INL/MIS-22-68431-Revision-0

Irradiations for Advanced Reactors

August 2022

Changing the World's Energy Future

Steven L Hayes, Nicolas E Woolstenhulme, Colby B Jensen, Christian Petie, Gordon Kohse

Idaho National Laboratory

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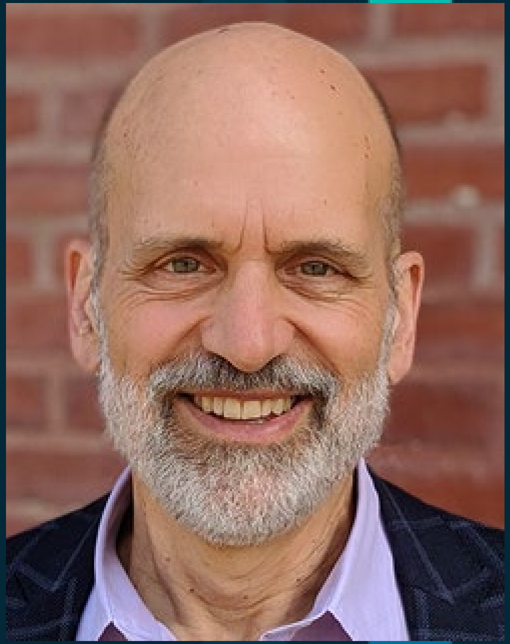

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Developing New Fuels and the Need for Test Reactors

Steven L. Hayes

Developing New Fuels and the Need for Test Reactors

Outline:

- Importance of Developing New Fuels
- Role of Irradiation Testing in Fuel Development
- Traditional Fuel Development Timeline
- Need for Accelerated Testing/Qualification

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Importance of Developing New Fuels

UCO TRISO

Annular Metallic Fuel

- Not every fuel is appropriately-suited for every reactor application.
- Some fuels play an important role in the enhanced performance and/or safety of a particular reactor design, e.g.,
  - ✓ TRISO fuels in gas-cooled reactors
  - ✓ Metallic fuels in sodium-cooled fast reactors.
- Some advanced reactor designs could benefit from new fuels with unique properties/performance.

Design of new advanced reactors should motivate the continued development and qualification of new fuels.

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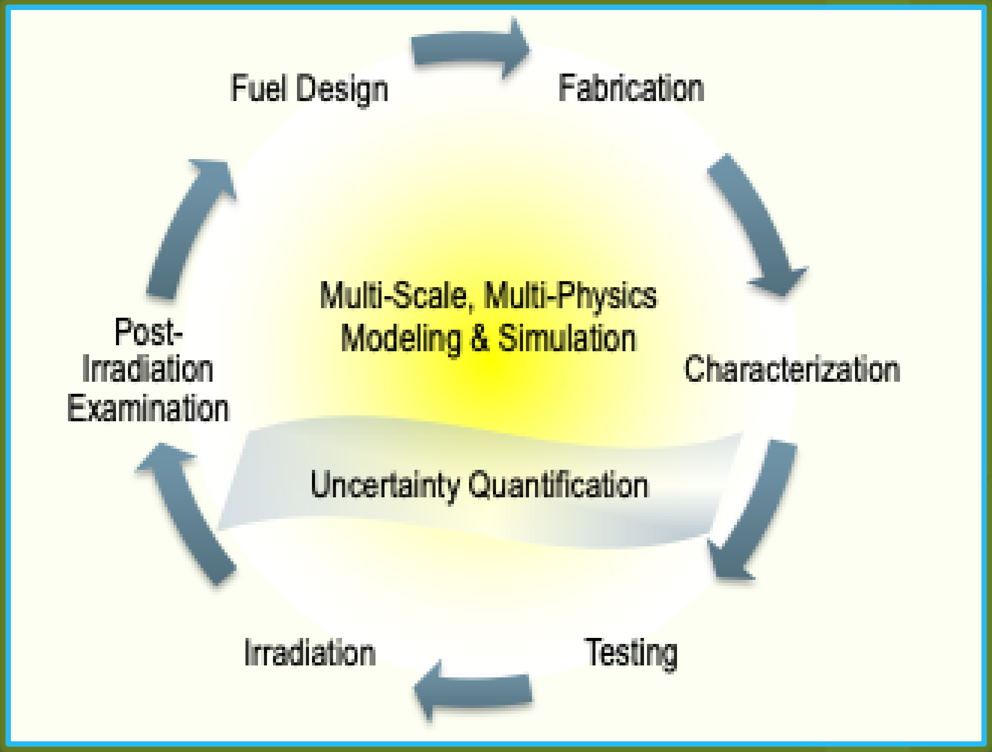
Role of Irradiation Testing in Fuel Development

Irradiation testing is an integral component of the fuel development R&D lifecycle

- *Used to assess phenomena envisioned to impact feasibility and fuel lifetimes (simple experiments)*
- *Used to determine sensitivity of performance to fuel design/fabrication variables and operating conditions, and to establish burnup limits and safety margins for various operating conditions (normal and off-normal)*
- *Used to demonstrate that fuel/fuel-assembly behavior is within the bounds of the fuel safety case (prototypic experiments)*

Fuel Development R&D Lifecycle



It is difficult to perform testing of this nature in NRC-licensed reactors.
DOE-authorized test reactors are best-suited for this purpose.

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Traditional Fuel Development Timeline

- **Historical Development/Qualification of New Fuels**
  - Review:

    Journal of Nuclear Materials, 371 (2007) 232-242
  - Takes 20–25 years
  - Highly empirical in nature
    - *Extensive steady-state irradiation testing to bound all operating conditions, accumulate adequate statistics, and collect data for validating fuel performance codes*
    - *Transient testing to bound all off-normal conditions*

The traditional fuel development timeline tends to stunt design innovation for new reactors.
Accelerated fuel testing/qualification is needed to support advanced reactor design and deployment.

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Journal of Nuclear Materials 371 (2007) 232–242

journal of nuclear materials

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An approach to fuel development and qualification

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Abstract

Some of the rationale for nuclear energy technology development in the US has been lost or forgotten over the past two decades with the lack of a focused reactor development program. But the emergence of new R&D programs points to a need to understand how best to plan for a long-term fuel development program. The rationale for such a program is not easily found in the literature, so the authors have suggested a structure and rationale. The approach is described as four phases, with emphasis on selecting a reference fuel concept, evaluating and improving the fuel to develop a fuel specification for a reference design, obtaining data to support a licensing safety case for the fuel, and final qualification of the fuel for a specific application. Because a fuel program requires long-lead-time irradiation testing, bringing a fuel design from the initial concept through licensing might take over 20 years.

1. Introduction

Emergence of nuclear energy R&D programs in the US (e.g., the Generation IV initiative [1,2], the Advanced Fuel Cycle Initiative [3], and perhaps the recently announced Global Nuclear Energy Partnership for which plans are just now being formulated [4]) and elsewhere has motivated consideration of reactor fuels for new applications [5,6]. To support long-range planning in the recent programmatic environment, it has been necessary to consider and describe the process needed to bring a new fuel type to implementation. Because the rationale used previously was not fully described in the literature, the authors attempt to do so here.

The fuel development approach described is based on experience with, or observations of, developing and improving fuels at various stages of technical maturity, reflecting previous and current efforts with fuels for gas-cooled reactors [7,8], fast reactors [9–12], research reactors [13], and even light water reactors [14]. Other descriptions may also be valid, but the structure and rationale here has recently been used to identify the tasks and sequencing that best serve US program needs. Whether all the elements of the full and generic program described here are necessary for a given application depends on the needs and technical maturity of the fuel technology being addressed. For example, fuel

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Need for Accelerated Testing/Qualification

- **Fuel qualification** refers to regulatory approval for using a specific fuel design in a specific reactor, under a set of identified conditions.
- Accelerating fuel qualification could include:
  - Accelerating fuel testing in test reactors
  - Expanded use and impact of in situ instrumentation in reactor experiments, and/or high-throughput PIE methods
  - Development/use of advanced, mechanistic, and predictive M&S methods
  - Development/use of novel qualification/regulatory paradigms.

While the number/nature of fuel experiments may change, **the need for irradiation testing of fuels in test reactors will continue.**

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Advanced Reactor Irradiations in ATR

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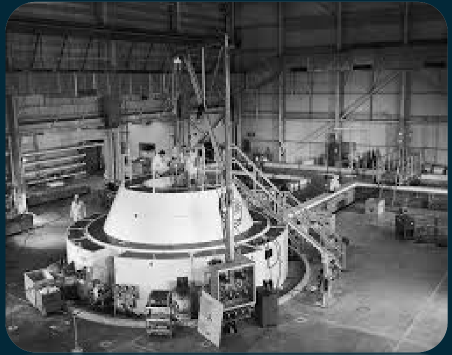
The Advanced Test Reactor (ATR)

- Constructed in 1967, it is third in a series of landmark materials test reactors (Materials Test Reactor [MTR], Engineering Test Reactor [ETR], Advanced Test Reactor [ATR]).
- It represented the pinnacle of materials test reactor technology in a golden era of American engineering, and was contemporary to legends such as Saturn V rockets and SR-71 spy planes.
- More than 50 years later, it remains operational as the nexus of research being conducted into nuclear fuels and materials.

MTR

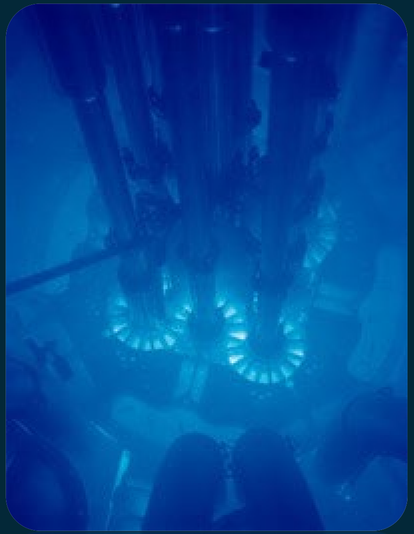


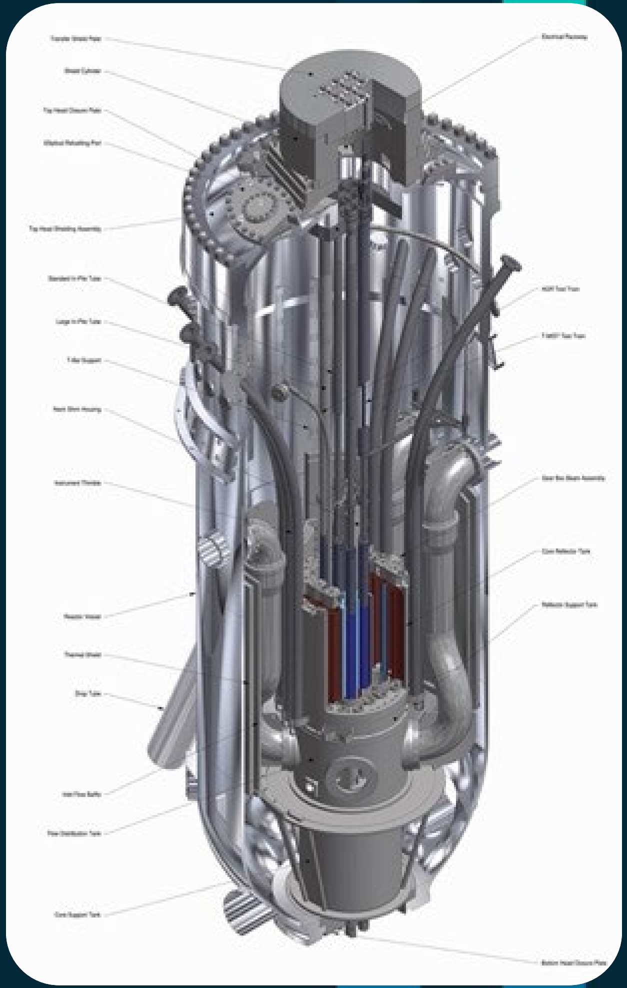
ETR

ATR

ATR Background

- High power density (1 MW/L) enabled by the core design
  - Pressurized light water (~60 C, 2.7 MPa)
  - Aluminum-clad, plate-type driver fuel, high coolant velocity (~15 m/s)
- Rotating beryllium control drums
  - Smooth “chopped cosine” axial flux profile across the 1.2-m-long core
  - Large power tilts across the core to suit various flux needs
- ~60-day cycles, with outages to refuel and reconfigure experiments
- ~200+ full-power days per year
- 2021 beryllium replacement and other refurbishments
  - Planned operations for decades to come





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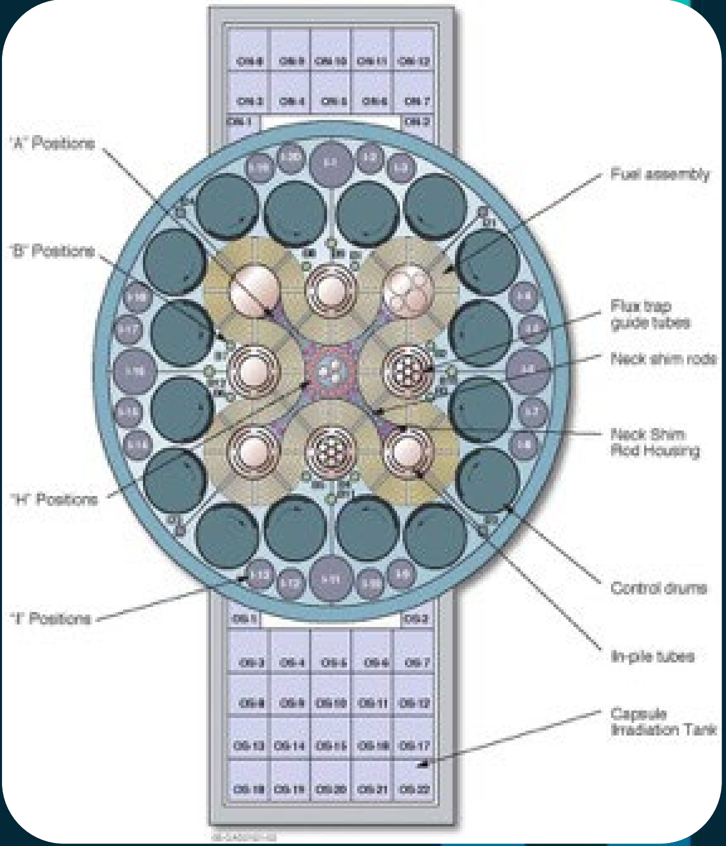
ATR Background

- Iconic serpentine fuel arrangement
  - Nine flux traps (six permanently used for water loops)
  - Numerous other test positions with varied size and flux capabilities
- Unmodified neutron spectra (thermal-to-fast ratio)
  - Inner core: ~1:1, Reflector: ~10:1

Table 2. Approximate peak flux values for various ATR capsule positions for a reactor power of 110 MW_{th} (22 MW_{th} in each lobe).

Position	Diameter (cm/in)[a]	Thermal Flux (n/cm^2-s)[b]	Fast Flux (E>1 MeV) (n/cm^2-s)	Typical Gamma Heating W/g (SS)[c]
Northwest and Northeast Flux Traps	13.3/5.250	4.4×10^{14}	2.2×10^{14}	
Other Flux Traps	7.62/3.000[d]	4.4×10^{14}	9.7×10^{13}	
A-Positions				
(A-1 - A-8)	1.59	1.9×10^{14}	1.7×10^{14}	8.8
(A-9 - A-16)	1.59/0.625	2.0×10^{14}	2.3×10^{14}	
B-Positions				
(B-1 - B-8)	2.22/0.875	2.5×10^{14}	8.1×10^{13}	6.4
(B-9 - B-12)	3.81/1.500	1.1×10^{14}	1.6×10^{13}	5.5
H-Positions (14)	1.59/0.625	1.9×10^{14}	1.7×10^{14}	8.4
I-Positions				
Large (4)	12.7/5.000	1.7×10^{13}	1.3×10^{12}	0.66
Medium (16)	8.26/3.500	3.4×10^{13}	1.3×10^{12}	
Small (4)	3.81/1.500	8.4×10^{13}	3.2×10^{12}	



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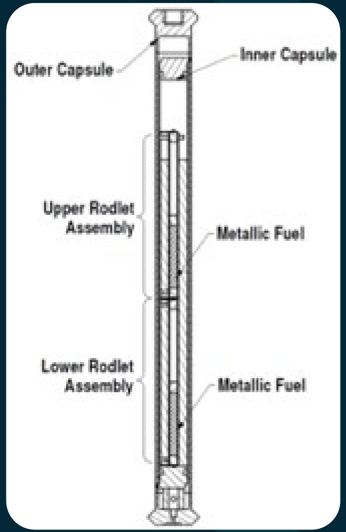
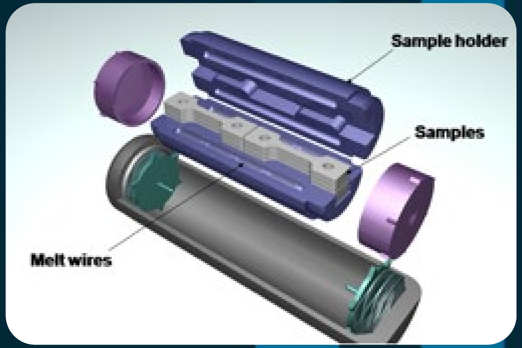
Drop-In Capsules

- Extensive experience with drop-in capsules
  - Inert gas gap between the specimen and the capsule wall in order to elevate the temperature
  - Set-and-forget method with passive temperature monitoring (measured post-irradiation)
  - Specimen geometries range from toothpick-sized to soda-bottle-sized
- First-to-data: cost & schedule efficiency
  - Structural material irradiations (~5 dpa/yr in SST)
  - Exploratory and parametric tests for fuel technology candidates
- High-flux, custom enrichment (INL fabrication)
  - Fuel rodlet accelerated burnup testing (10 at%/yr) in a liquid sodium thermal bond capsule





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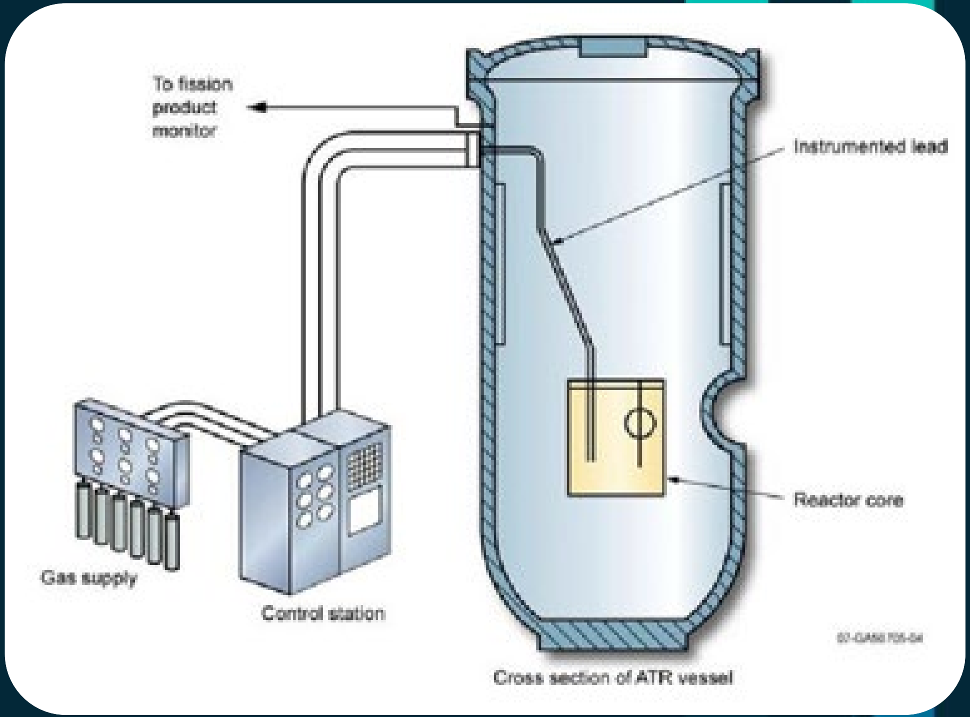
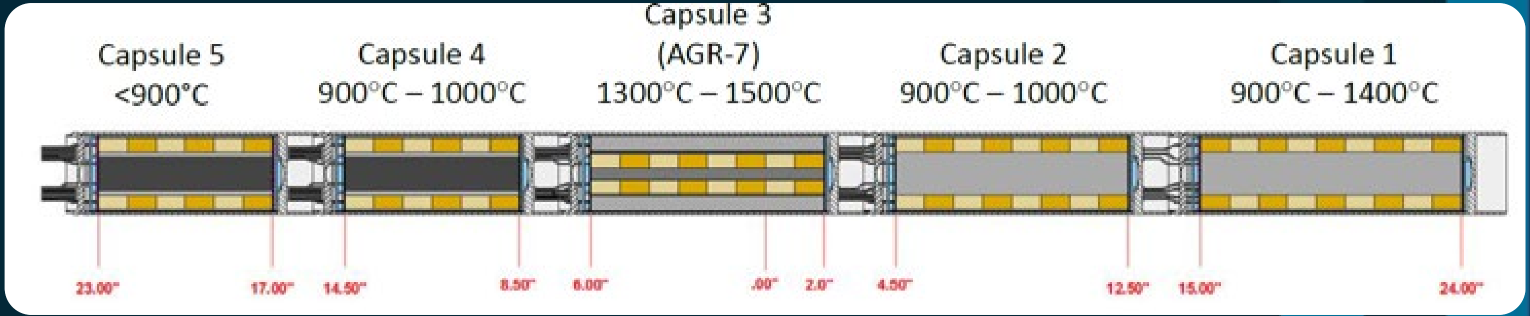
Gas Lead-Out

- A thimble tube carries gas and instrumentation lines
  - Viable in flux traps and reflector positions
- Canonical example: AGR TRISO irradiation series
  - Multiple capsule elevations, each with real-time gas composition control
  - Concerted with Hf flux-shaping to achieve desired temperatures and fission rates
  - Real-time fission product monitoring of exit gas sweep
- Future capability under development, real-time neutron filter adjustment via the ATR ^{3}He pressure control system
  - Power cycles/transients, minutes-to-hours time scale (e.g., load following)

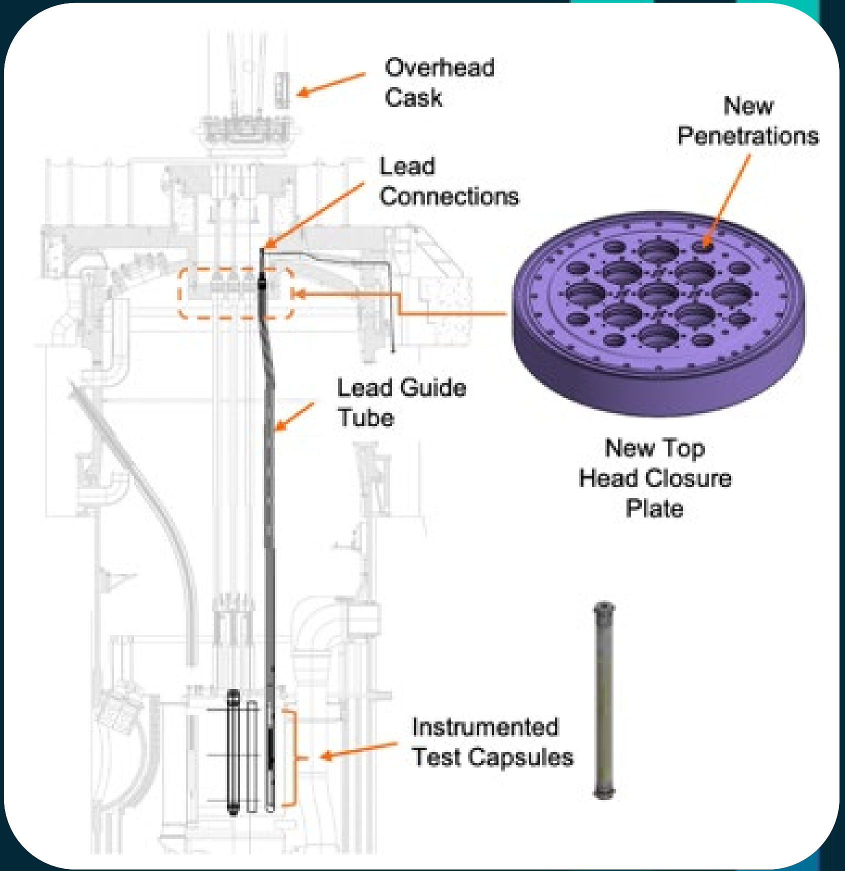


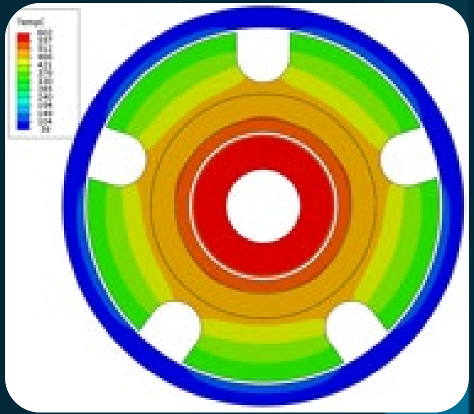


Michael Davenport, D. A. Petti, Joe Palmer, "STATUS OF TRISO FUEL IRRADIATIONS IN THE ADVANCED TEST REACTOR SUPPORTING HIGH-TEMPERATURE GAS-COOLED REACTOR DESIGNS", 2016 International Topical Meeting on High Temperature Reactor Technology.

Instrumented Capsules

- Compared to drop-in capsules, full gas lead-outs represent a major increase in design/fabrication efforts
  - Yet worth it (depending on the need)
  - Still, some needs sit somewhere between the bicycle and Cadillac options
- Introducing a new approach: instrumented capsules (the Honda Civic option)
  - Long capsules with instrument leads
  - Manual gas gap adjustment during outages in order to reduce plant plumbing/operational costs
  - Enabled by new penetrations in the ATR top closure flange, facilitating installation/removal from the vessel
- First tests to begin in 2023
  - Metallic fuel and advanced ceramic rods with real-time thermal conductivity probes
  - Fast-neutron spectrum via Cd basket (see next slide)





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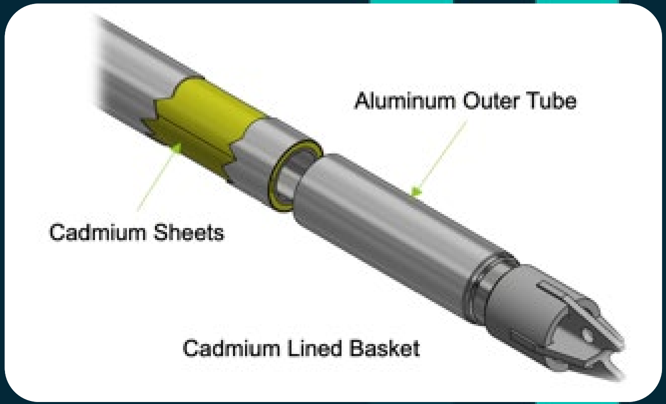
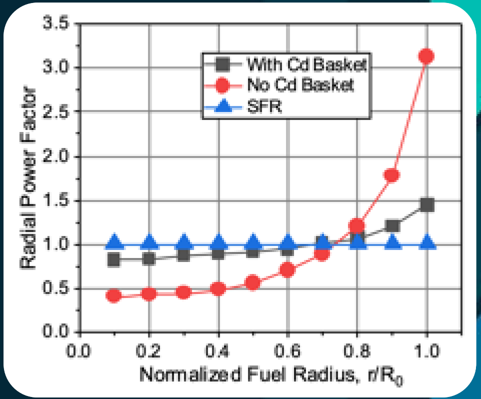
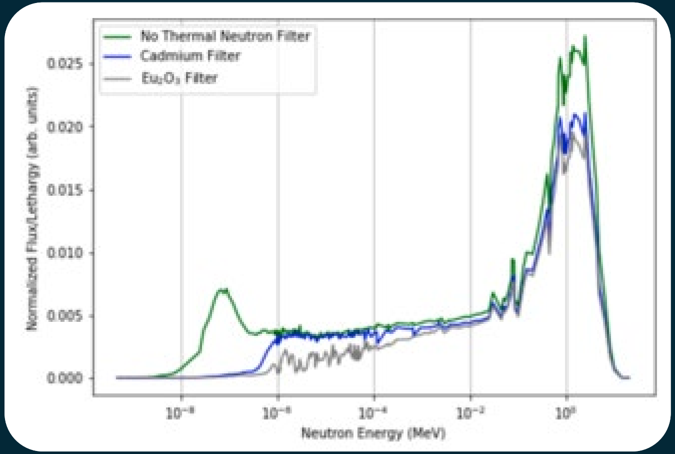
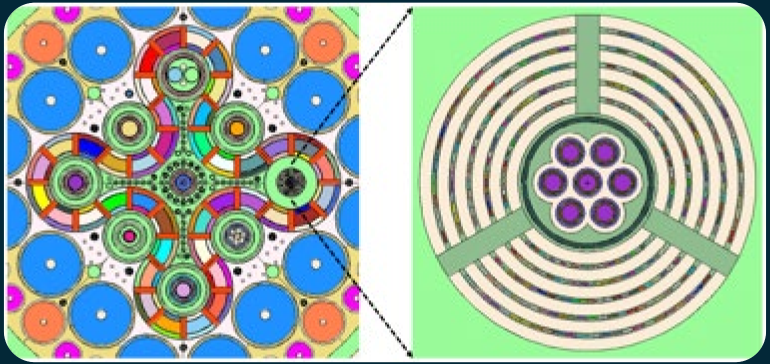
Spectral Modification

- Years of experience in using thermal neutron filtering in inner core positions
  - The cadmium basket somewhat mimics the fuel radial power profile in true fast spectrum reactors (INL/EXT-17-41677)
  - Prototypic SFR diameter & fission heating rates, ~3 at% burnup per year short fuel length rodlet (3.8 cm)
- Next-gen fast-flux-boosted test development underway
  - Booster fuel rings and thermal neutron filter in the ATR flux trap
  - Fast flux (>0.1 MeV): ~4.5e14 n/cm2s; thermal-to-fast ratio: ~1:100
  - Full-scale, full length, 7-pin fast reactor test section with prototypic linear heating rates









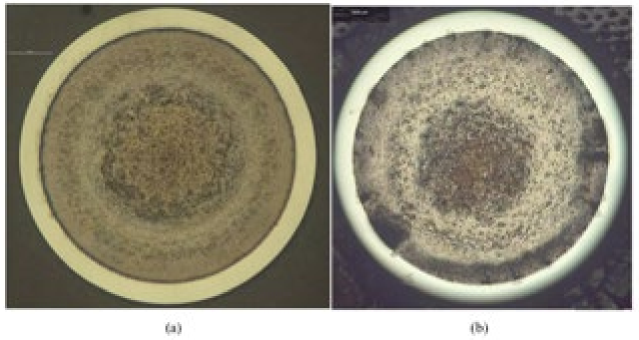


Figure 69. Metallography of Pu-12Am-40Zr irradiated to approximately 20% burnup in: a) Phénix fast eactor (FUTURIX-FTA DOE2), and b) cadmium-filtered position in ATR (AFC-1D R4).

Nuclear Engineering and Design 387 (2022) 111623

Transient Reactor Test Facility (TREAT)

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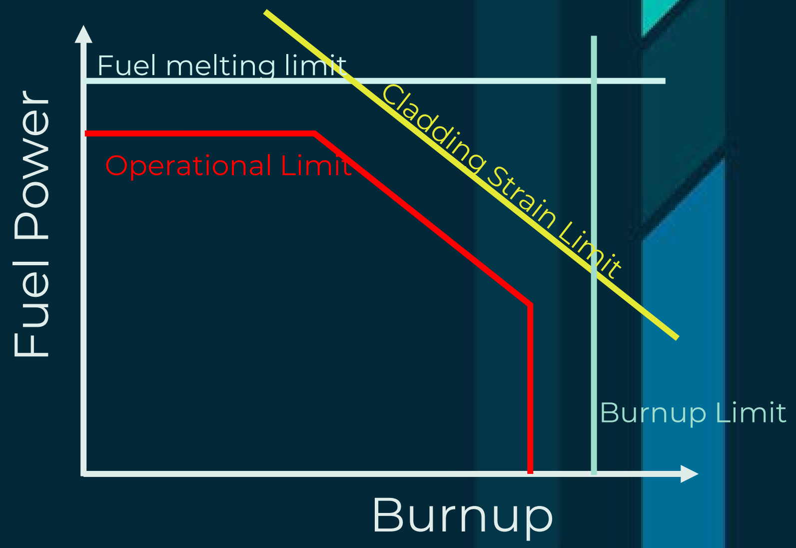
Why Transient Test Nuclear Fuels & Materials?

- Transient testing is like car crash testing for nuclear fuels.
- Licensing a fuel system *requires* (see NUREG-0800):
  - identification of all **degradation mechanisms and failure modes**
  - definition of the **failure thresholds** corresponding to each degradation mechanism
  - applies to **normal operations, anticipated operational occurrences, and design basis accidents**
- Many operational limits are dependent on degradation and failure thresholds.
- Such testing enables economic reactor operations via improved fuel designs and a better understanding of fuel performance.



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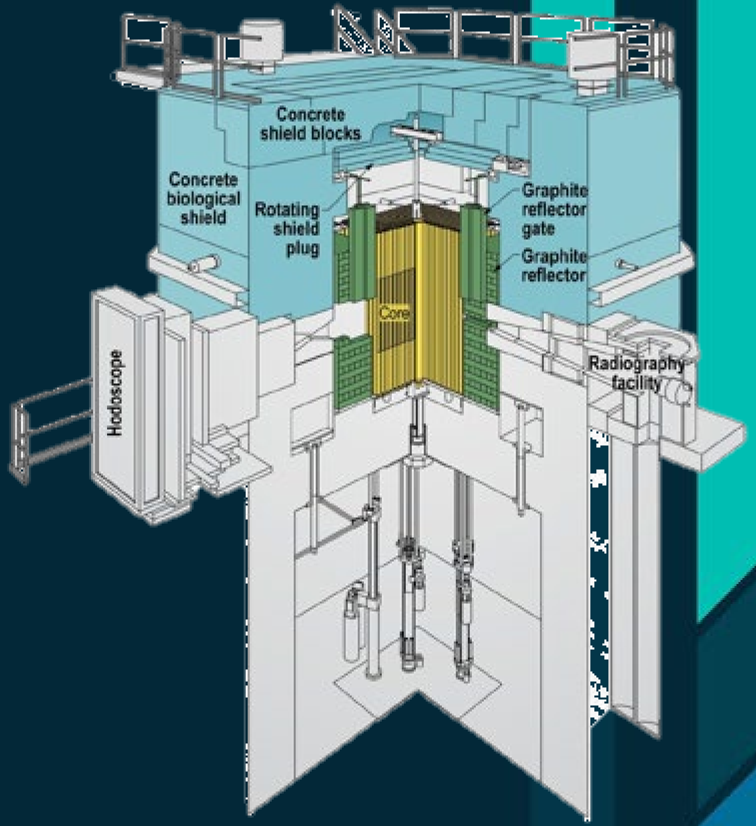
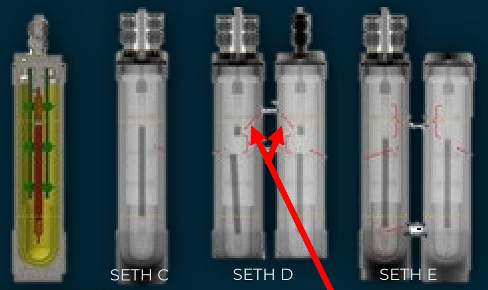
TREAT Design & Experimental Approach

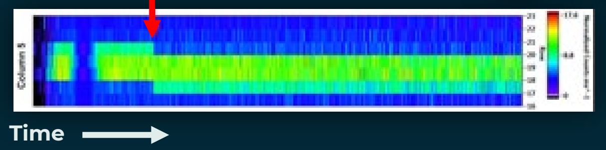


- The Transient Reactor Test (TREAT) Facility, operated during 1959–1994, resumed operations in 2017 to support fuel safety testing and other transient science efforts.
- Zircaloy-clad graphite/fuel blocks comprise the core, cooled by air blowers.
  - Virtually any power history is possible within **2500 MJ max core transient energy**
  - No reactor pressure vessel/containment, facilitates **access for in-core instrumentation**
  - 4 slots view core center, 2 in use for the **fuel motion monitoring system & neutron radiography**
- To test specimens, the reactor provides brief (and typically extreme: up to 10^{17} $n \cdot cm^{-2} \cdot s^{-1}$) shaped neutron flux histories.



TREAT neutron radiography shows post-test fuel condition

The TREAT fuel motion monitoring system detects fuel motion in situ



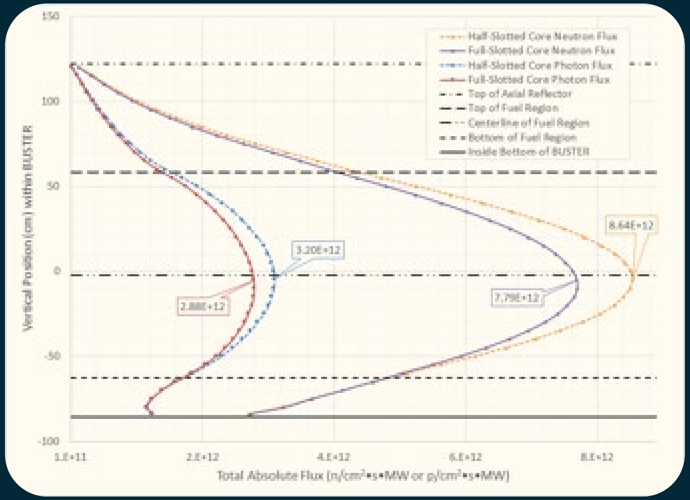
- The experiment vehicle does everything else.
  - Safety containment, specimen environment, and instrumentation
- The TREAT Restart project only addressed facility refurbishment.
  - Historic assembly and support infrastructure had largely dissolved
- TREAT's experiment history is diverse, its capabilities are unique, and the predicted needs of the scientific community are vast.
  - ➢ **Emphasized the need for modern, multipurpose experimental tools**

Virtual facility tour: https://inlgov360.b-cdn.net/TREAT/tour.html

TREAT Experimental Approach

- For a given experiment, the typical residence time in the TREAT core is a few days
  - Of this period, the transient itself only lasts somewhere from a few milliseconds to minutes
  - Even with multiple transients being conducted on the same specimen, the fluence is extremely low (a dpa of effectively zero)
- **TREAT is best suited for testing fuels, as well as materials that could interact with fuels** (e.g., cladding and ducts), under extreme conditions for nuclear-heated safety research
  - If significant structural burnup or material fluence is needed prior to transient testing, it would be better to accomplish it in another reactor (e.g., ATR or HFIR)

Axial flux profile



Neutron Spectra & Transient Shift

Figures from: Bess et al., Ann. Nucl. Energy 124 (2019) 270

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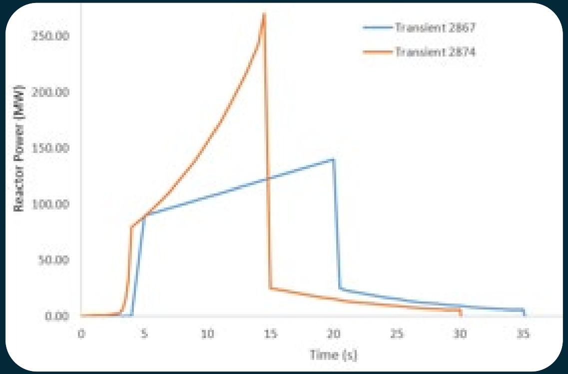
Reactor Power Control

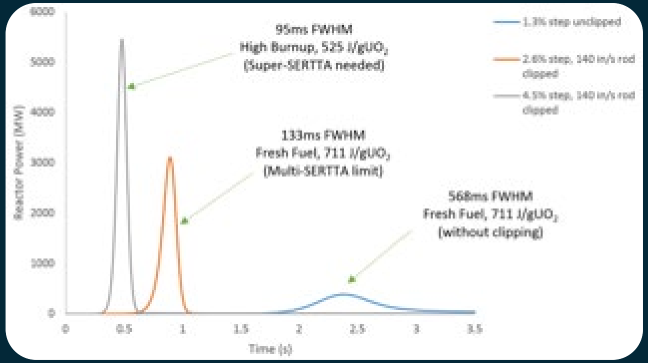
- TREAT is a transient reactor, not just a pulse reactor – “windowed” power histories
- Graphite heat sink, nimble control rod system → flexible power maneuvers
- The rod control system can accept feedback from experiment instrumentation


TREAT subpile room

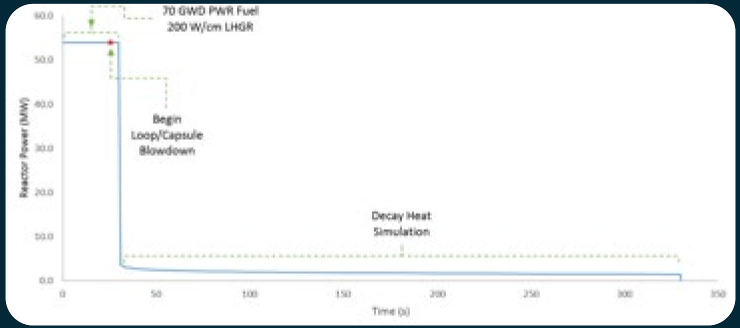
Power Ramps



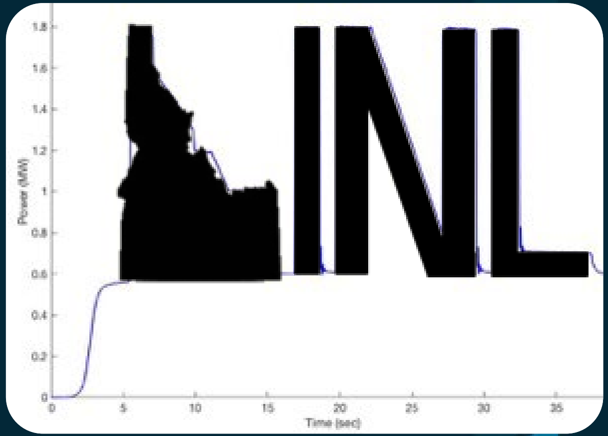
RIA “Pulse”



Loss of Coolant



Complex-Shaped
Transient
and so on...

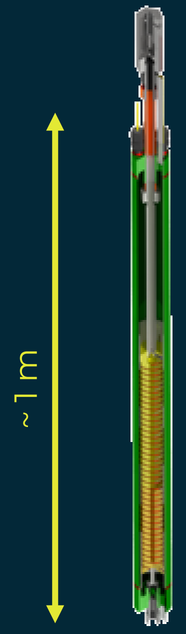


Transient Testing Testbeds and Infrastructure

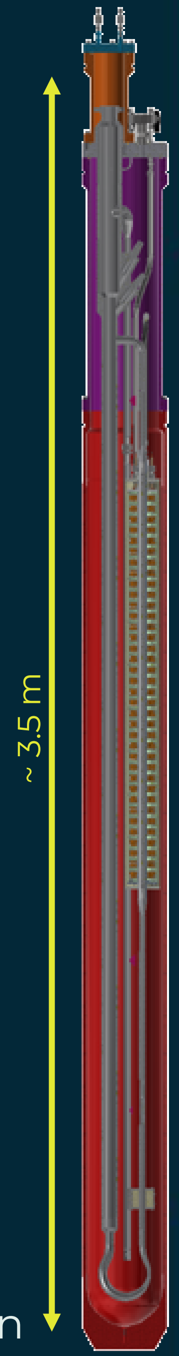
- TREAT programs fall into one of five product lines, each supported by its own testbed infrastructure

- Light-Water Reactor
- Sodium Fast Reactor
- Advanced Reactor
- Space Nuclear Propulsion
- National Security
- Instrumentation and Modeling Platform

- Each testbed naturally divides in two size scales, also distinguished by passive or active cooling.
- Most R&D plans include tests from <u>both</u> size scales.



Cartridge-in-console architecture is used for the general infrastructure, whenever possible



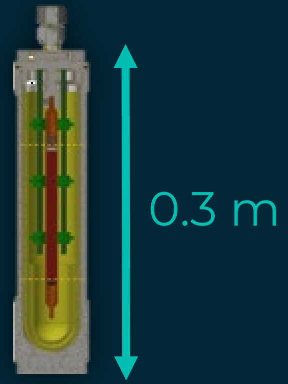
“Capsule-scale”
More affordable static-environment devices

“Loop-scale”
Devices featuring active thermal-hydraulic manipulation

TREAT Experimental Testbeds

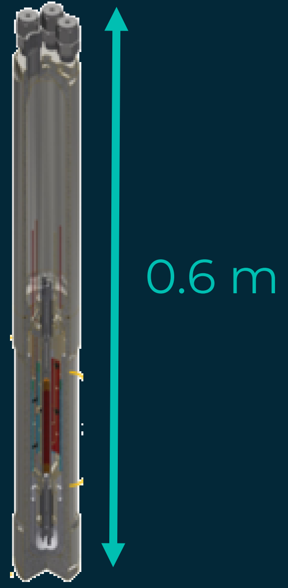
- Reactor *<u>and</u>* hot cell facility integration

General purpose

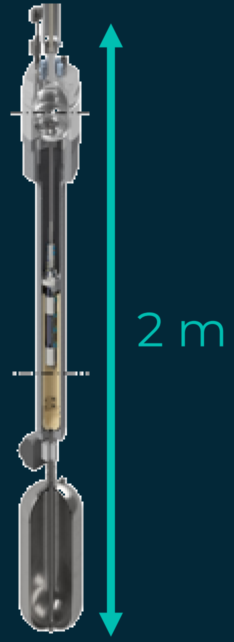


SETH
Separate effects static capsule

Water testbed

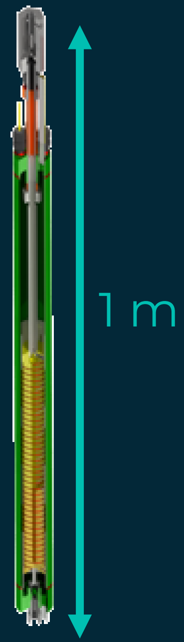


SERTTA
Static pressurized-water capsule

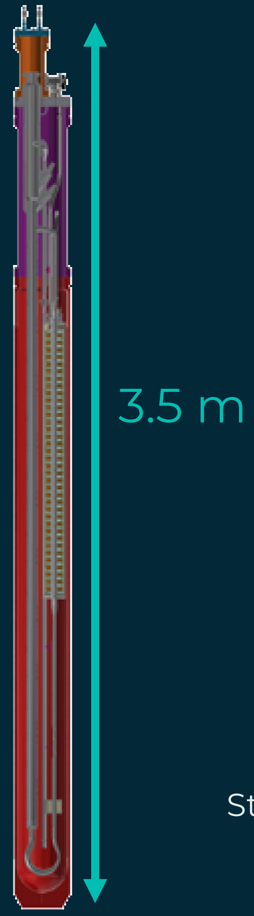


TWIST
Pressurized-water blowdown capsule

Sodium testbed

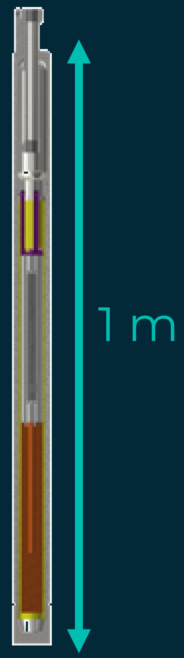


THOR
Liquid-metal-bonded heat-sink capsule

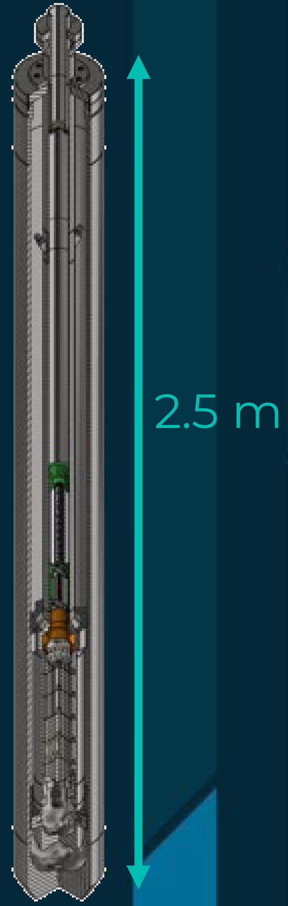


"Mark-IV"
Flowing Na loop

Space nuclear testbed



"SIRIUS"
Static gas capsule



Flowing hydrogen loop

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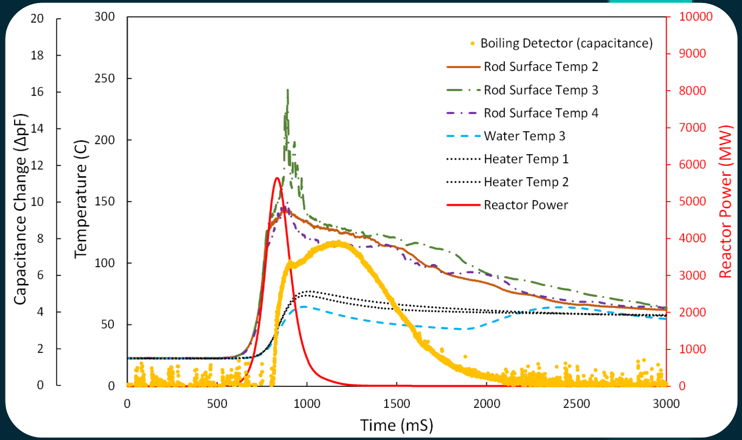
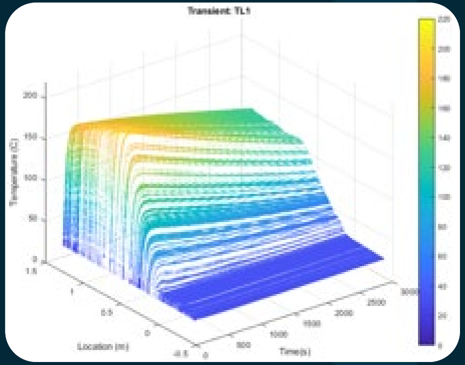
In Situ Instrumentation

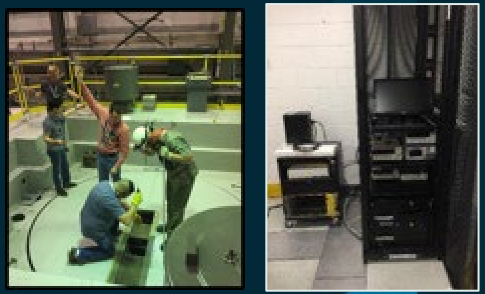
- Extensive in situ measurements are routine in TREAT.
  - Unparalleled core access
- Desired data should be an important initial consideration.
  - A wide array of options is now available.
  - Development of custom approaches is expected and welcome.
- Laboratories have dedicated facilities and expertise for designing, fabricating, qualifying, and interpreting advanced instrumentation.





In situ data using advanced instrumentation

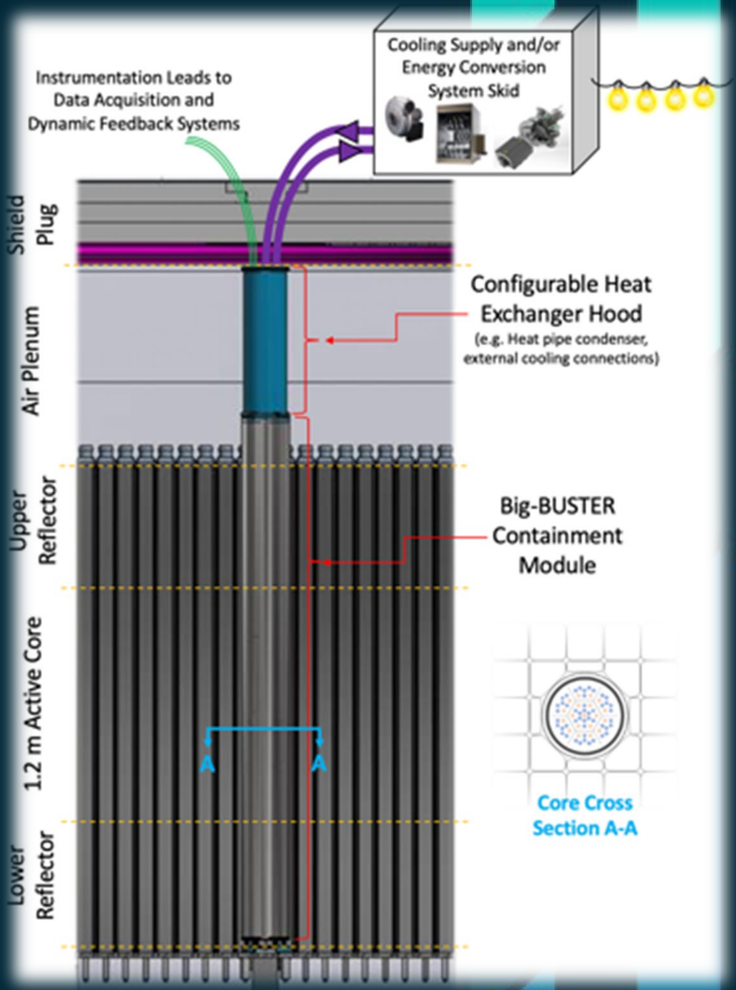
Out-of-pile characterization



In-pile characterization

Non-Traditional Applications of the TREAT Facility

- Fission effects on material properties
- Nuclear materials under extreme conditions (very high temperature properties and behaviors)
- Dynamic multiphysics instrumentation testing and model benchmarking
- System-scale core component testbed with nuclear heating
  - Non-safety-category automatic reactor control system



NIMBLE

Reactor Core Component System Testbed

Irradiation testing in the High Flux Isotope Reactor

Chris Petrie

Oak Ridge National Laboratory

Contributions from P. Mulligan, S. Chapel, N. Russell, R. Howard

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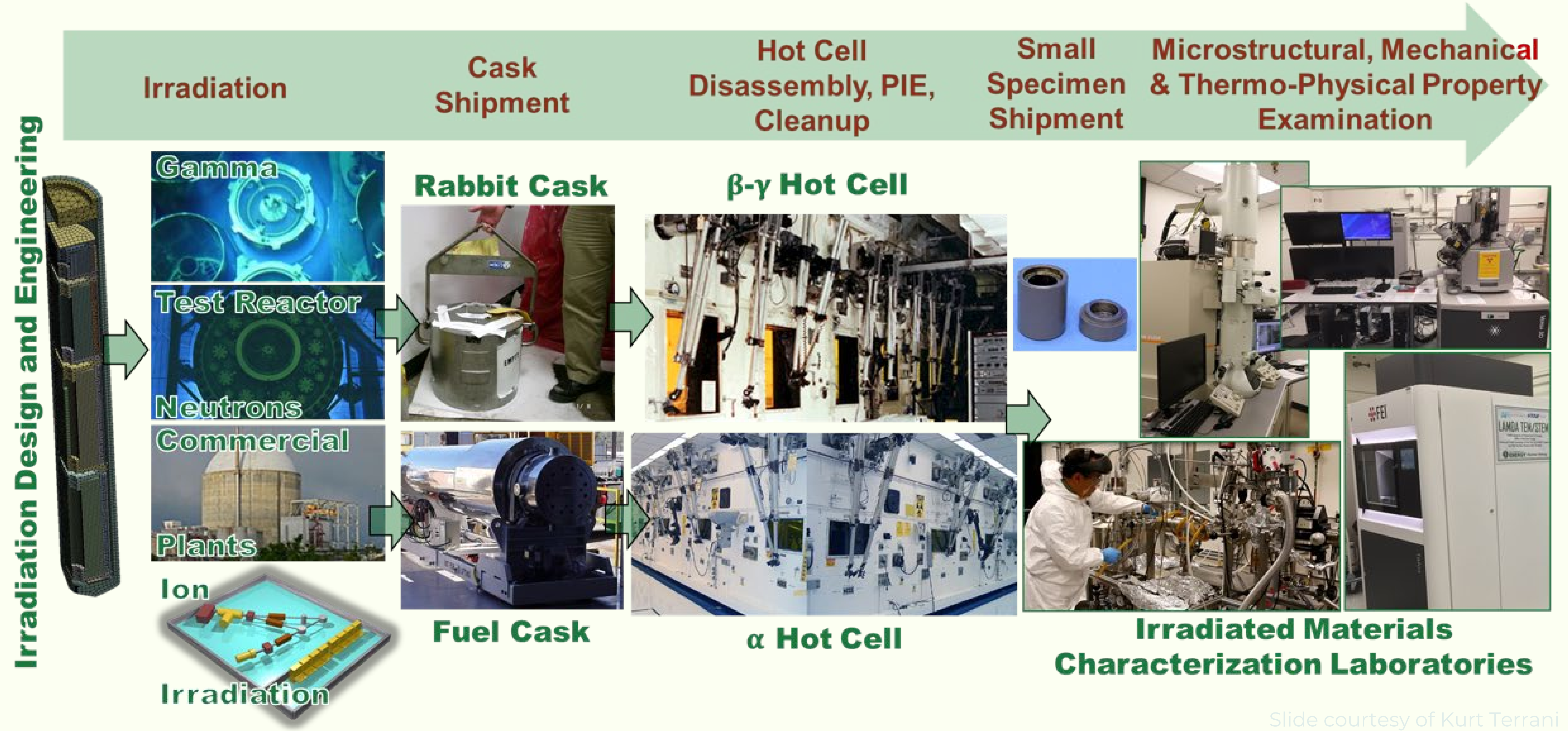
OAK RIDGE National Laboratory

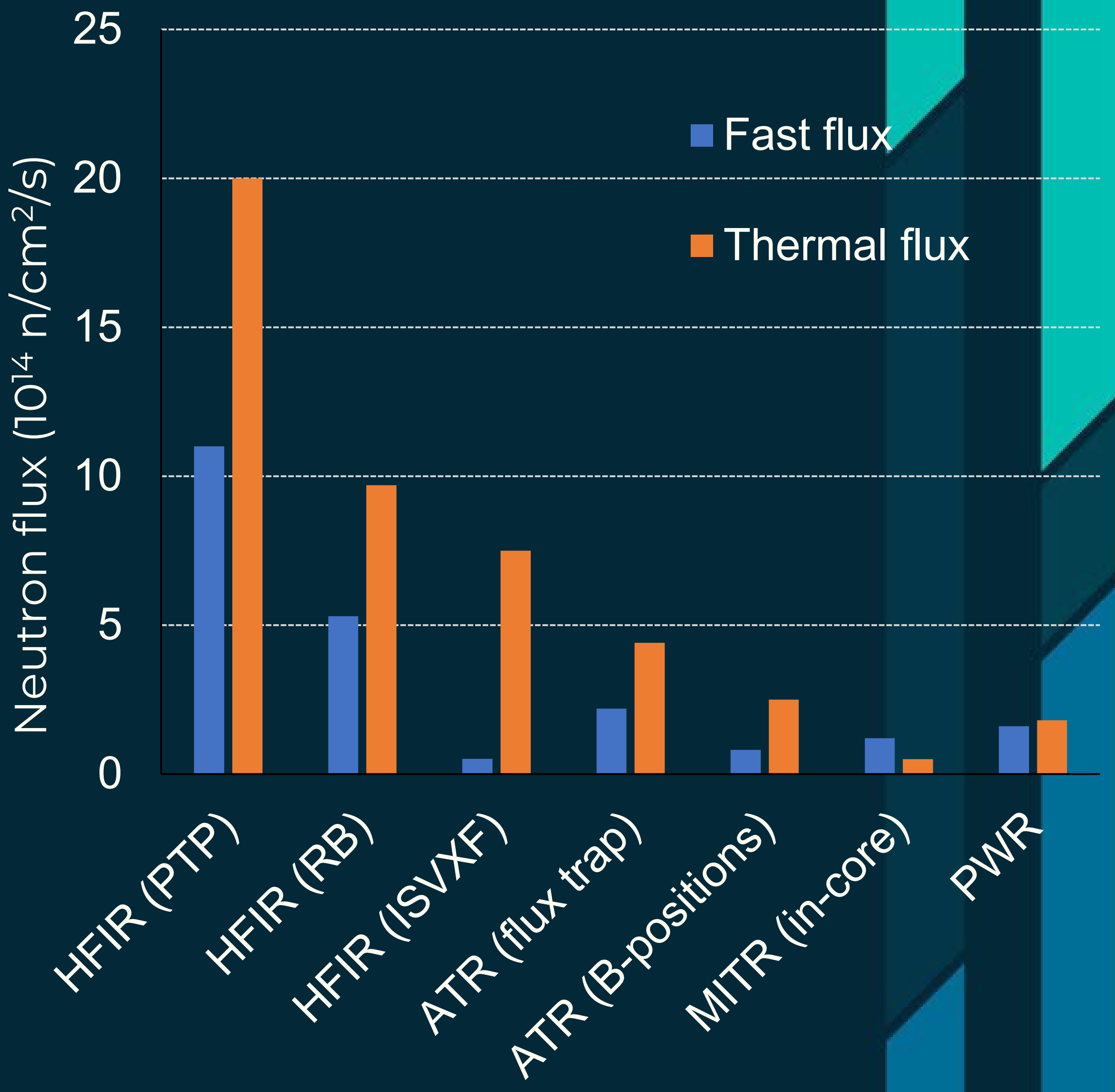
Comprehensive nuclear fuels and materials research at ORNL spans a chain of specialized facilities.



Slide courtesy of Kurt Terrani

High Flux Isotope Reactor (HFIR)

- Constant 85 MW reactor power
- 23–25 days/cycle, 6–8 cycles per year
  - Operating 40–50% of the calendar year
- ~50°C, ~3 MPa light-water coolant
  - Internal experiment temperatures can range from ~100 to >1000°C
- Highest steady-state neutron flux in the U.S.
  - Most economical option for many materials irradiations
  - End-of-life light-water reactor (LWR) dose (~20 dpa) or fuel burnup (~60 MWd/kg U) achievable in 1–2 calendar years
- Funded by the DOE Office of Science for neutron scattering experiments (beam lines)
  - Also supplies many of the world's radioisotopes
  - No neutron fees for fuel/material irradiations if the results are published



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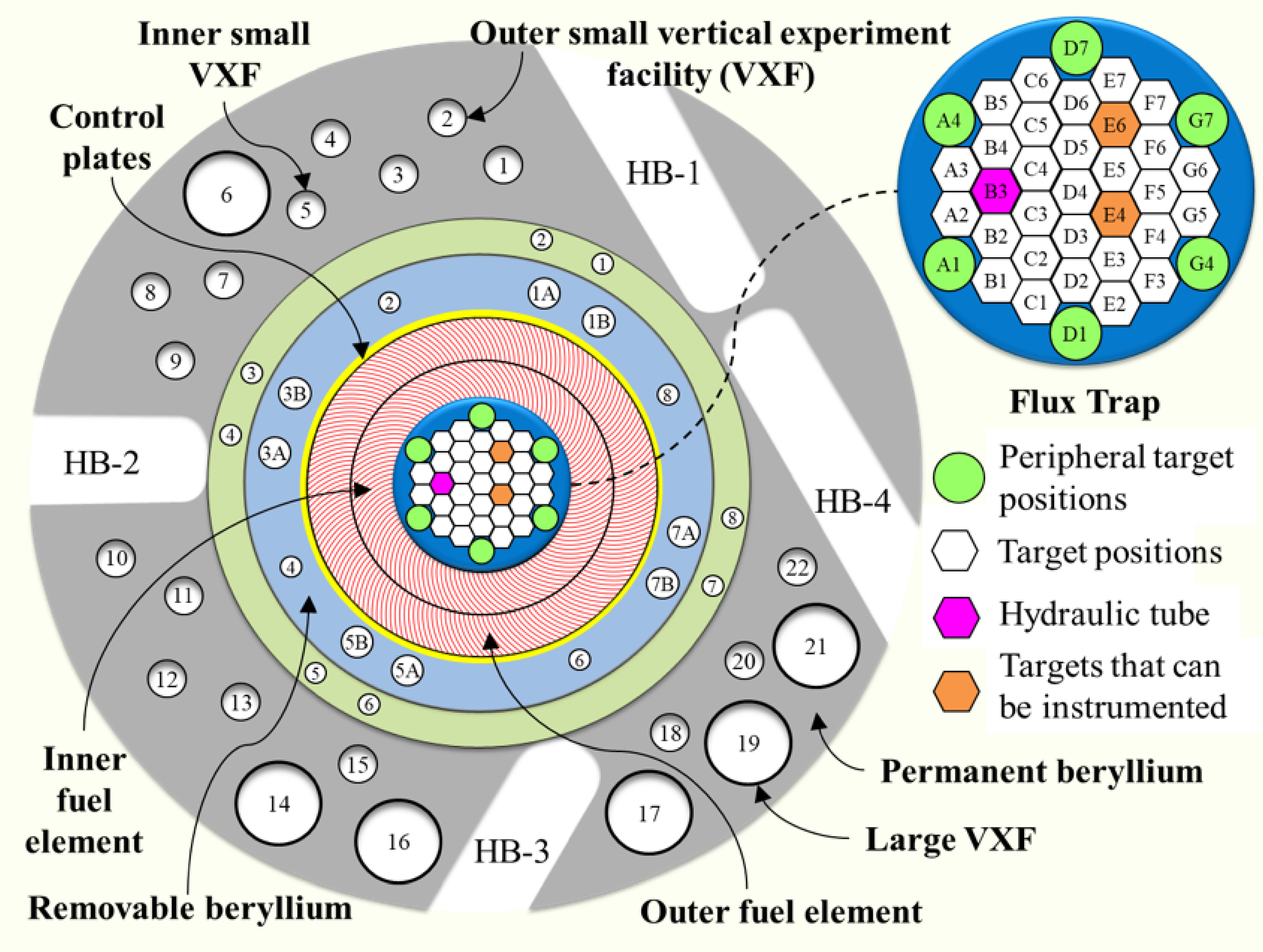
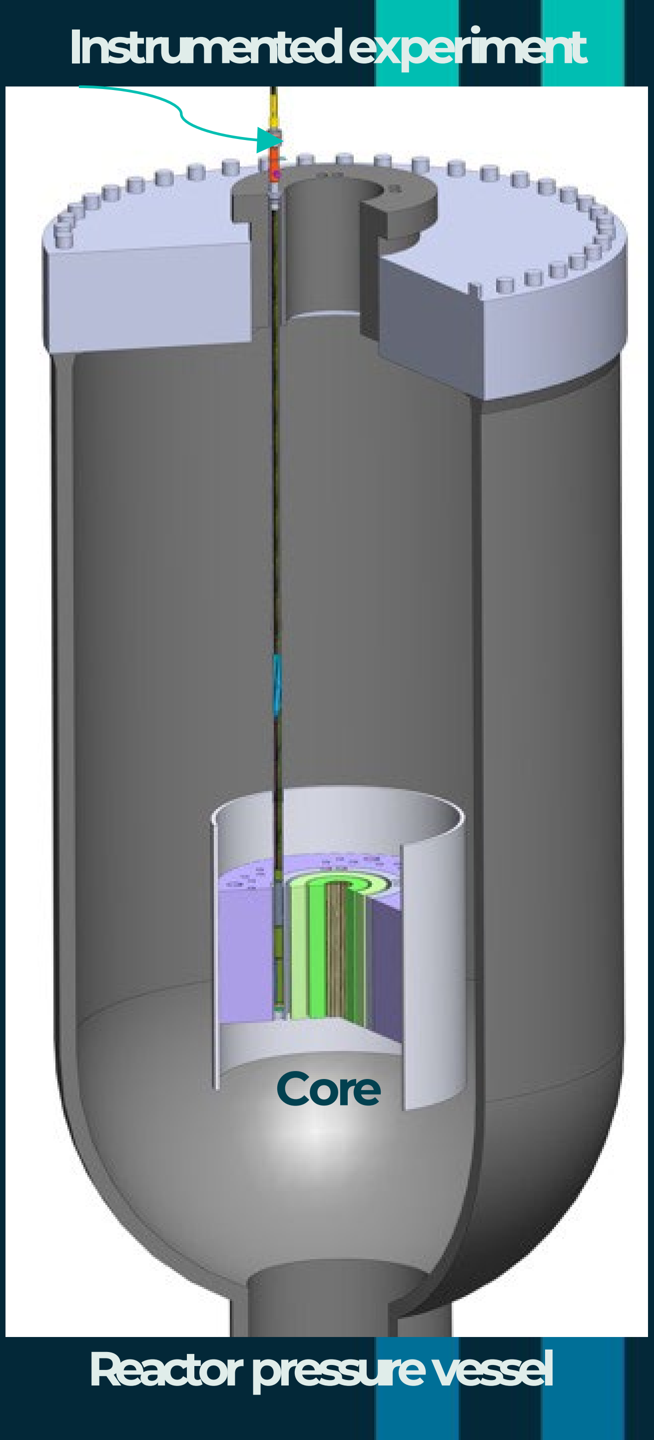
HFIR core layout





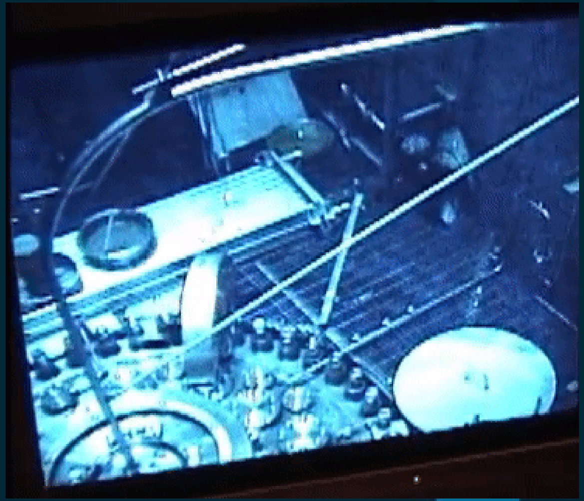
Flux trap: materials

- Small (Ø9.5 mm × 60 mm long) rabbit capsules
  - Sealed or perforated (in-coolant)
  - ~100 available positions
  - Partial cycles possible in hydraulic tube
- Larger full-length targets (Ø11.3 mm × 50 cm long)
- Highest flux
  - 1.1×10^{15} $n_{fast}/cm^2/s$
  - 2×10^{15} $n_{thermal}/cm^2/s$
  - 12.6 dpa/CY in steel
- **Routinely used to rapidly accumulate dpa in structural materials and assess changes in mechanical properties**

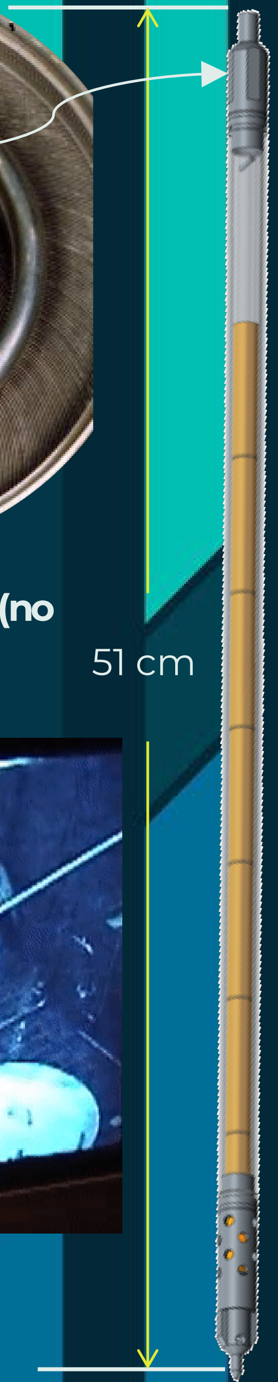
Metal tensile specimens

In-core liquid metal corrosion testing

HFIR fuel and flux trap (no hydraulic tube)

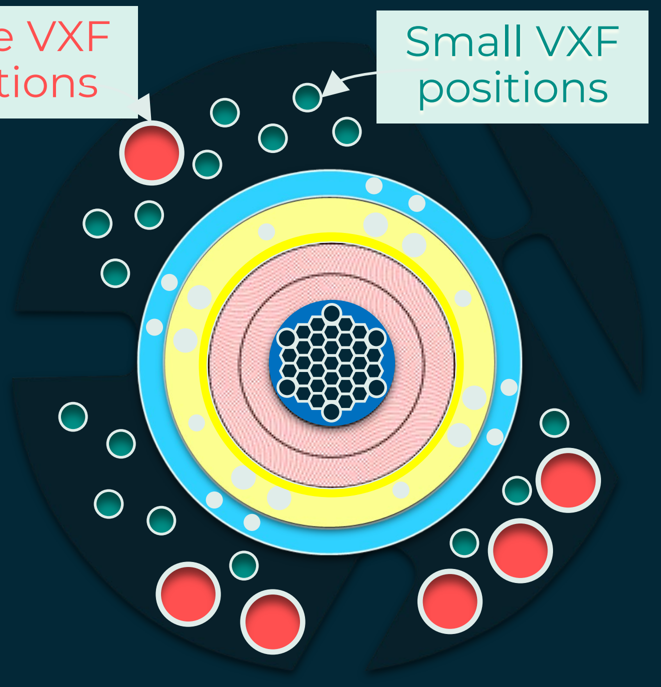


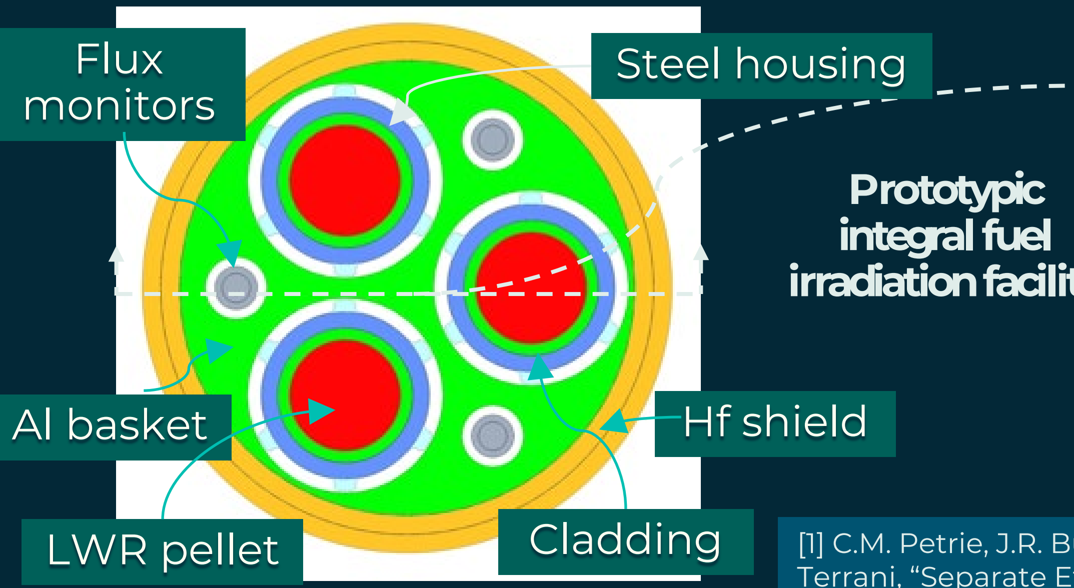
Rabbit ejection from hydraulic tube



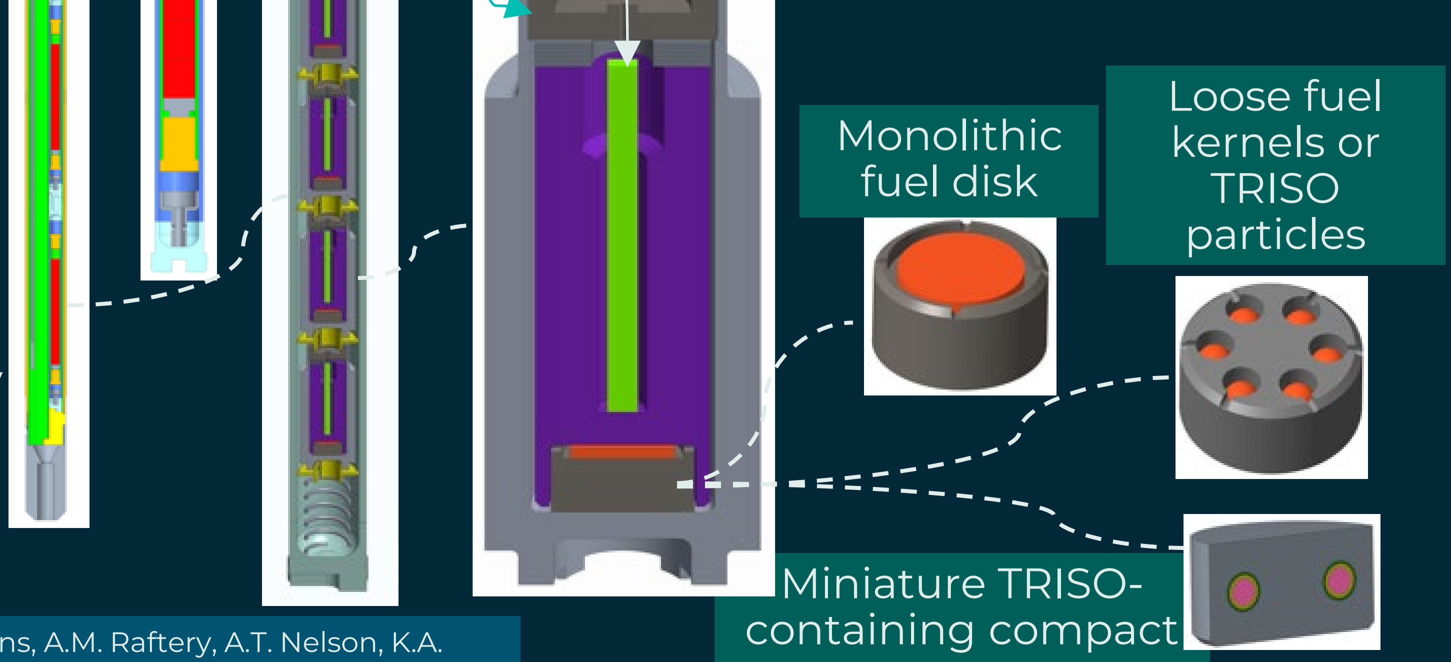
Permanent beryllium reflector: fuels

- **A high thermal flux is ideal for fuels testing**
  - 7.5×10^{14} $n_{thermal}/cm^2/s$
- Prototypic integral fuel tests
- Accelerated separate effects (isothermal) testing using MiniFuel
  - Rapid burnup accumulation in small samples to assess basic irradiation performance (swelling, fission gas release)
  - Up to 60 MWd/kg U (~6% FIMA) of burnup per year using natural uranium by leveraging high flux and equilibrium breeding/burning of ^{239}Pu





Prototypic integral fuel irradiation facility

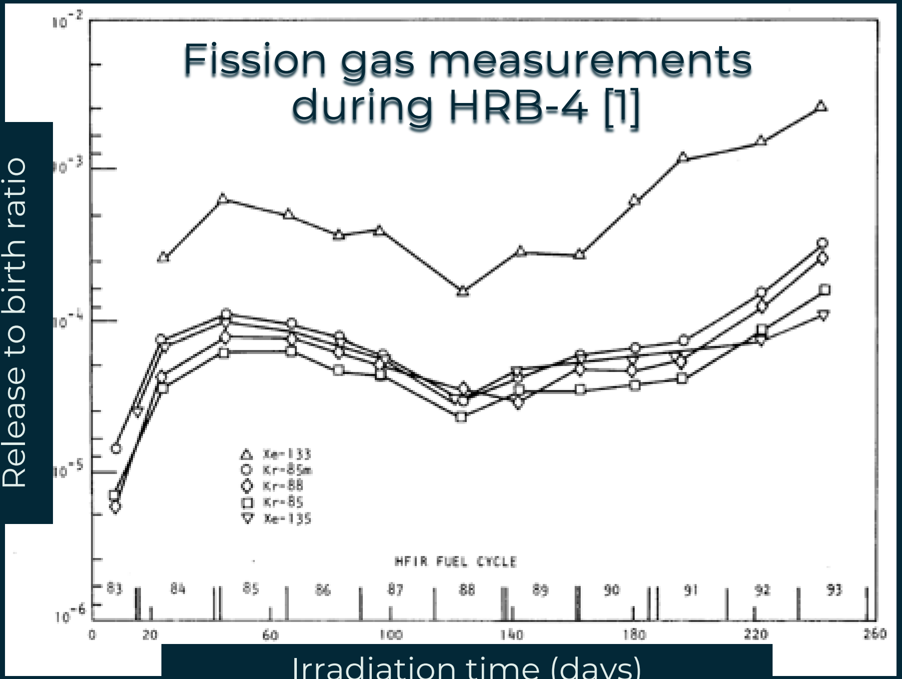


MiniFuel targets (no neutron shield) [1]

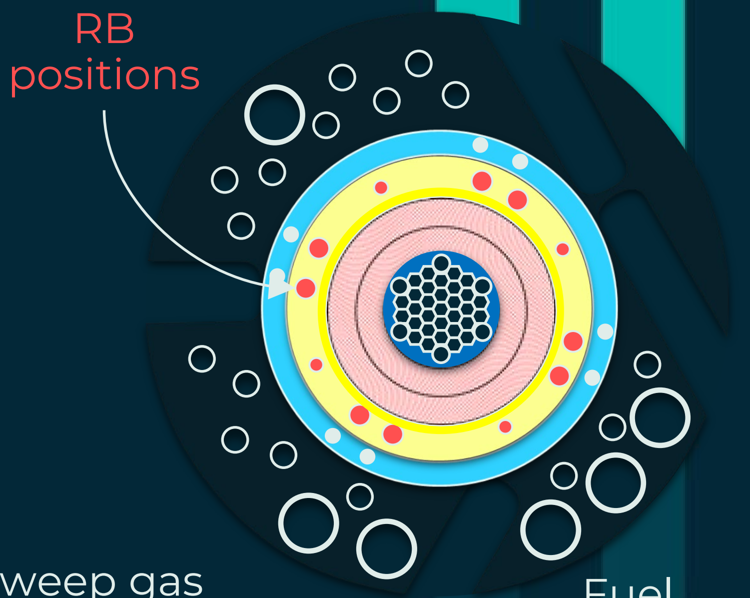
[1] C.M. Petrie, J.R. Burns, A.M. Raftery, A.T. Nelson, K.A. Terrani, "Separate Effects Irradiation Testing of Miniature Fuel Specimens," *J. Nucl. Mater.* **526** (2019) 151783.

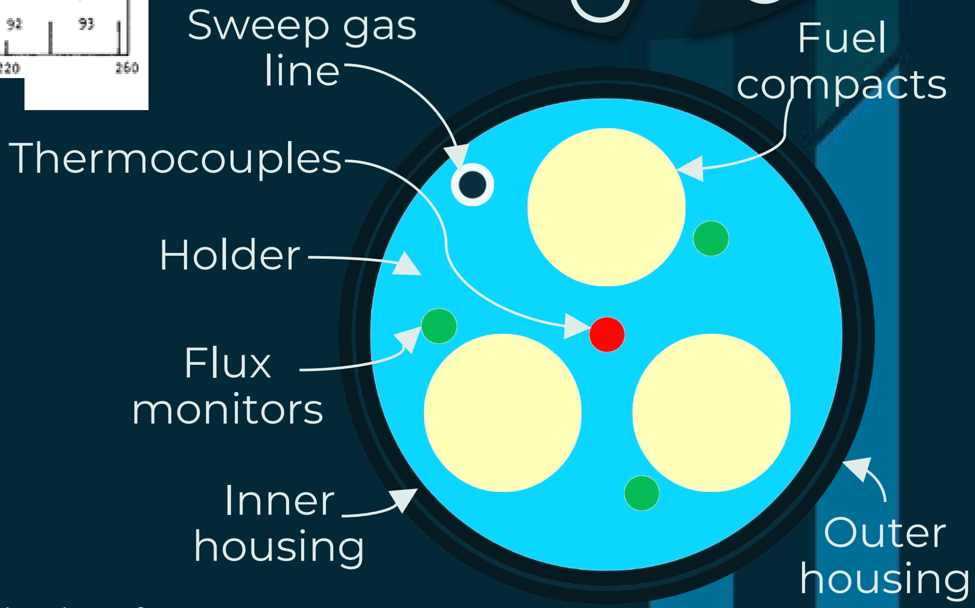
Removable beryllium (RB) reflector: fuels, materials, sensors

- Larger size (Ø37 mm)
  - Long (~4 m) instrumented experiments or smaller MiniFuel targets
- Direct access from the vessel head
  - **Ideal for instrumented fuels, materials, and sensor testing**
- High flux
  - 5×10^{14} $n_{fast}/cm^2/s$
  - 1×10^{15} $n_{thermal}/cm^2/s$
- >20 gas-cooled reactor fuel tests in the HFIR RB (HRB) series throughout the 1970, 80s, and 90s
  - BISO or TRISO fuel in graphite compacts
  - In situ monitoring of peak fuel temperature (>1000°C) and fission gas release



In situ fission gas monitoring system





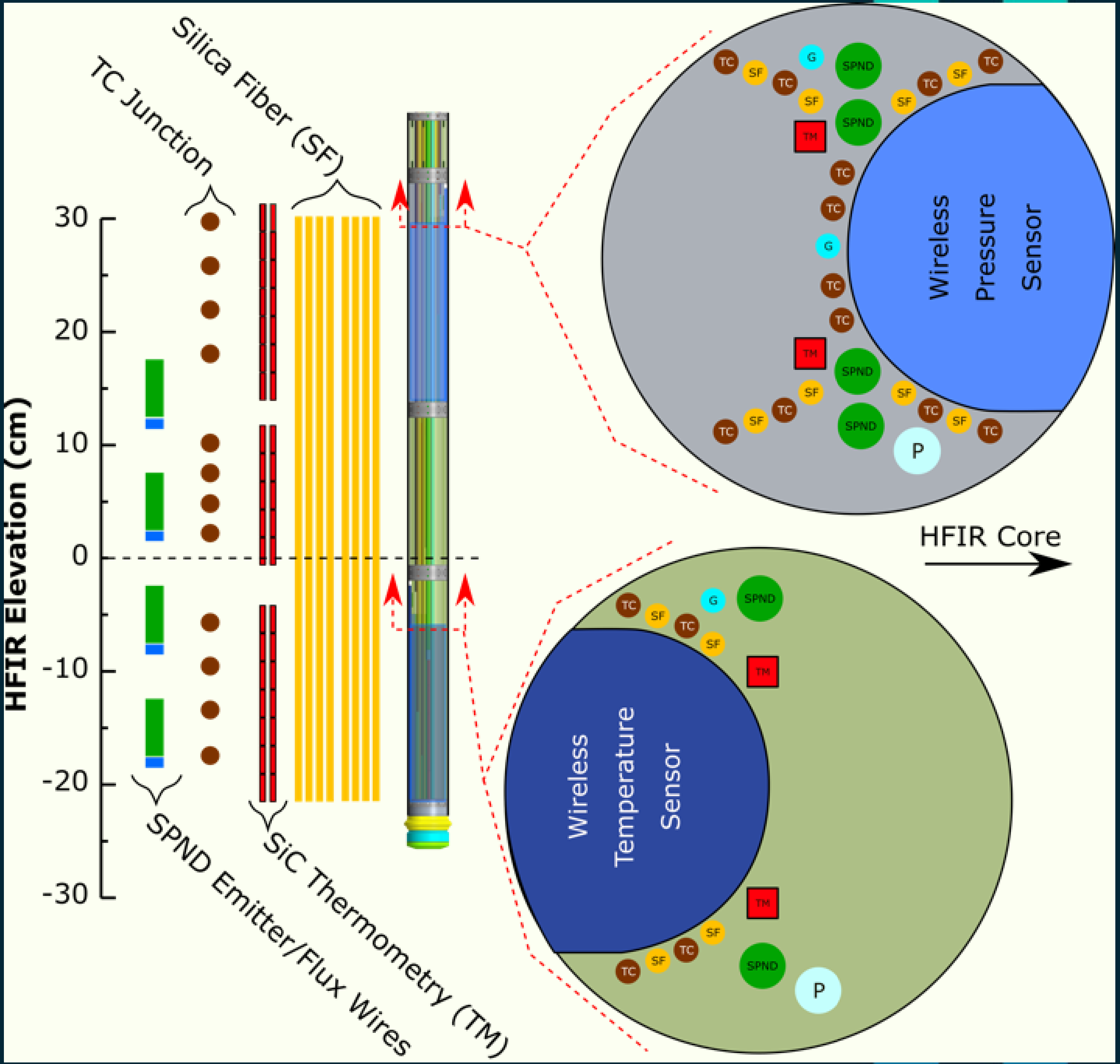
Typical HRB experiment configuration

[1] Scott, C. B., and D. P. Harmon, "Postirradiation examination of capsules HRB-4, HRB-5, and HRB-6," GA-A-13267, General Atomics, San Diego, CA (1975).

WIRE-21 RB experiment

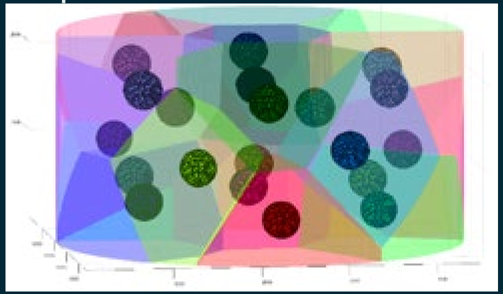
WIRE-21: Most highly instrumented experiment ever conducted in HFIR

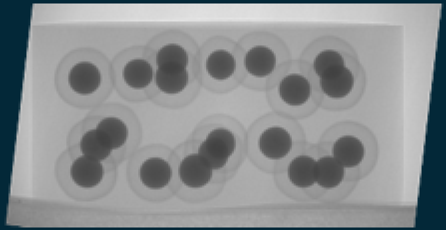
- Wireless temperature and pressure sensors
- Thermocouples
- Distributed fiber-optic temperature sensors
- Self-powered neutron detectors
- Passive SiC TMs and flux wires



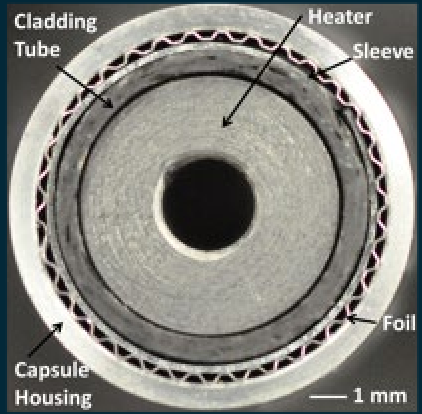
Examples of HFIR experiments supporting industry

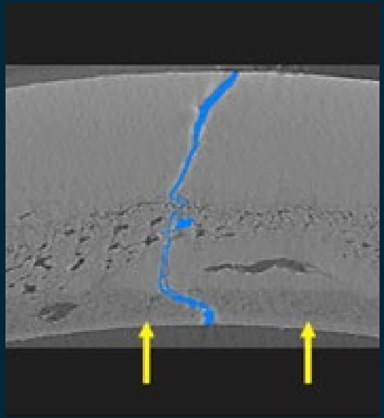
Kairos Power: High-power TRISO compact irradiations for their fluoride high-temperature reactor



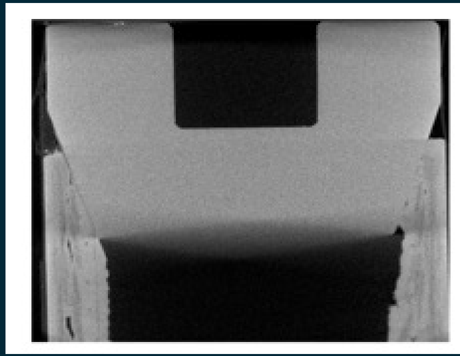


General Atomics: SiC/SiC composites





Cladding irradiation with radial heat flux



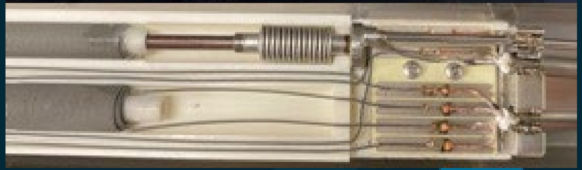
Post-irradiation cladding endplug

Bowing evaluation under flux gradients

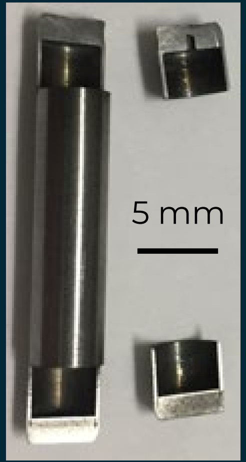
General Atomics: UC irradiations for their gas-cooled fast reactor

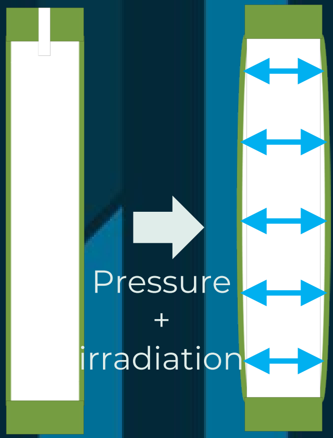
Framatome: Irradiation of coated Zr alloy cladding

Westinghouse: In situ irradiation of wireless fuel rod temperature and pressure sensors



General Electric: Irradiation of FeCrAl and coated Zr alloy pressurized creep tubes





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National Reactor Innovation Center

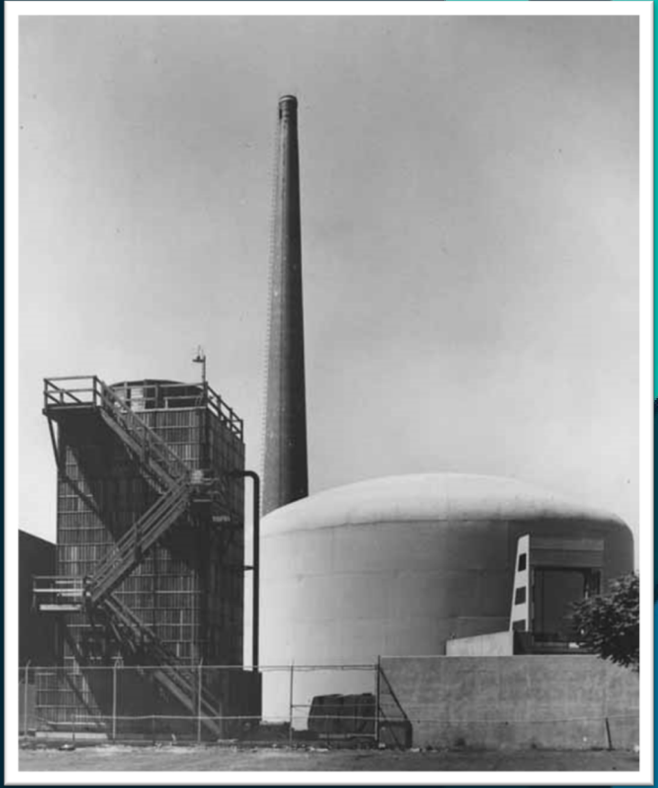
Massachusetts Institute of Technology

Advanced Reactor Irradiations in MITR

Gordon Kohse
kohse@mit.edu

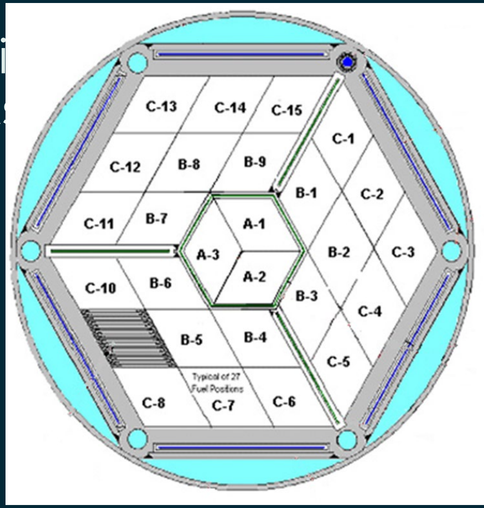
The MIT Research Reactor (MITR)

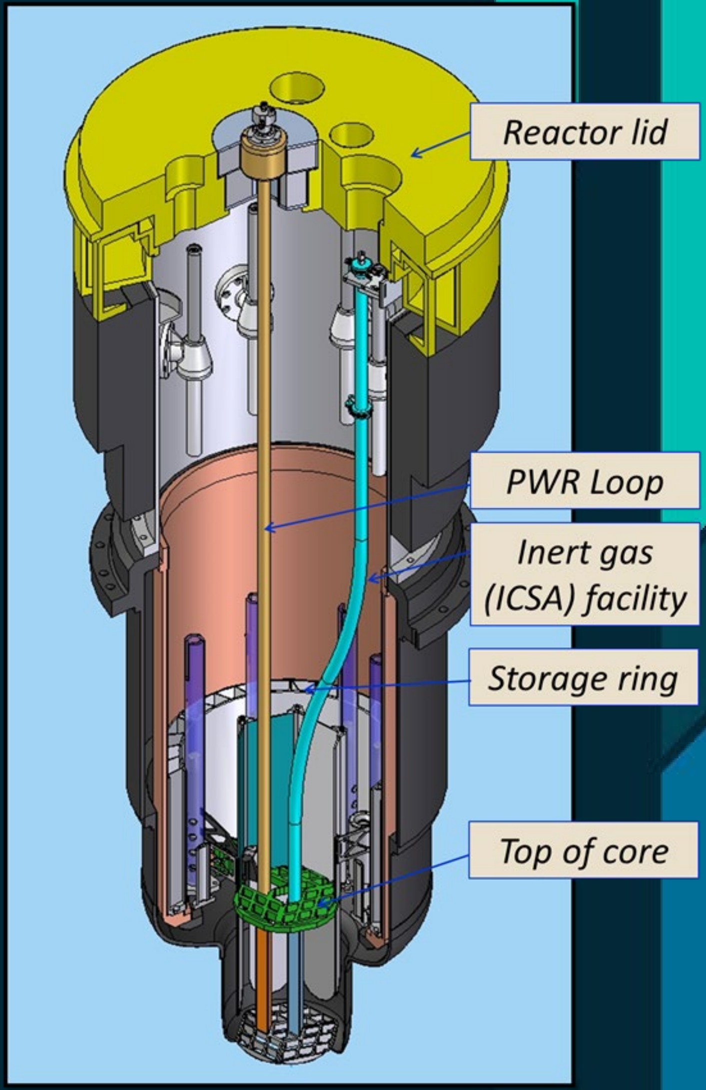
- 5 MW MITR-I first criticality in 1958, MITR-II redesigned core first criticality August 14, 1975, power increase to 6 MW in 2010
- Has provided neutrons for research in physics, medicine, materials science and nuclear engineering, along with opportunities for research and training of students
- Current focus is on irradiation effects in materials, fuels, and sensors for application in current and next generation reactors



MITR Basics

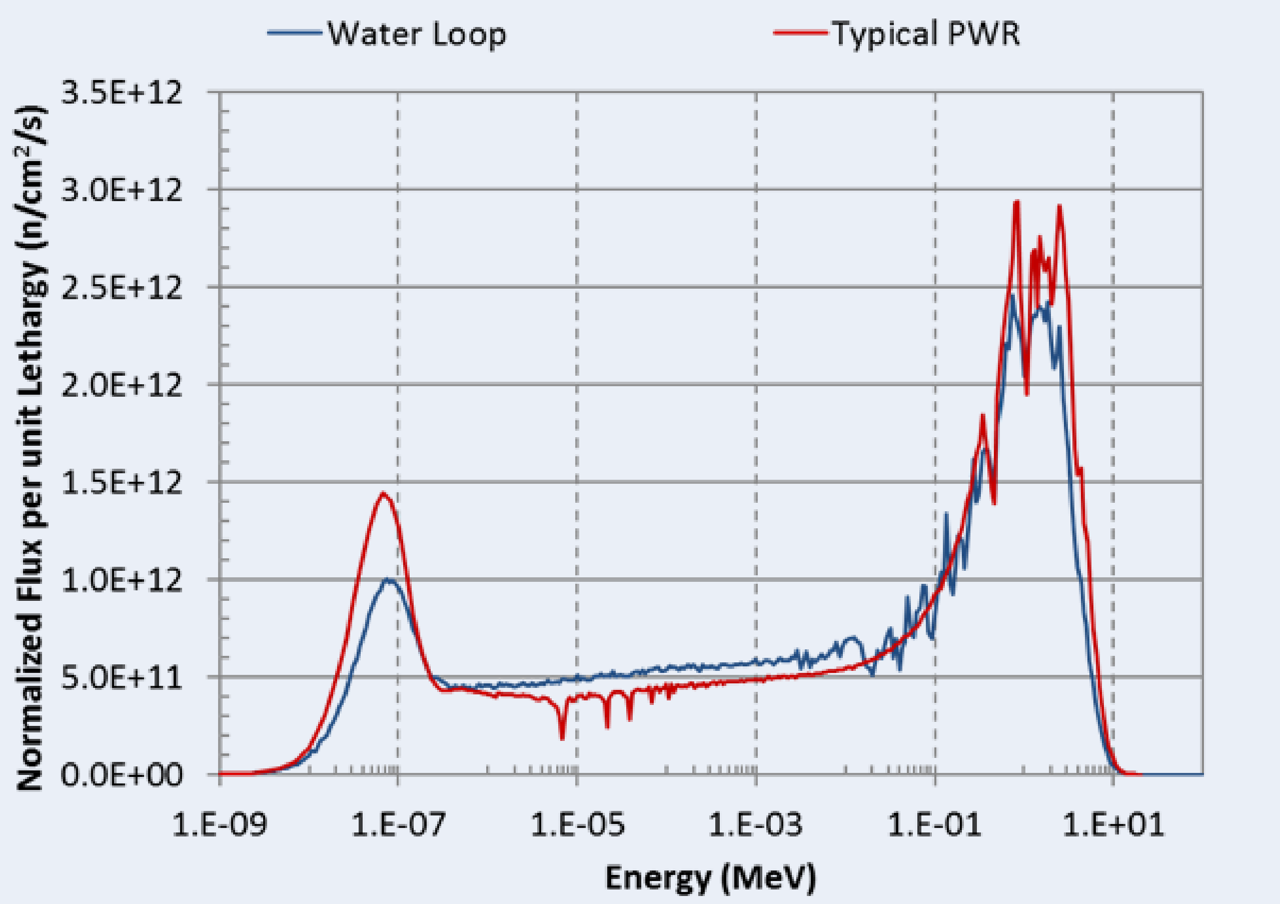
- Tank type reactor with power density approximately the same as a current generation light water power reactor (LWR)
  - Forced light water cooling at atmospheric pressure, core outlet ≈ 50 °C
  - Aluminum-clad plate type fuel
- 6 large boron-containing control blades on the outer surfaces of the hexagonal core, one cadmium-containing “regulating rod”
  - Cosine-shaped power distribution with peak near the core axial midline
- 60-70 day cycles (one per calendar quarter) wi
outages to refuel and reconfigure experiment
- 200-220 full power days per year





MITR Flux levels

Facility	Size	Neutron Flux (n/cm2-s)
In-core	3 available Max in-core volume ~ 1.8” ID x 24” long	Thermal: 3.6×10^{13} Fast: up to 1.2×10^{14} (E>0.1 MeV)
Beam ports	Various radial: 4” to 12” ID	Thermal: 1×10^{10} - 1×10^{13} (source)
Vertical irradiation position	2 vertical (3GV) 3” ID x 24” long	Thermal: 4×10^{12} - 1×10^{13}
Through ports	One 4” port (4TH) One 6” port (6TH).	Average thermal: 2.5×10^{12} to 5.5×10^{12}
Pneumatic Tubes	One 1” ID tube* (1PH1)	Thermal: up to 8×10^{12}
	One 2” ID tube* (2PH1)	Thermal: up to 5×10^{13}
Thermal Beam Facility (TNB)	Beam aperture ~ 6” ID	Thermal: up to 1×10^{10}



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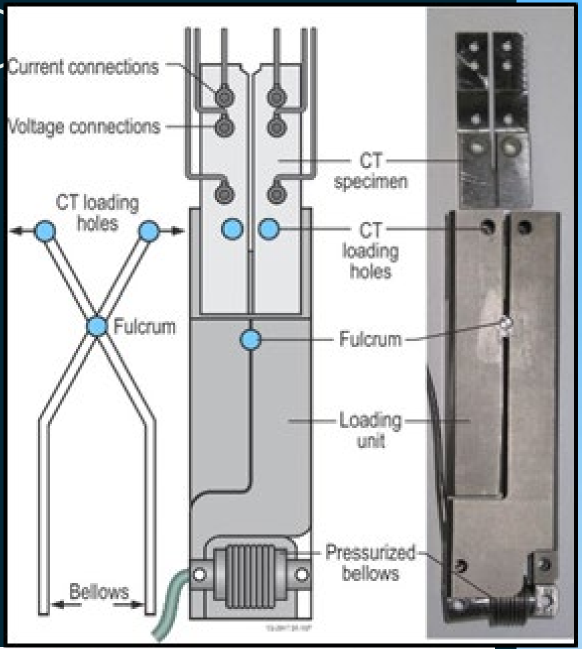
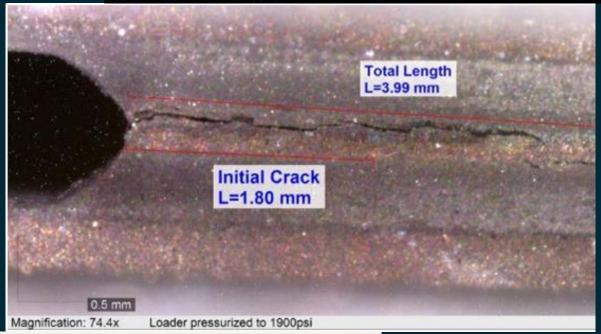
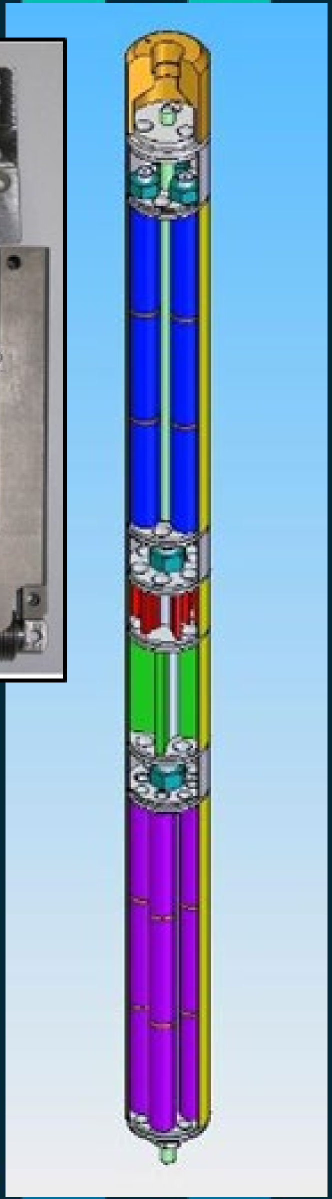
High Temperature Water Lo

- Major focus of the in-core irradiation program since the 1990s
  - High-temperature, high-pressure, titanium autoclave in the “B3” core position
  - Current max operating temperature 308 °C, upgrade planned to 340 °C, with flow velocities to several m/s
  - Sample volume available ≈ 30 mm diameter by 500 mm long
  - Range of water chemistries available: Li/B PWR, controlled oxygen and hydrogen BWR, non-standard chemistries
- Wide variety of sample materials and geometries can be irradiated, ≈ 1 dpa/yr in stainless steel
  - Fuel cladding materials including composites and coatings
  - Passively loaded creep tests and actively loaded crack growth tests
  - “Gamma susceptor” internally heated fuel clad test under development (note - fueled samples cannot be irradiated in the water loop)
- Sealed sample capsules can be used to provide well-controlled temperature without wetting samples

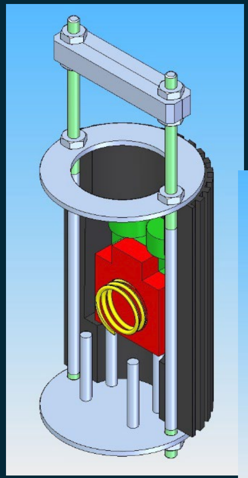


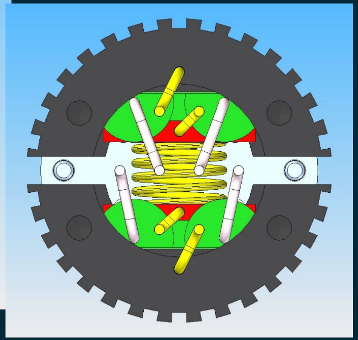


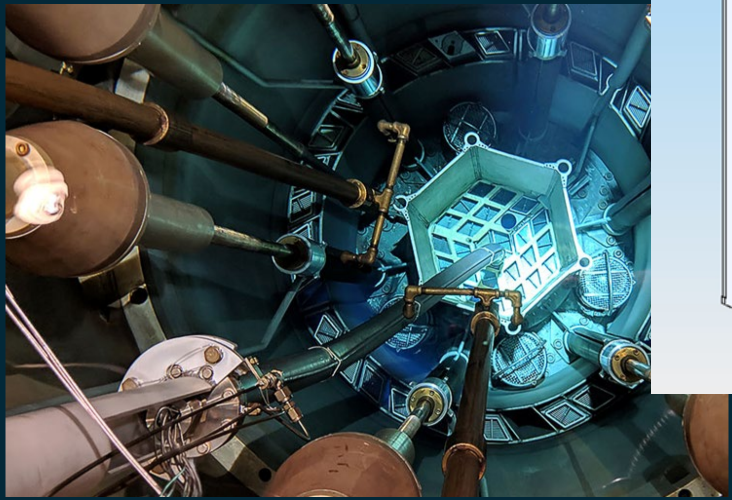


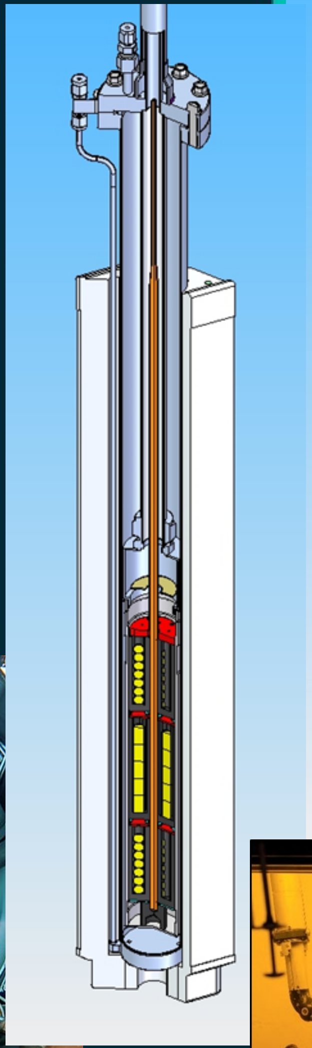
Controlled Temperature Inert Gas Facilities

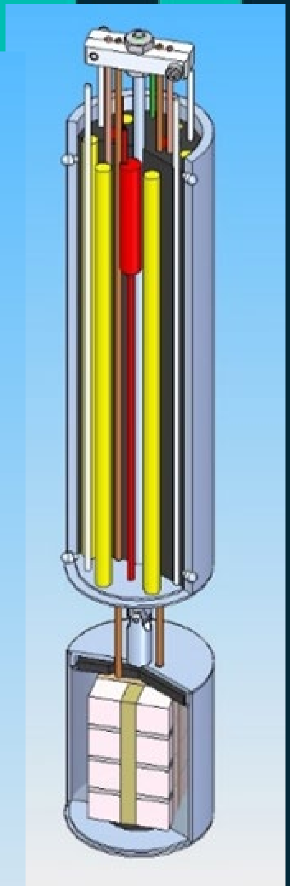
- Instrumented with thermocouples for temperature control by He/Ne gas mixture with additional electric heating if necessary
- Standard temperature range 300-900 °C, specialized versions have operated to 1400 °C
- Highlights of previous irradiations
  - Flibe salt irradiations at 700 °C
  - Ultrasonic, fiber optic, and radiation sensor testing
  - Instrumented thermo-electric generator tests

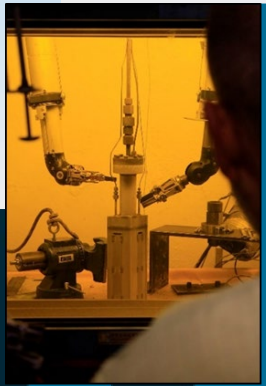






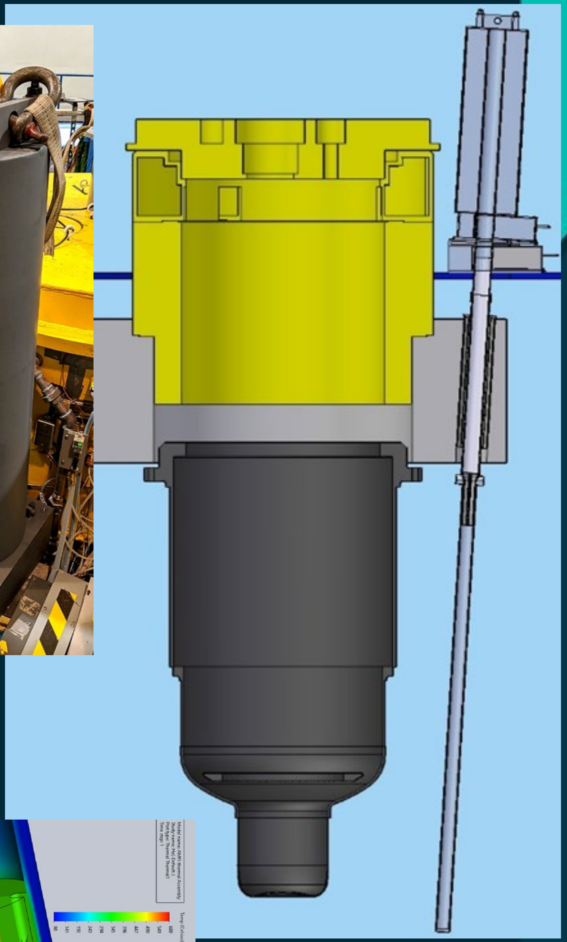


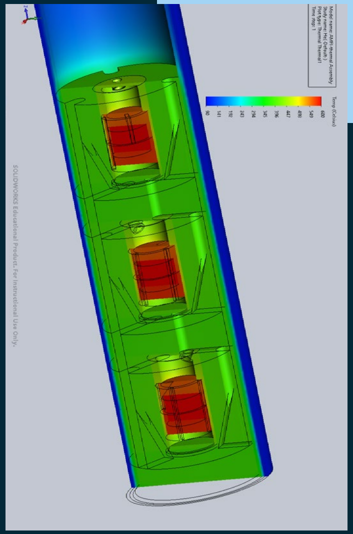


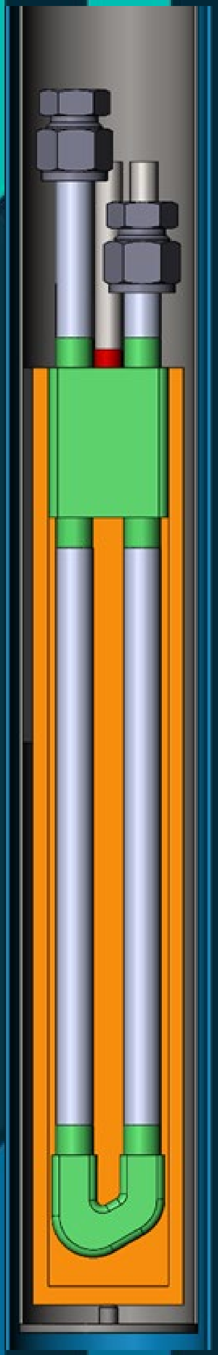


Graphite Reflector Positio

- Lower neutron flux than in-core, mostly thermal
  - Easier access and somewhat larger than in-core positions, irradiations less than a full cycle are possible
  - Fewer constraints on fuel irradiations than in-core (e.g. fueled molten salt, ramp test under development for NASA nuclear thermal propulsion program)
  - Good option if neutron activation is important but fast neutron damage isn't necessary
- One position currently available with another under development
- Recent and planned irradiations
  - Irradiations for ORNL additive manufactured fuel
  - Tritium permeation tests for fluoride-salt-cooled reactor
  - Natural convection molten salt loop to be irradiated next reactor cycle







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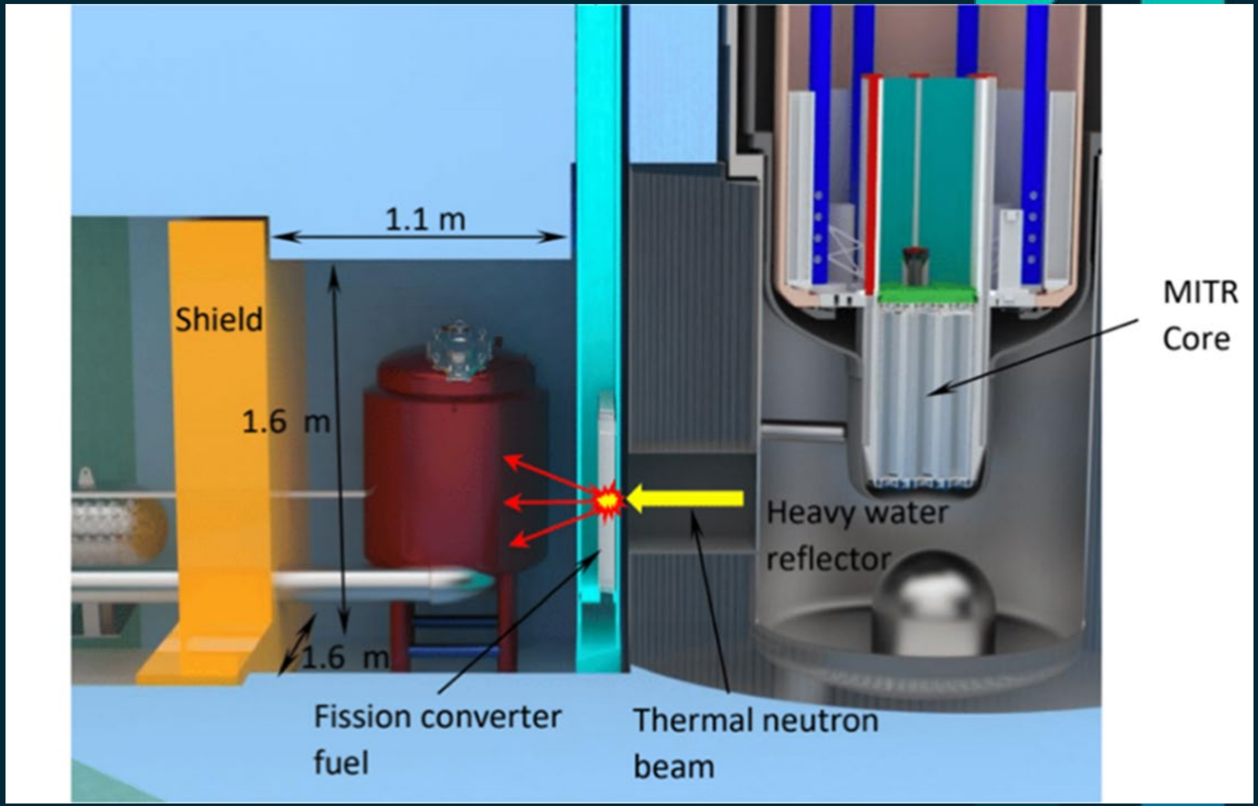
The MCube Large Volume Irradiation Space

- New irradiation facility under development, completion expected in 2023
  - Uses thermal neutrons and a sub-critical array of MITR fuel elements to produce significant fast flux in a large irradiation space
  - Neutron flux is approximately 2 orders of magnitude less than in-core, but significant fast flux is available (contrast to 3GV positions)
  - Fewer constraints on fuel irradiations and flux tailoring than in-core
  - Shielded space available for support systems
- First experiments will be a forced-convection, fueled molten salt loop and a fusion reactor magnet irradiation



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NRIC TECH TALKS

Q&A

NRIC TECH TALKS

Thank you!

All proceedings from this webinar will be posted under Resources on the NRIC website.

Contact: NRIC@inl.gov

Website: nric.inl.gov

Backup Slides

INL capabilities to fabricate and examine irradiation testing specimens

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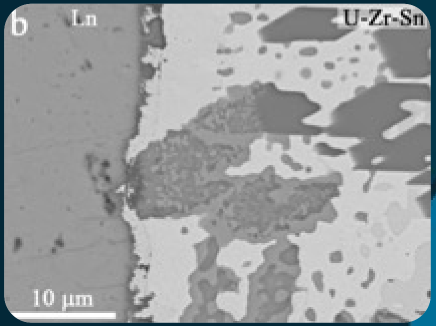
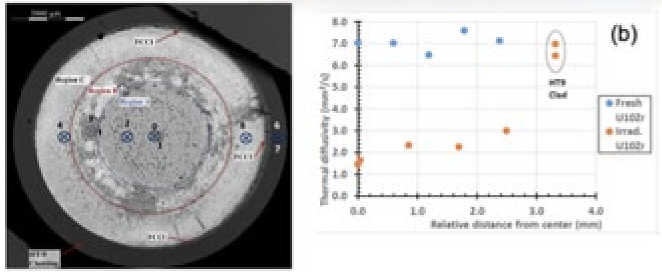
INL Fuel Specimen Fabrication

- Expertise in specification, contracting, and inspection of fuel specimens
  - Established relationships with BWXT and ORNL for TRISO specimens
- Hazcat III fuel fabrication facilities (<700 g ^{235}U)
  - **Ceramic/intermetallic fuels**
    - Lab-scale synthesis, direct metal nitridation, direct melting for carbide and silicide
    - Various presses, mills, and controlled-atmosphere furnaces for conventional powder processing, sintering, and centerless grinding
    - Spark plasma sintering capability
  - **Metallic fuels**
    - Various alloying, casting, and post-machining capabilities
    - Extrusion capabilities
  - **Cladding and assembly**
    - Pressure resistance, laser, and TIG end cap welding
    - Some experience with the assembly of ceramic cladding (SiC)
    - Sodium bonding capability
- Hazcat II fuel fabrication facilities (<700 g ^{235}U and transuranic)
  - Similar to the list above, except that transuranic gloveboxes constraints typically limit the specimen size (~25 cm length)
- Fresh-fuel characterization capabilities
  - Microstructure: SEM/TEM, EPMA, APT, XRD
  - Thermal characterization: DSC, TGA, dilatometry, laser flash diffusivity

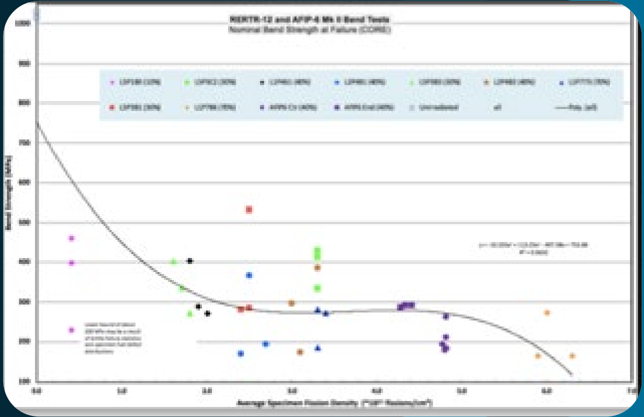


Shielded Exams and Testing

- ATR located ~25 km from post-irradiation exam facilities at the Materials and Fuels Complex
- Engineering-scale exams in the Hot Fuel Exam Facility (HFEF)
  - Metrology down to optical metallography
  - Fuel rod puncture and fission gas release analysis
  - Neutron radiography and re-irradiation for short-lived isotopes
  - Furnace for TRISO high-temp accident with real-time fission product release testing
  - Mechanical properties testing
  - Sizing, instrumentation, assembly, pressurization, and seal welding specimens for testing in furnaces, TREAT, and ATR
- Microstructural exams in the Irradiated Materials Characterization Laboratory
  - Microstructure, properties, etc.
  - Plasma FIB, EPMA, SEM, TEM, thermal conductivity microscope



(a) SEM image of a polished section of the irradiated U-10 wt.% Zr fuel pin. The TCM measurement locations on the profile of the pin are marked along with the measurement number.



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Legacy Materials

- Legacy materials originate from a variety of programs, including historical fast reactor programs as well as space program for UN
  - Some material is from small rodlets from the last ~15 years of DOE fuels research programs
  - U-Zr, U-Pu-Zr, MA-bearing, nitrides
- EBR-II/FFTF fuels and materials stored at INL (RSWF facility)
  - U-Fs, U-Zr, U-Pu-Zr
  - MOX
  - UN, UC
- Includes a variety of stainless-steel cladding alloys, including austenitic, ferritic/martensitic, and ODS materials

INL facilities at the Materials and Fuels Complex

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