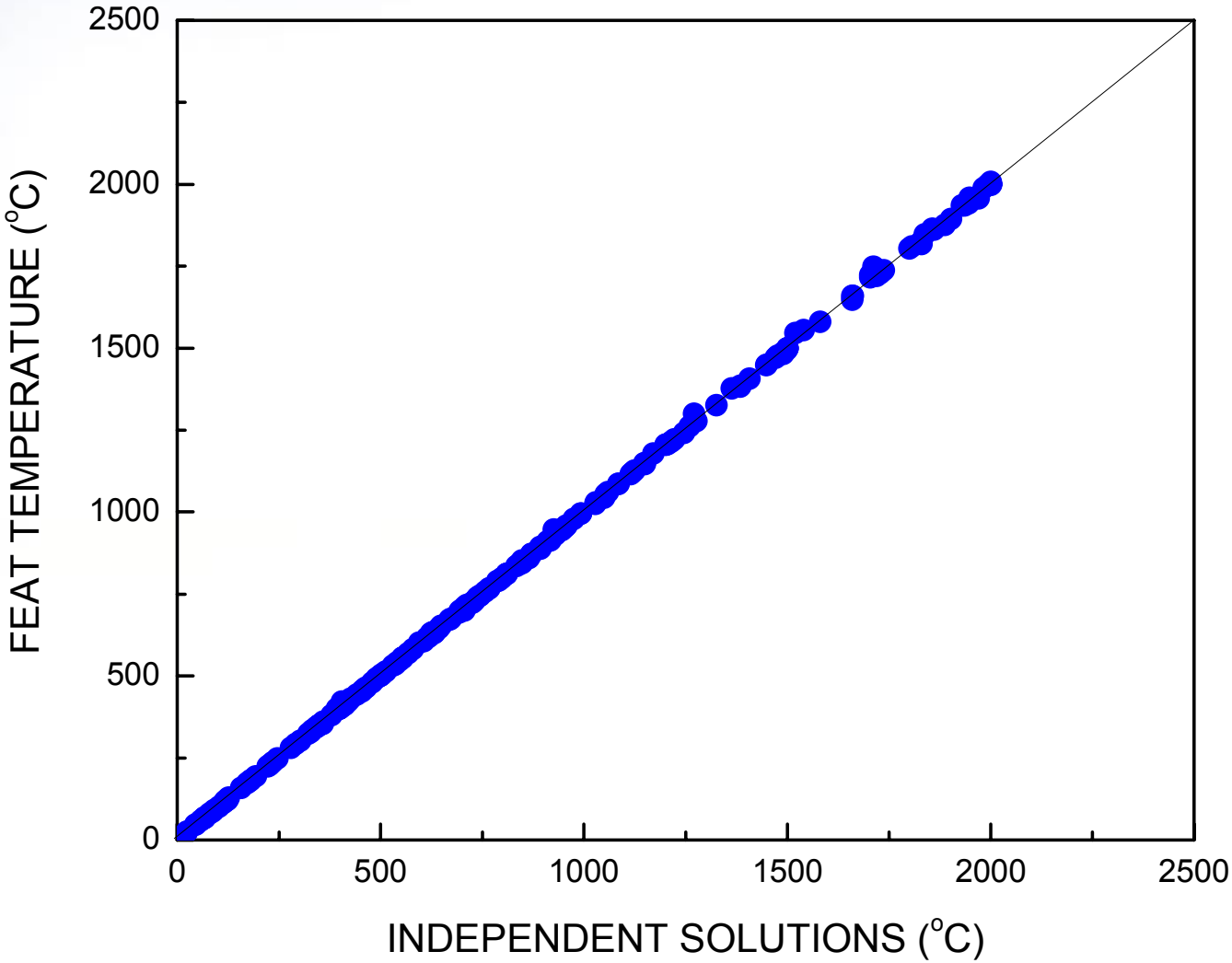
BOW vs. CANFLEX PIE



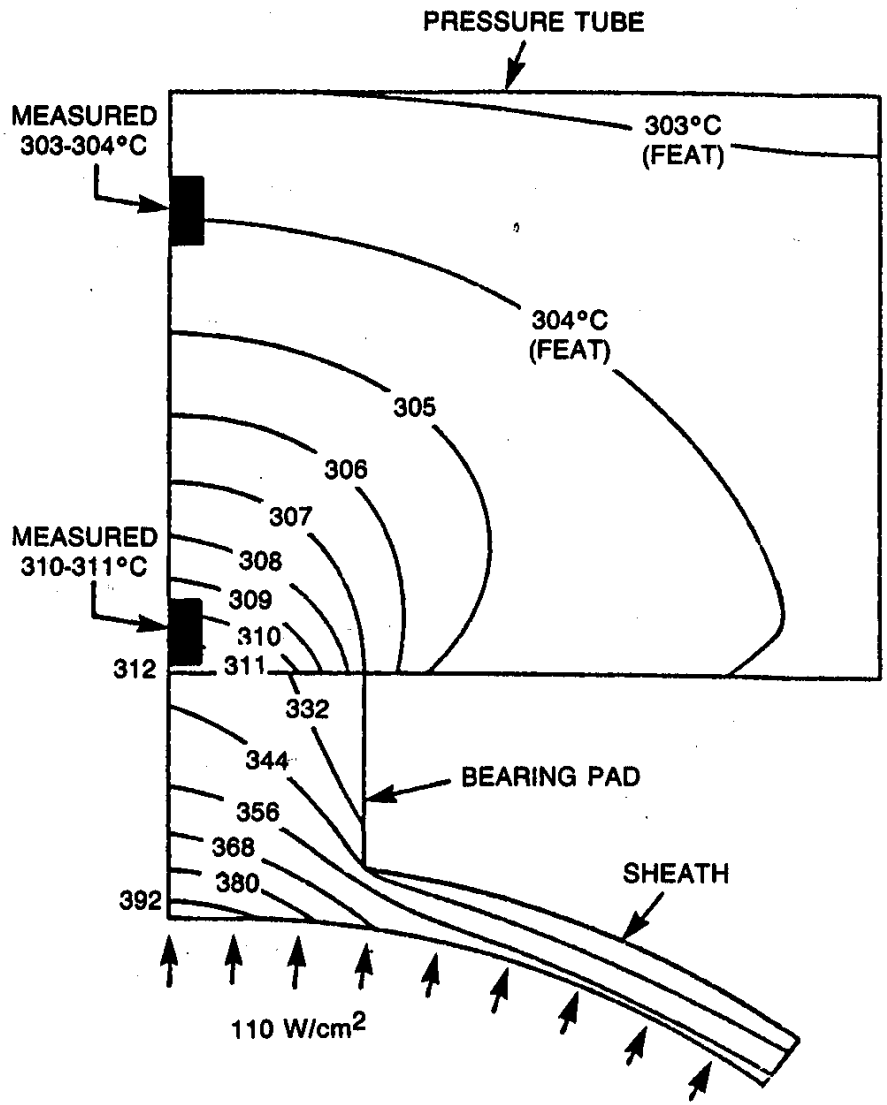
FEAT

- Used to calculate local peak temperatures
  - end-temperature peaking, crevice corrosion between bearing pad and pressure tube, braze void between clad and bearing pad
- 2-D heat conduction, non-linear heat transfer
- Steady-state or transients (e.g. LOCA)
- Flux depression (diameter, enrichment, burnup)
- Variations in material properties (thermal conductivity, specific heat and density)
- Time-dependent boundary conditions
- Multiple bodies (gaps between pellet and clad)
- Finite elements: arbitrary shapes
- Validation against 4 test measurements and 27 independent analytical solutions

FEAT vs. Analytical Solutions



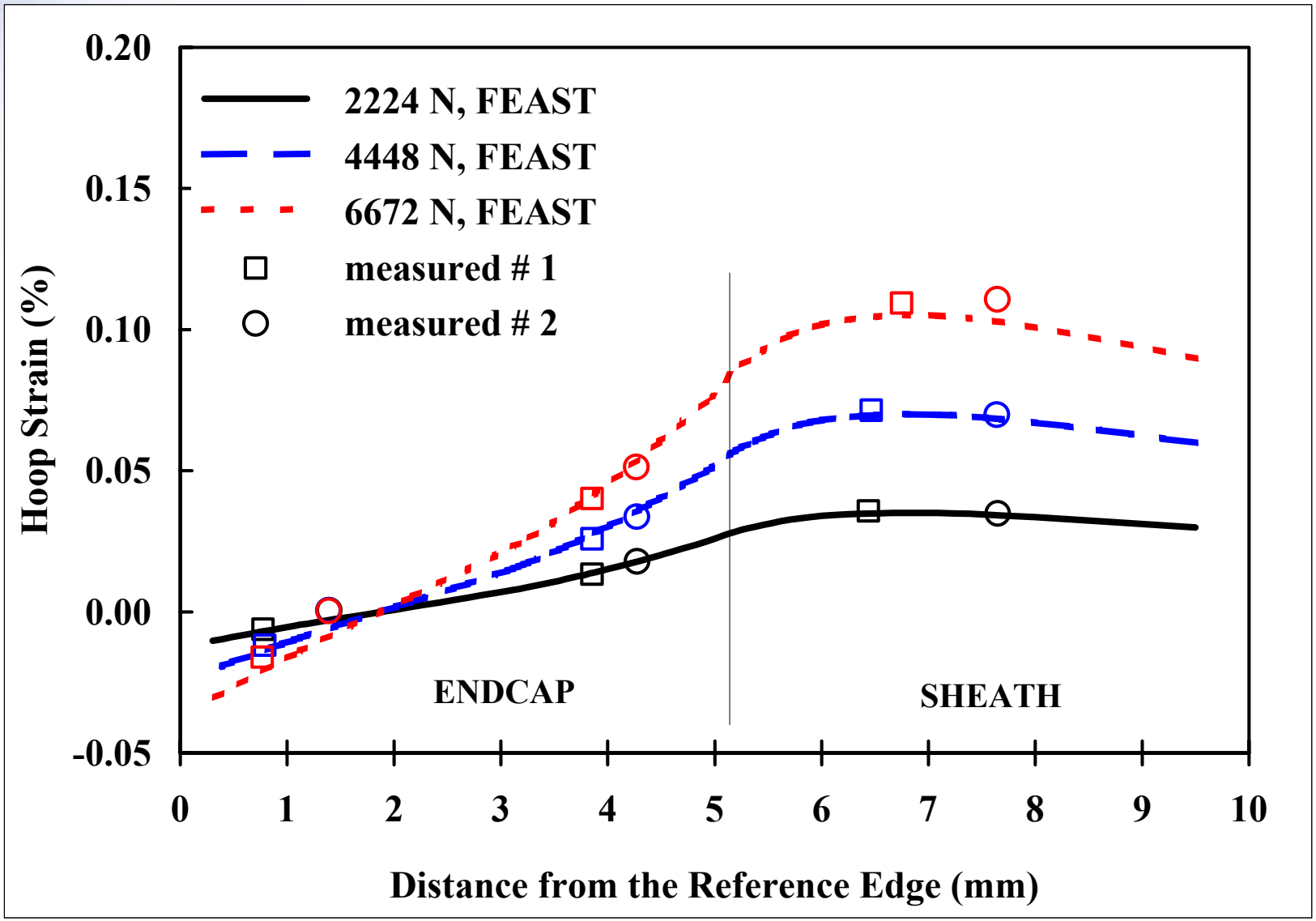
FEAT vs. Measurements



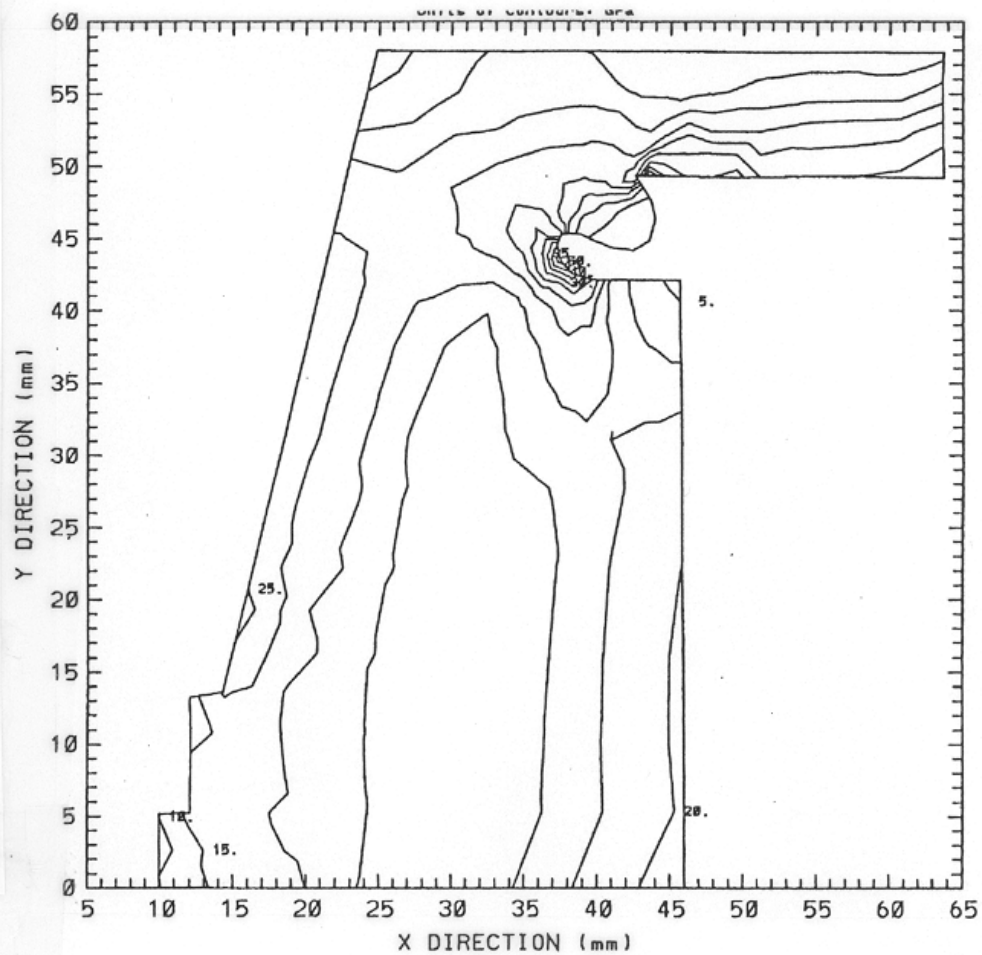
FEAST

- Used to calculate detailed local stresses, strains
  - near endcap, at ridge
- Work density calculation for input into stress corrosion cracking
- Thermal, elastic, plastic, creep, stress relaxation
- Finite element method, 2-dimensional, non-linear stress analysis
- Validated against
  - 2 experimental measurements
  - 18 independent analytical solutions (max diff. of ±1% for half the cases, peak deviation among all cases ±5%)
- Also captures well the observed cracking near endcap junctions

FEAST: Endcap Strains

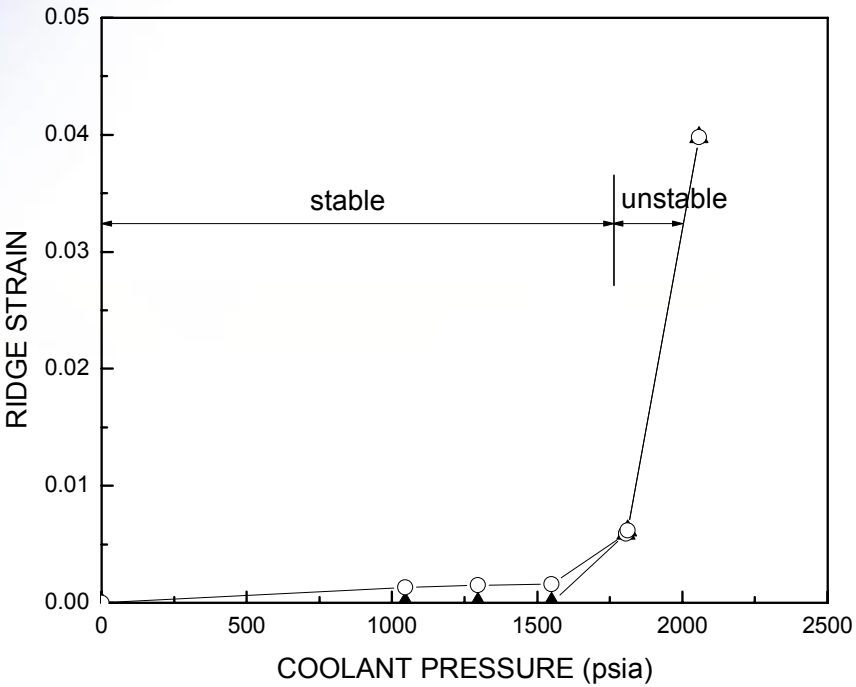


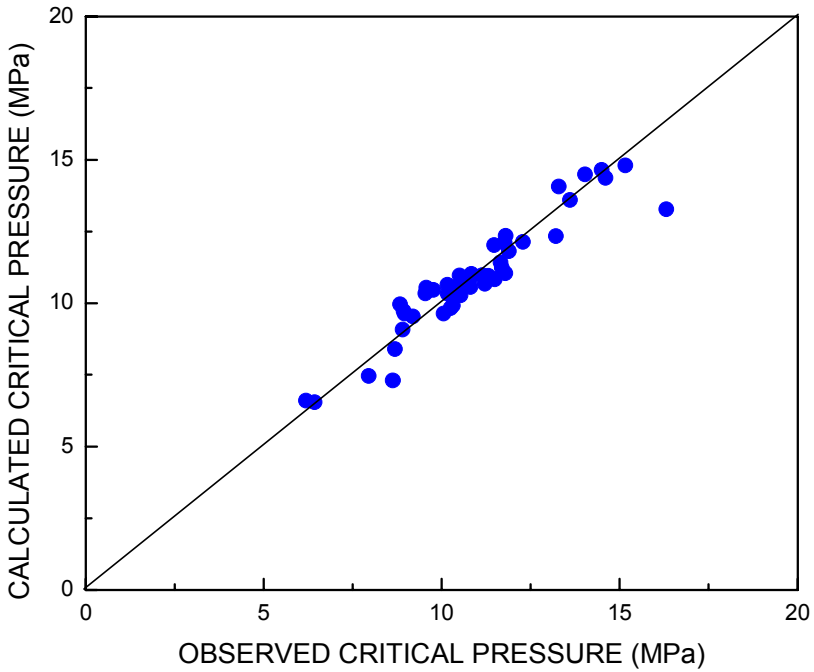
FEAST: Stress Concentrations



LONGER

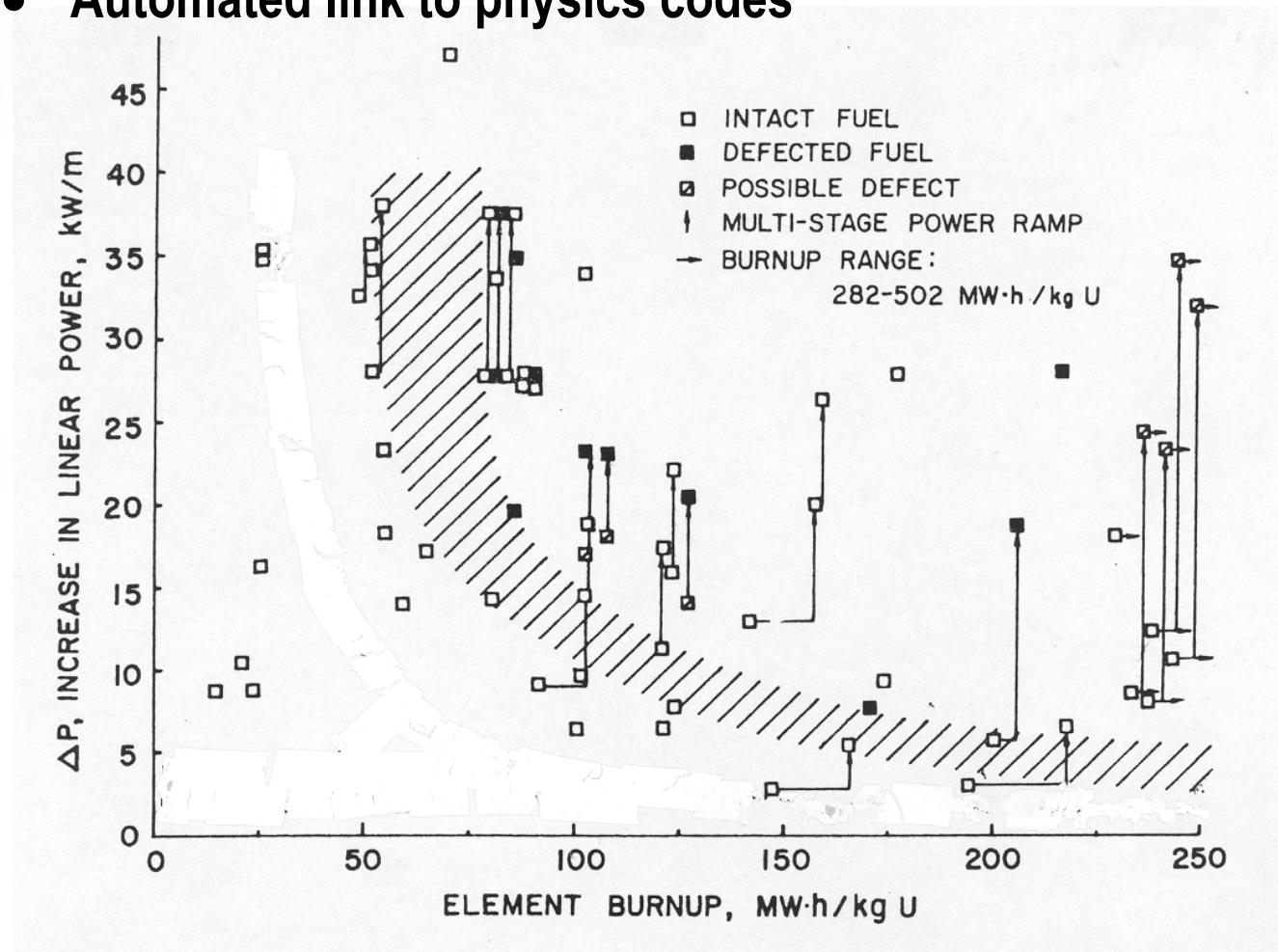
- Used to calculate clad collapse due to coolant pressure
- 48 data points



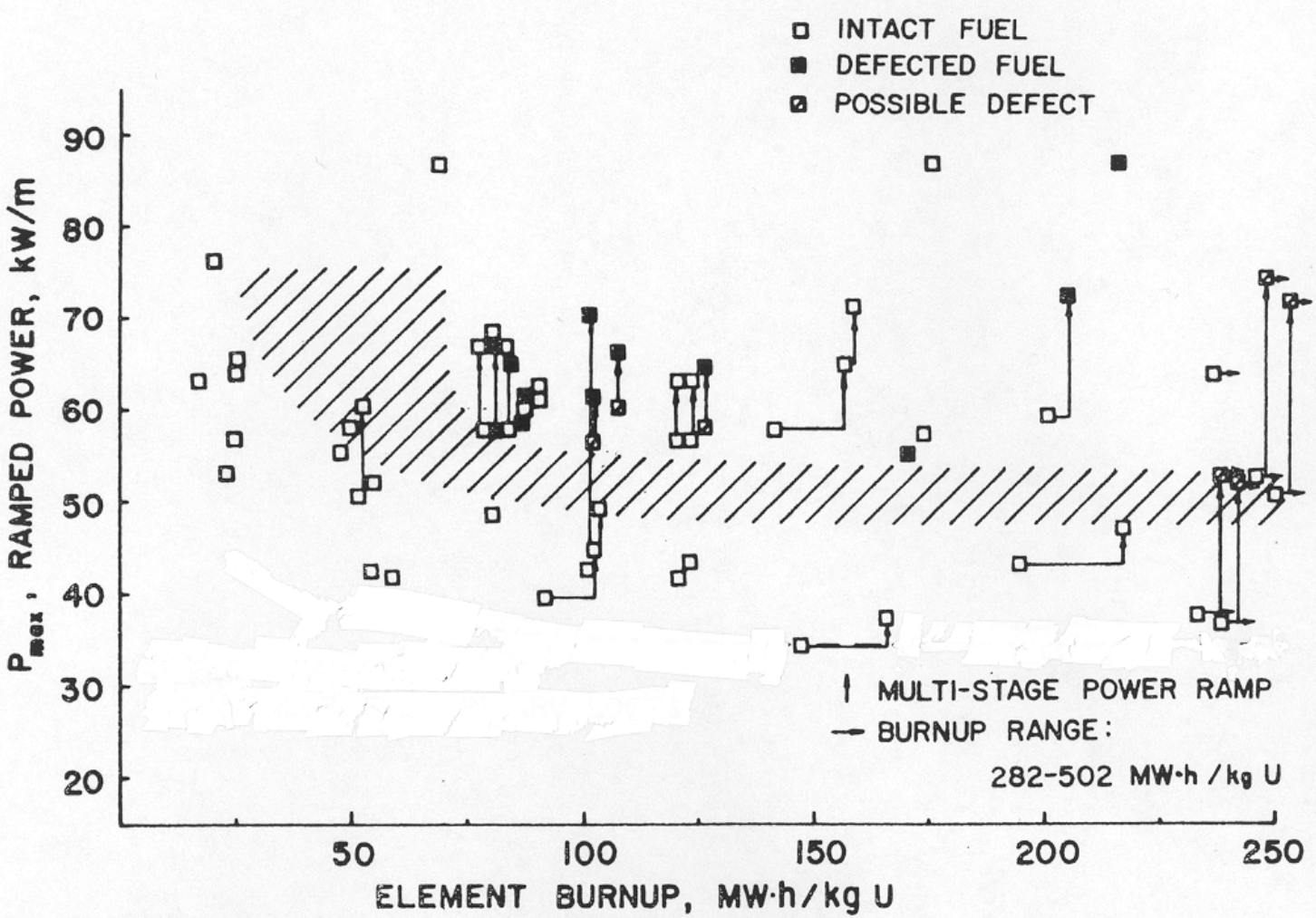


HISTOBUN

- Used to predict core-wide defect probability due to stress-corrosion cracking, using any one of several correlations
- Automated link to physics codes

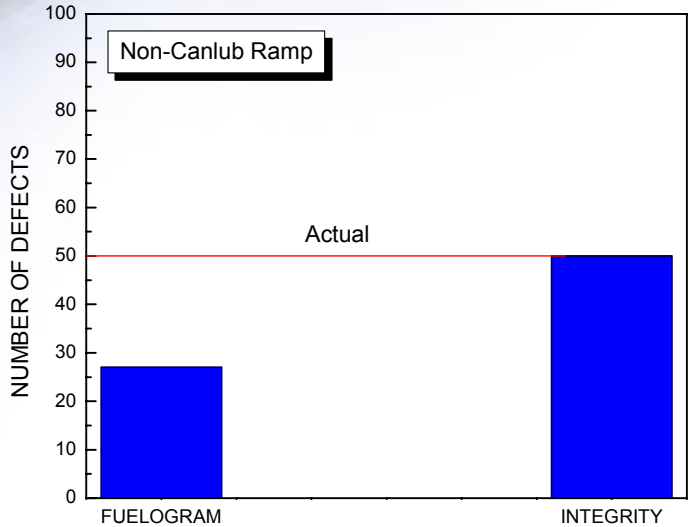


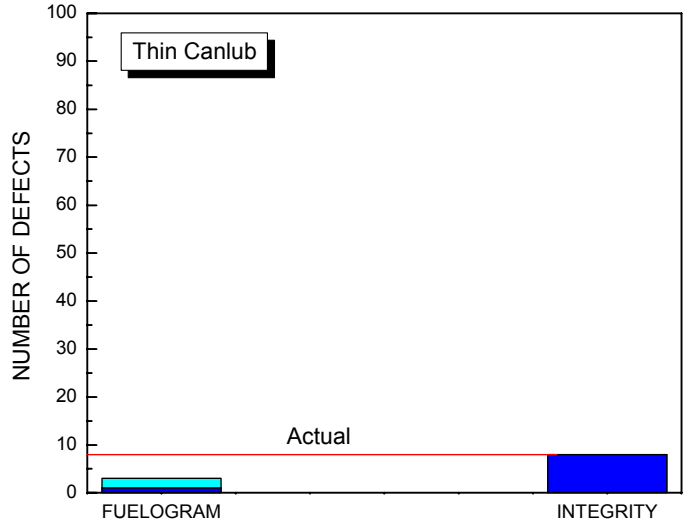
HISTOBUN (continued)

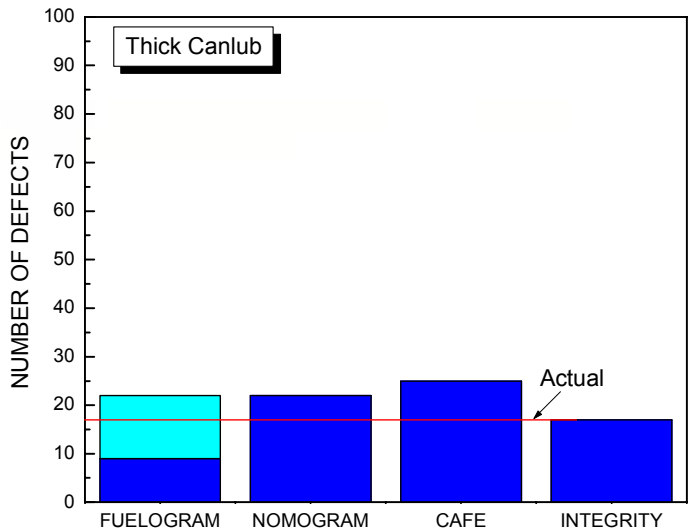


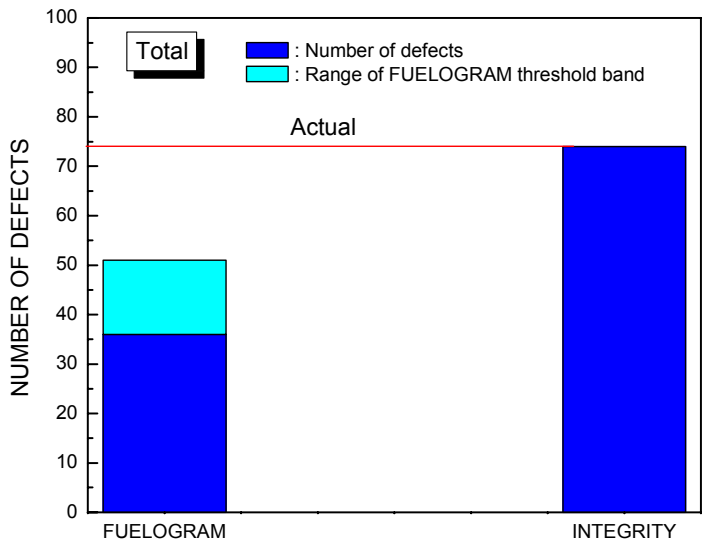
INTEGRITY

- Mechanistic calculation of environmentally-assisted cracking
  - single power ramps (stress corrosion cracking)
  - multiple power ramps (stress corrosion fatigue)
  - cyclic power changes (corrosion assisted fatigue)
- More confident extrapolations to relatively data-sparse regions
  - extended burnups
  - design changes (e.g. pellet shape, element diameter)
  - specification ranges (e.g. diametral clearance)


Non-Canlub Ramp
NUMBER OF DEFECTS
Actual
FUELOGRAM
INTEGRITY


Thin Canlub
NUMBER OF DEFECTS
Actual
FUELOGRAM
INTEGRITY


Thick Canlub
NUMBER OF DEFECTS
Actual
FUELOGRAM
NOMOGRAM
CAFE
INTEGRITY


Total
: Number of defects
: Range of FUELOGRAM threshold band
NUMBER OF DEFECTS
Actual
FUELOGRAM
INTEGRITY

Overall Summary

- The ACR, and its fuel, are based on well-established CANDU technology
  - evolutionary enhancements over existing CANDU 6 reactors
- ACR fuel is based on 3 underlying CANDU fuel technologies:
  - the CANFLEX fuel bundle geometry
  - use of enrichment to achieve extended burnup
  - LVRF for tailoring coolant void reactivity
- ACR fuel qualification will supplement existing data and experience
  - will follow US NRC licensing requirements

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