The Advanced Test Reactor Capabilities Overview

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Idaho National Laboratory

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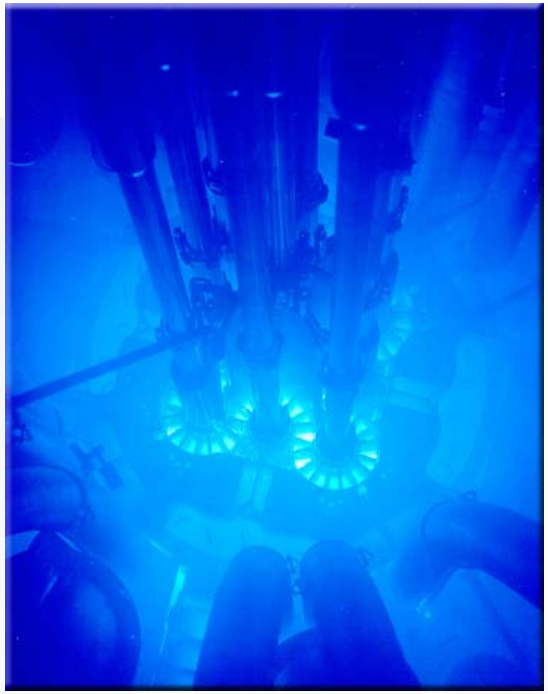
Outline of Presentation

- Advanced Test Reactor Description
- Unique Design Features
- ATR Test Configurations
- Experiment Programs
- Post Irradiation Examination Capabilities
- National Scientific User Facility



ATR Historical Perspective

Materials Test Reactor (MTR)

- 1952 through the early 1970's
- First of its kind to study material behavior in a radiation field

Engineering Test Reactor (ETR)

- 1958 through the early 1980's
- Studied fuel performance and reactor components, including sodium reactor experiments

Advanced Test Reactor (ATR)

- Initial operation in 1967 – continuous operation until present
- Fuels and materials development for the Naval Nuclear Propulsion Program, the Department of Energy, and others

ATR Description

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Reactor Type

Pressurized, light-water moderated and cooled; beryllium reflector

250 MWt design

Reactor Vessel

Stainless steel

12 ft (3.65 m) diameter

36 ft (10.67 m) high

Maximum Flux, at 250 MW

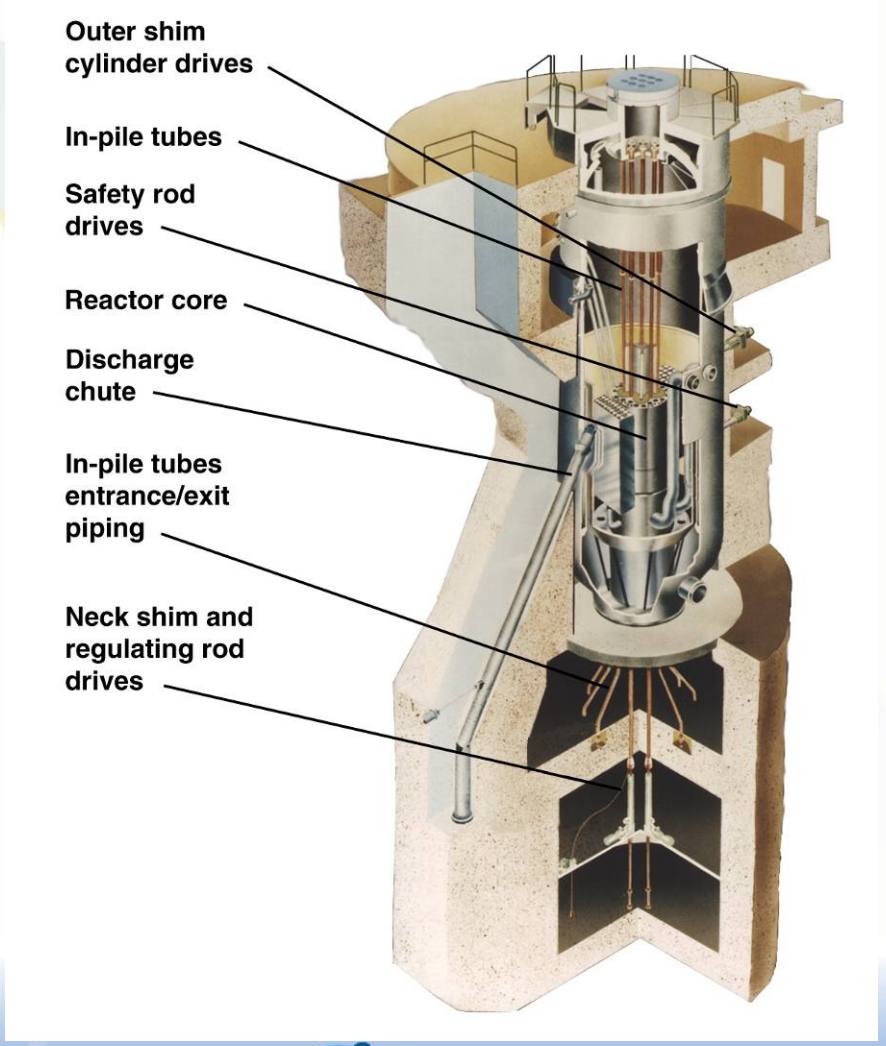
1×10^{15} n/cm^2-sec thermal

5×10^{14} n/cm^2-sec fast (E>1.0MeV)

Reactor Core

40 fuel assemblies

U-Al plates – 19/assembly



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Comparison between ATR and Power PWR

Reactor Features	ATR	PWR (Typical)
Power (MW_{th})	250 (Max Design)	~ 3,800
Operating Pressure (psig)	~ 355	~ 2235
Inlet Temp. (F)	~ 125	~550
Outlet Temp. (F)	~ 160	~620
Power Density (kW/ft^3)	~ 28,300	~ 2,800
Fuel	Enriched U-235	3 – 4 % U-235
Fuel Temp. (F)	~ 462	> 1000

ATR Core Cross Section

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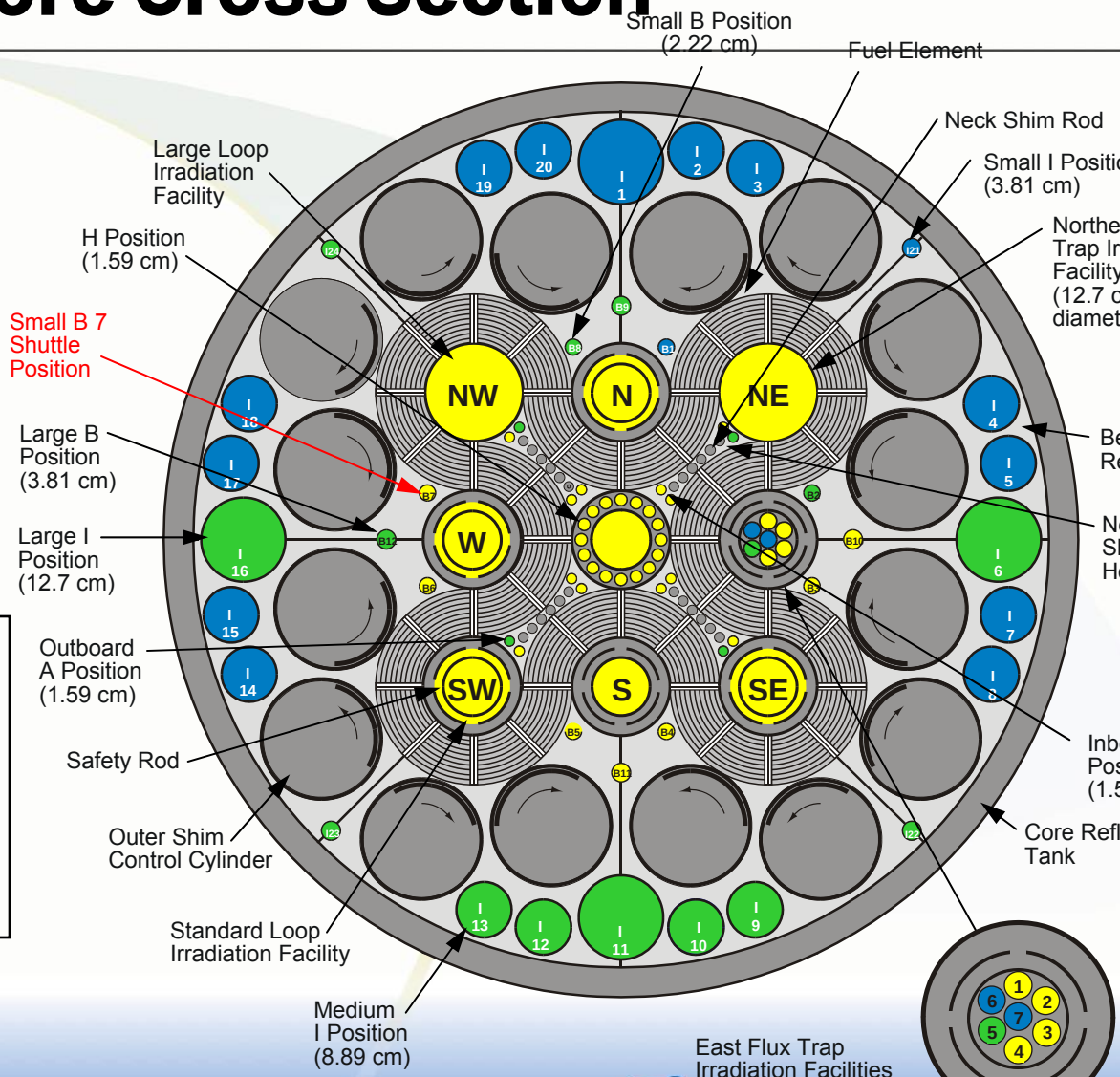
77 irradiation positions
- 4 flux traps
- 5 in-pile tubes
- 68 in reflector

Diameters: 0.5” – 5.0”
All positions 48” height

- Currently in use
- Planned use in 2010 and 2011
- Currently unplanned (or until 2012)



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Unique Design Features

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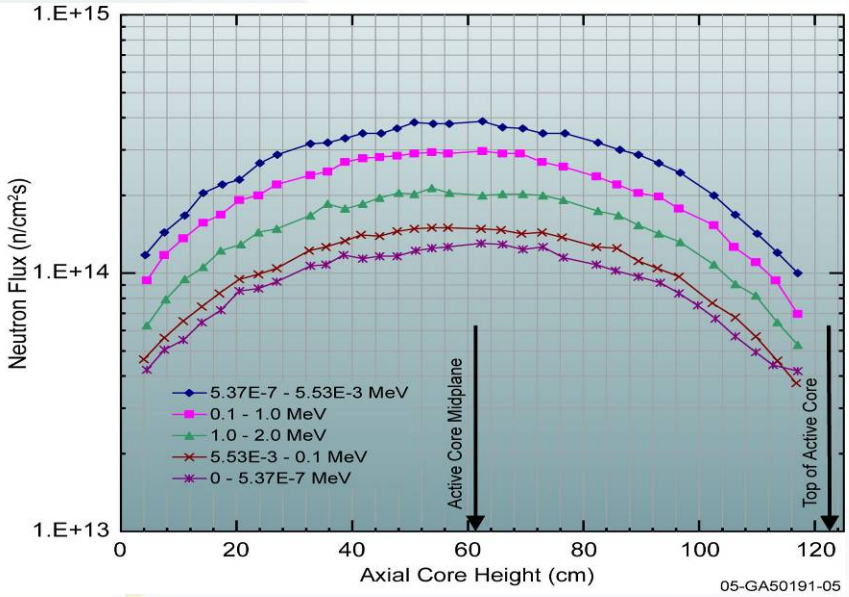
- Combination of high flux and large test volumes
- Symmetrical axial power profile
- Power tilt capability between all four corners of core (≤3:1 ratio)
- Individual experiment parameter control for multiple tests in a single irradiation position
- Individual experiment control in separate loops
- Accelerated testing for fuels – up to 20x actual operation time for some fuel types
- No design limited lifetime: expected to operate for many more years
  - CIC outages – new reactor internals
  - Large stainless steel reactor vessel – minimal embrittlement



Center Flux Trap Flux Profile (125 MW)

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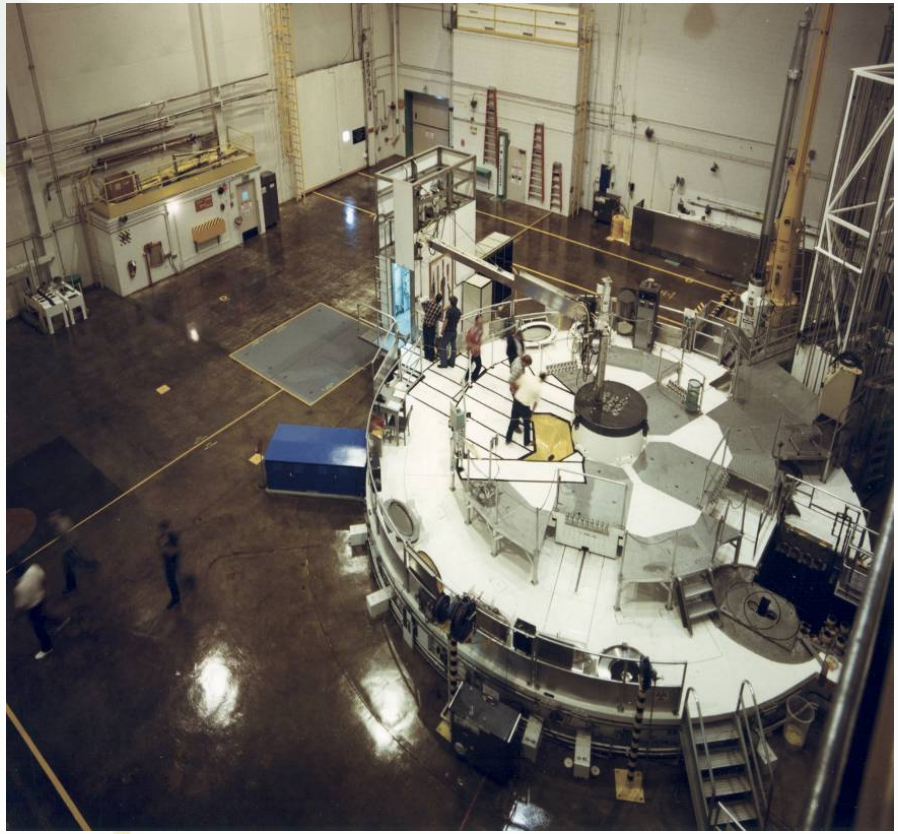
ATR Operations

- Operating Cycles
  - Standard operating cycle is 6 to 8 weeks
  - Occasionally short high power cycles of 2 weeks
  - Standard reactor outages are 2 weeks
  - Operations for approximately 250 days per year
- Core Internals Changeout (CIC), every 7 to 10 years
- ATR Critical Facility – used for reactivity measurements



Experiment Configurations

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Simple Static Capsules

- Reflector positions or flux traps
- Isotopes, structural materials, fuel coupons or pellets

Instrumented Lead Experiments

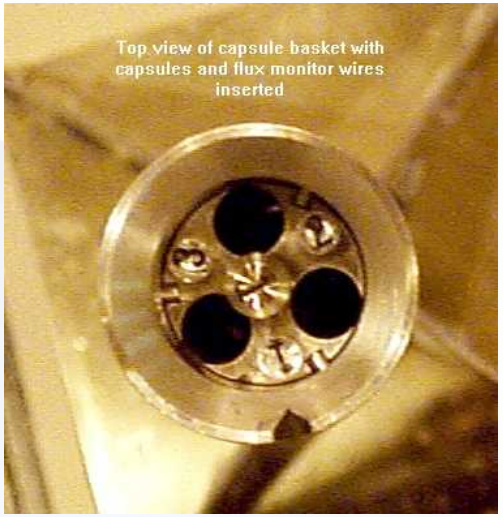
- On-line experiment measurements
- With or without temperature control
- Structural materials, cladding, fuel pins

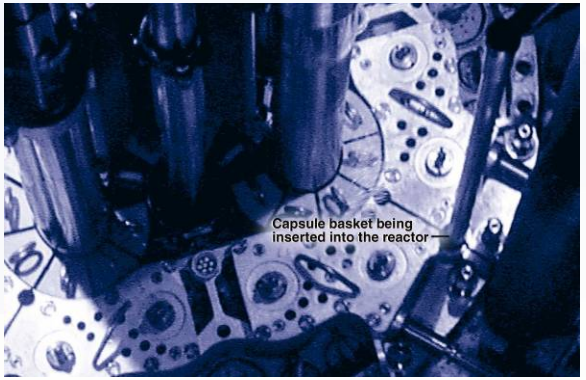
Pressurized Water Loops

- Five presently installed in flux traps
- Control pressure, temperature, chemistry
- Structural materials, cladding, tubing, fuel assemblies

Hydraulic Shuttle Irradiation System

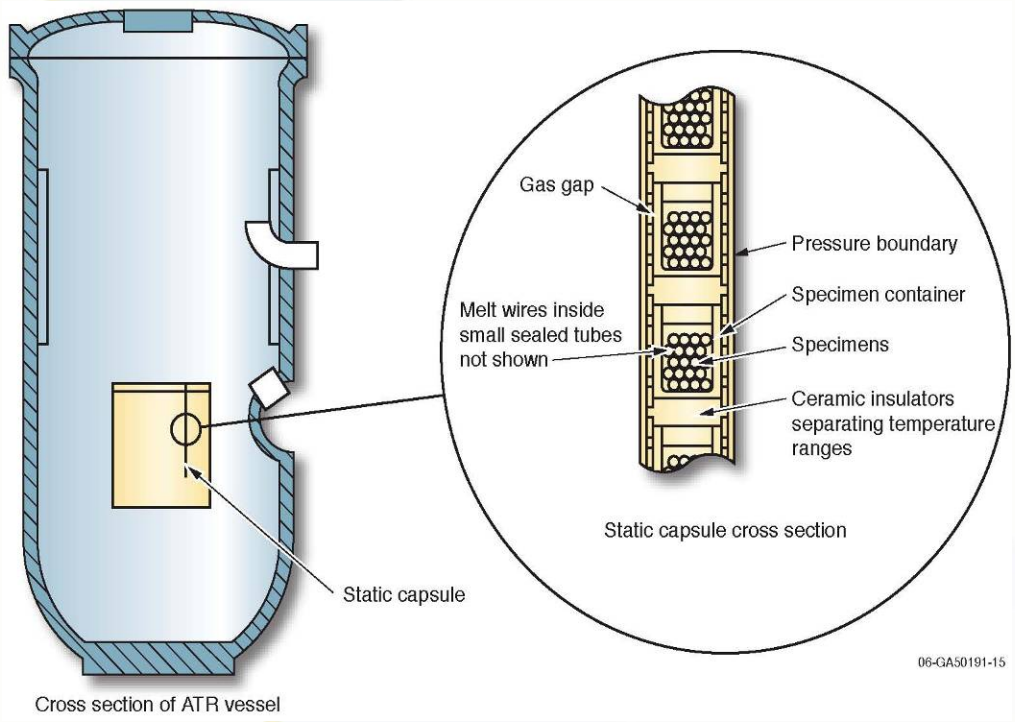
- Insert and remove experiments during operation





Simple Static Capsule Experiments

- Passive instrumentation (flux wires, melt wires)
- Enclosed in sealed tube, or fuel plates
- Temperature target controlled by varying gas mixture in conduction gap and with material selection; cooled by ATR primary coolant
- Lengths up to 48”; diameter 0.5” – 5.0”
- Used for isotope production, fuel and material testing



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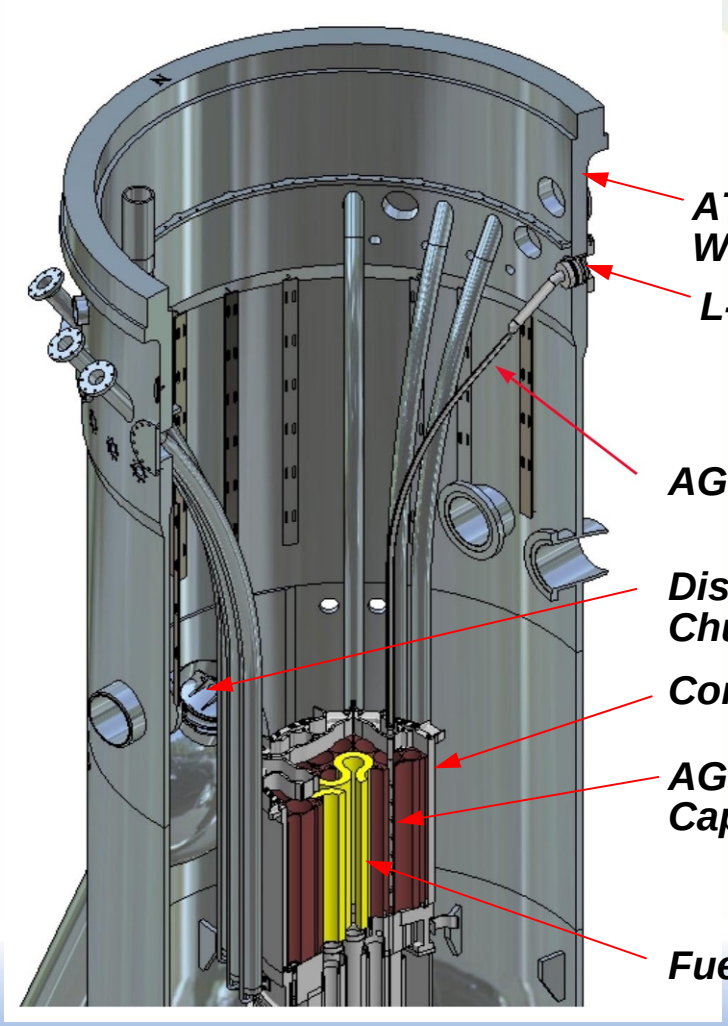
Instrumented Lead Experiments



- On-line experiment measurements
- Temperature control range 250-1300°C, within +/- 5°C
- Monitoring of temperature control exhaust gases for experiment performance (e.g., fission products, leaking materials, etc.)
- Specialized gas environments (oxidized, inert, etc.)

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Pressurized Water Loop Tests

- Five flux trap positions currently have pressurized water in-pile loop tests (1 large diameter, 4 small diameter)
- Separate from ATR primary coolant system
- Each loop has its own temperature, pressure, flow, and chemistry control systems – can exceed current PWR operating conditions
- Transient testing capabilities (cycle/seconds)
- Potentially feasible to simulate boiling water reactor void conditions

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ATR Standard Loop Layout

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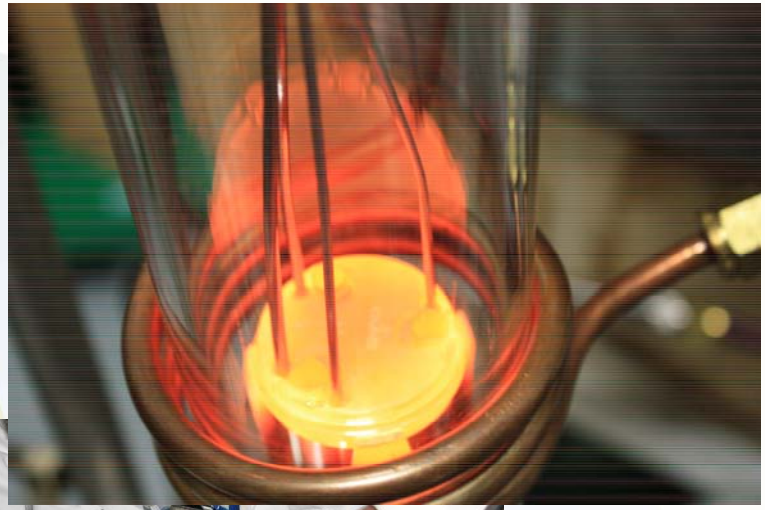
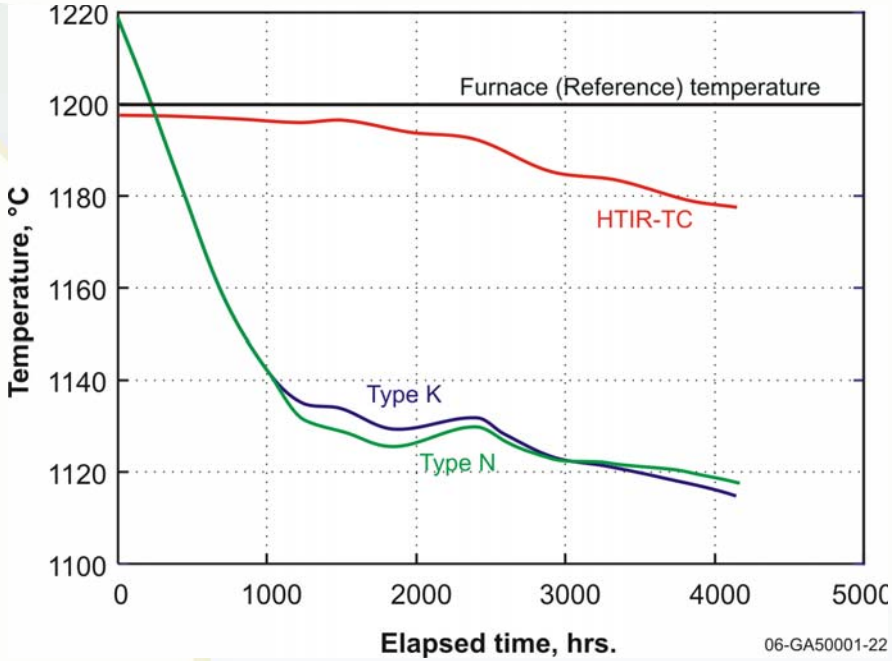
Test Train Assembly

- Induction brazing of instrument leads (e.g. thermocouples) to penetrate capsule boundaries
- Electro-plating, typically in support of induction brazes
- Thermocouple potting and splicing
- Heat treatment and (specimen) vacuum drying ovens
- Welding - Standard GTAW and MIG, tube welder
- Pressure and helium leak testing



Experiment Measurement Capabilities

- Flux Wires
  - Co-Al
  - Fe-Ni
  - Others
- Thermocouples
- Tritium Monitors
- Moisture Monitors
- Fission Product Monitors
- Specimen Creep
  - Tension (in development)
  - Compression
- Oxidation Measurements on Graphite Experiment



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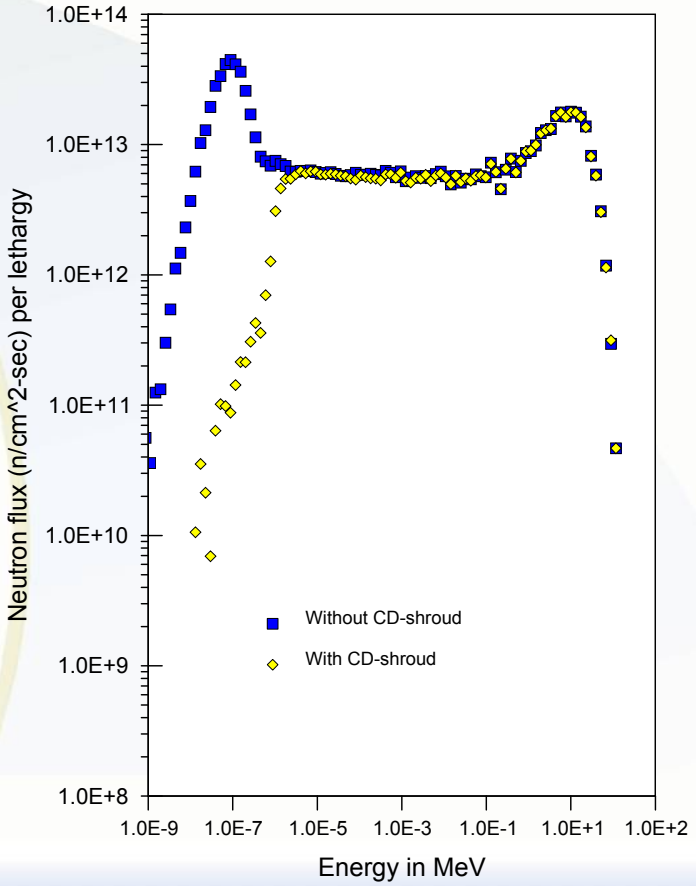
Previous Research Programs in the ATR

- Naval Reactors Fuels and Materials
- Material and Fuels for High Temperature Gas Reactor in 1990's
- Graphite Oxidation and Aging Studies for Magnox
- Pu-238 Production Studies
- Weapons Grade Mixed Oxide Fuel
- Reduced Enrichment for Research and Test Reactors (RERTR) – High Density, Low Enrichment Fuel
- Plant Maintenance Technology & Welding of Irradiated Materials
- Isotope Production

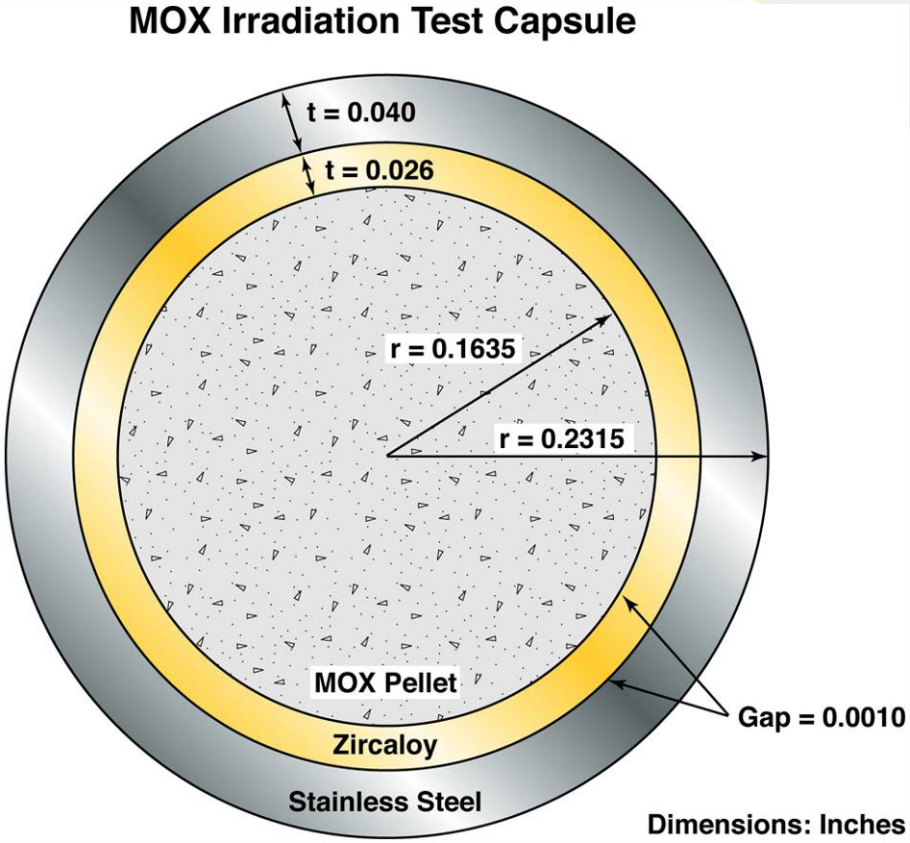
AFCI Flux Spectra with Cadmium Sleeve

- Hard Spectrum Achieved in ATR by Use Of .045 inch Thick Cadmium
- > 97% of Thermal Flux is Removed

	Thermal neutron flux (E < 0.625 eV) n/cm^2-sec	Fast neutron flux (E > 1.0 MeV) n/cm^2-sec
With CD-shroud	8.46E+12	9.31E+13
Without CD-shroud	3.71E+14	9.39E+13
Ratio	2.28%	99.14%
Note: the flux tallies are normalized to a E-lobe power of 22 MW.		



Mixed Oxide (MOX) Fuel Irradiation



- Purpose: Obtain Mixed Oxide Fuel (MOX) fuel and cladding irradiation performance data on fuel pins made with weapons grade plutonium
- PWR temperature at surface of fuel pin cladding
- Linear heat rate requirements
  - 6 KW/ft minimum
  - 10 KW/ft maximum
- Capsule designed to ASME Section III Class 1 requirements
- Small (0.001”) insulating gas gap between fuel pin and capsule provided desired temperatures

MOX Test Fuel Pellets

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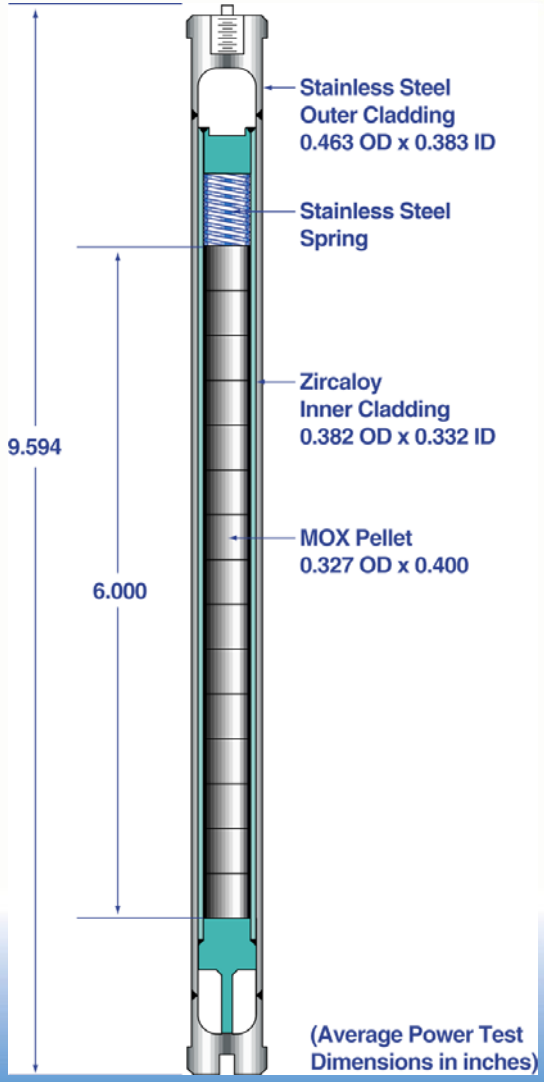

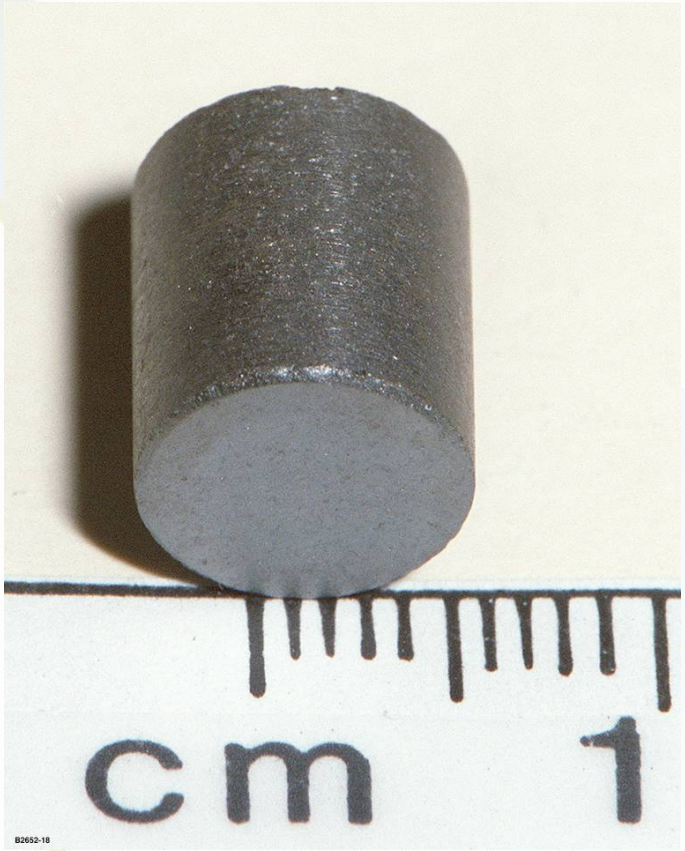

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Current ATR Irradiation Projects

- Naval Reactors Materials
- Advanced Fuel Cycle Initiative (AFCI)
- NGNP, Particle Fuel and Graphite
- NSUF University Materials, Starting Fuels
- Tritium Barrier Material
- RERTR Fuel
- LWR Fuel (in development)
- Cobalt-60
- Zirconium
- Gd-153

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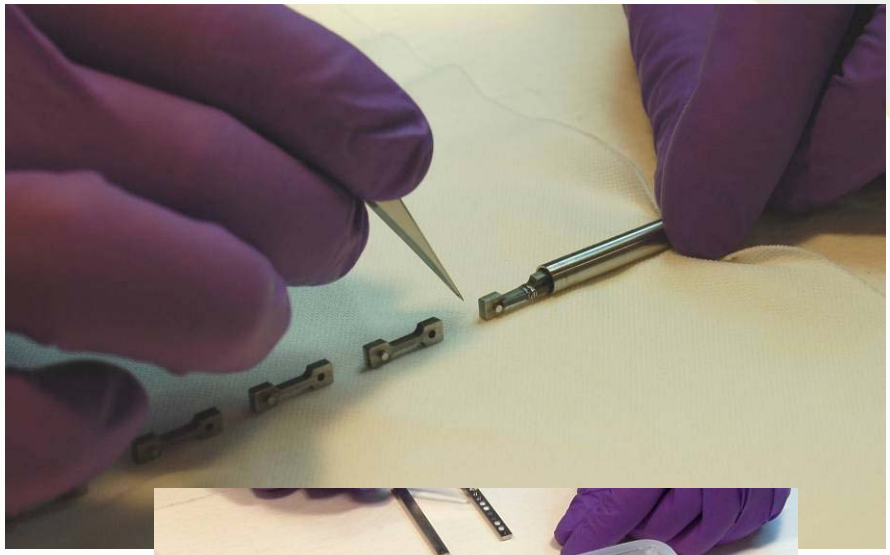
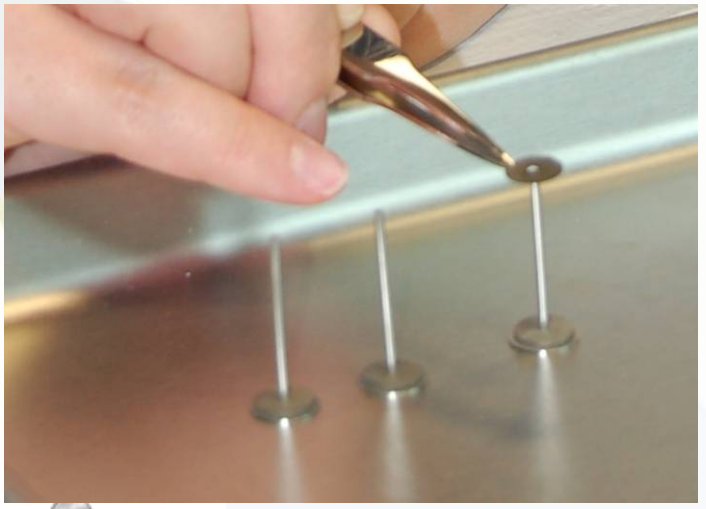
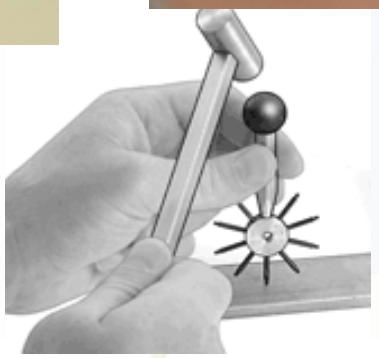
NSUF Sample Loading

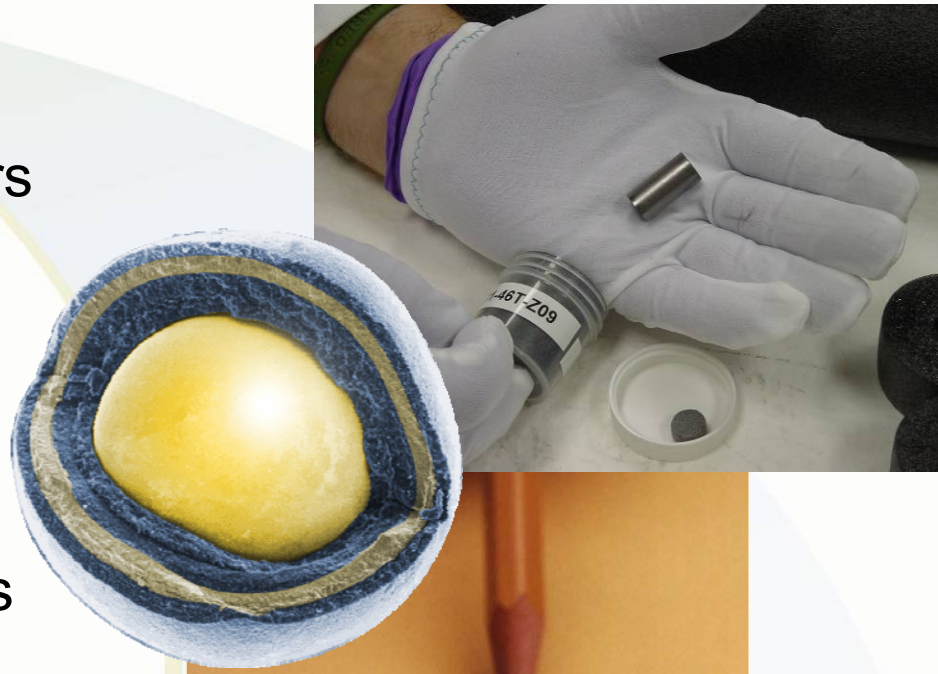






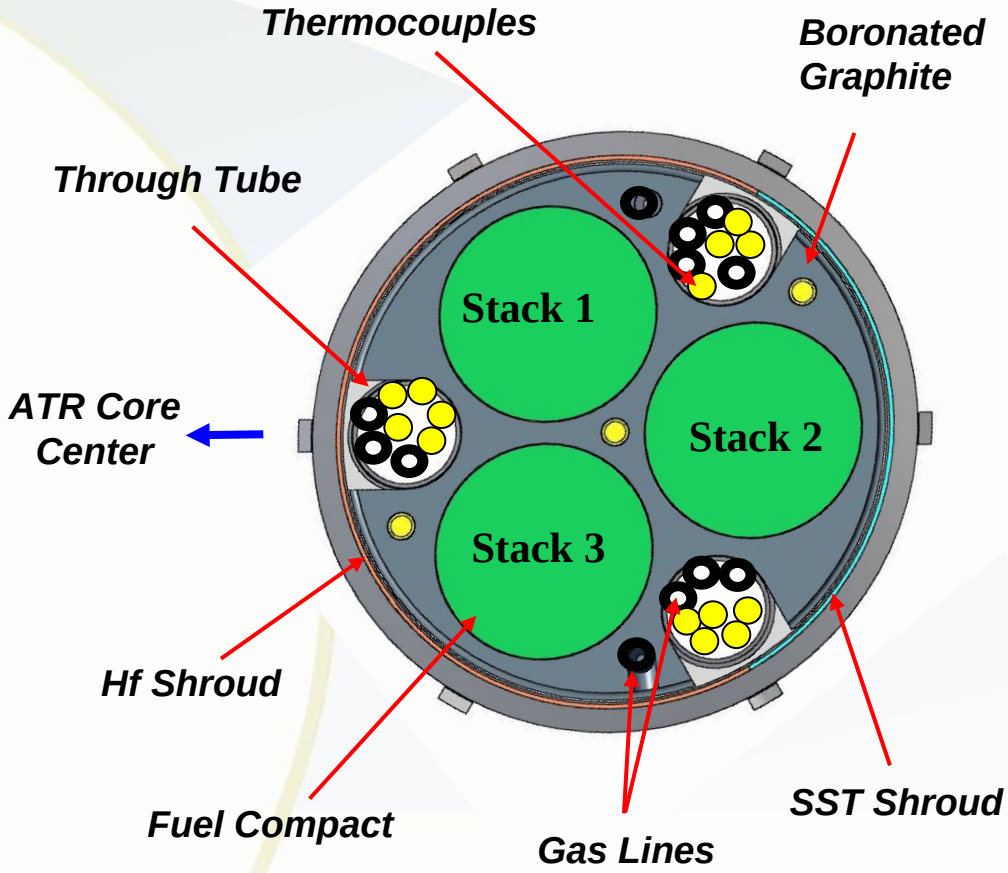
AGR-1 Fuel & Irradiation Requirements

- Purpose is to support development of next generation Very High Temperature Reactors
- TRISO-coated, Uranium Oxycarbide (UCO)
- Low Enriched Uranium (LEU), <20% enrichment
- Particle dimensions
  - 350 µm diameter fuel kernels
  - 780 µm diameter particles
- Fuel compact details
  - 12.4 mm diameter
  - 25.4 mm length
- Fast neutron fluence target
  - 1.5 -5 x 10^{25} n/m2 (E>0.18 MeV)



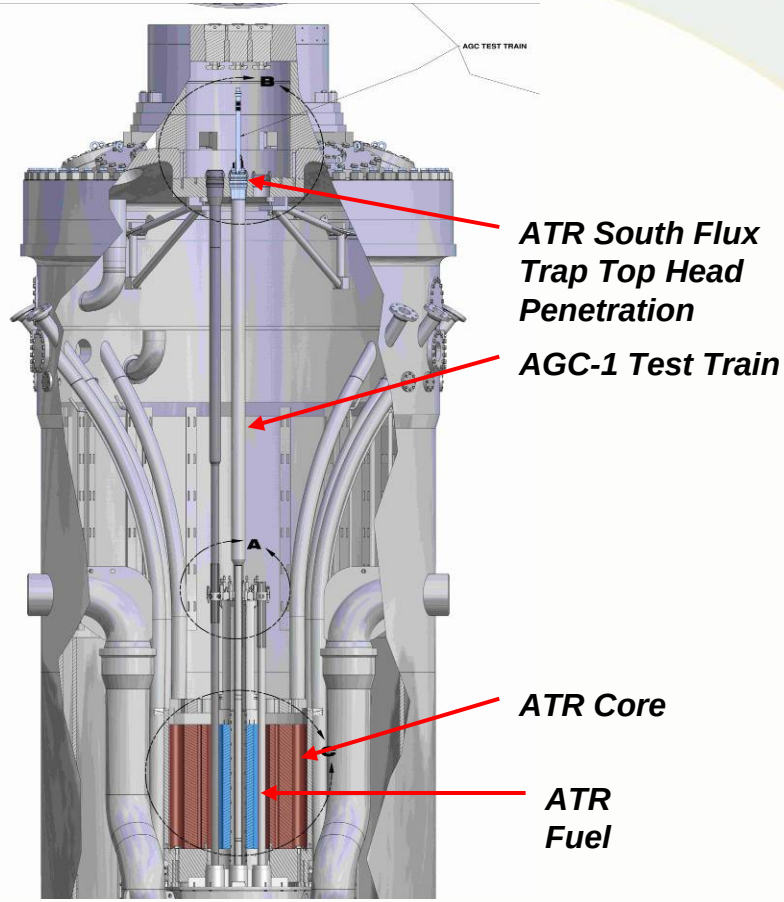
AGR-1 Capsule Design Features

- Fuel stacks
  - 3 fuel compacts/level
  - 4 levels/capsule
  - Total of 12 fuel compacts/capsule
  - Surrounded by nuclear grade graphite
- Neutron shrouds
  - Boronated Graphite
  - Hafnium shroud toward core
  - SST shroud away from core to increase flux rate to stack 2



AGR-1 Capsule Cross Section

Advanced Graphite Capsule (AGC)

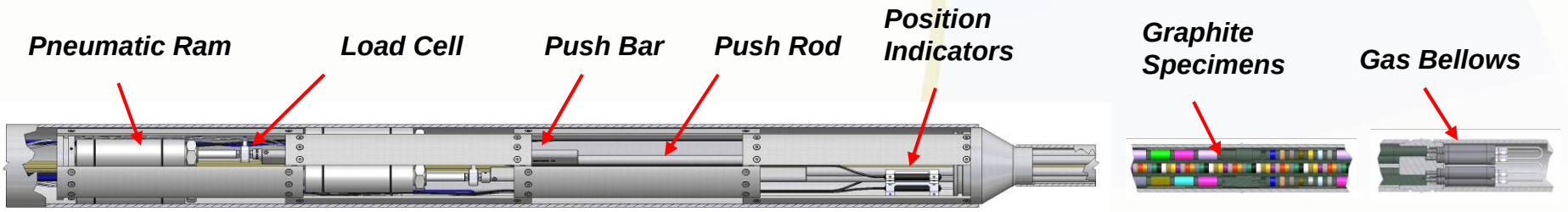


AGC-1 Experiment in ATR Reactor Vessel

- Nuclear grade graphites used in previous gas reactors are no longer available due to loss of feedstock
- AGC-1 is the first of six graphite irradiations to obtain irradiation creep data
- Experiments will be conducted at:
  - 600, 900 and 1200ºC
  - 4 to 7 dpa fast neutron damage levels (5.5 and 9.6 x 10^{21} n/cm^2 for E > 0.1 MeV)
  - Compressive loads of 2 to 3 ksi (14 to 21 MPa)

AGC-1 Compressive Load System

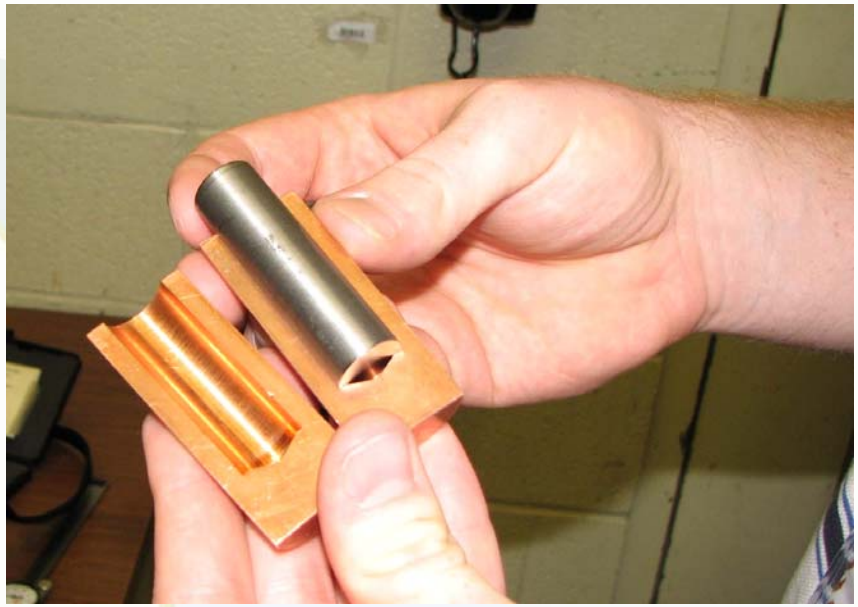
- 6 specimen stacks around capsule perimeter with compressive load on upper half of stack, center specimen without compressive load
- 6 pneumatic rams above core to provide compressive load on specimens in peripheral stacks during reactor operation
- 3 different compressive loads on the peripheral graphite stacks
- Load cells located between pneumatic rams and push bars to monitor specimen load
- Heat shield between graphite and capsule boundary to limit radiation heat transfer



AGC-1 Test Train

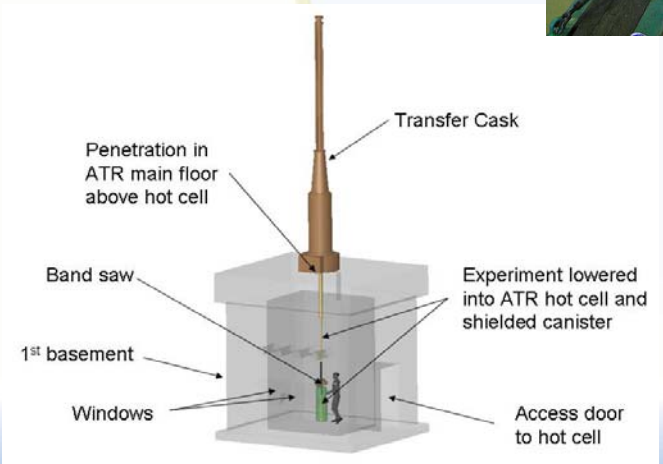
Hydraulic Shuttle Irradiation System

- 16 shuttle capsules
- Simultaneously irradiated
- Flux, at 110 MW:
  - Thermal: 2.5E14 n/cm^2-s
  - Fast (>1MeV): 8.1E13 n/cm^2-s
- Dimensions:
  - ~ 0.55” ID, ~2.1” IL
  - ~ 7 cc useable volume
- ~35 gm Contents
- Shuttle Wall Temp:
  - ~ 180 F to 240 F
- Send/receive station in ATR canal
- Utilizes ATR primary coolant to insert & remove capsules



Capsule Handling and Packaging

- Primary Shipping Container is GE-2000
- Most Experiments are Wet Loaded in ATR Canal
- New Capability to Dry Load
- INL Investigating Alternate Shipping Containers



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Future Activities for the ATR

- Next CIC tentatively scheduled for 2015
- Reactivation of Pressurized Water Loop
  - PWR testing for LWR industry
  - Possibly BWR testing
- Experiment Instrumentation Development
- Expanded Educational Opportunities
- ATR Life Extension and Modernization Projects
- Core Physics Modeling Upgrades, ATRC Testing

ATR Critical Facility

Summary

- ATR has Unique and Versatile Capabilities
  - High flux/large test volumes
  - Simultaneous tests in different testing environments – power and flux tilt
  - Constant Axial Flux Profile
- ATR Expected to Operate for Many More Years
- ATR Research Aligned with INL Facilities - Test Assembly, PIE, Analytical Labs
- INL Leads and Supports Major DOE Nuclear Missions
- More Information: www.inl.gov