ATR Reactor Capabilities Overview

Kevin Clayton
Project Management, Irradiation Testing

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Idaho National Laboratory

ATR National Scientific User Facility

ATR
National Scientific User Facility

Test and Research Reactor Utilization

- Material Degradation Studies
- Nuclear Fuel Testing and Development
- Radioisotope Production
  – Medical
  – Industrial
- Neutron/particle Beam Research
  – Physics
  – Medical
- Neutron Activation Analysis (NAA)
- Silicon Doping
- Gemstone Color Enhancement
- Education and Training

INL Idaho National Laboratory

Test Reactors, International

Reactor Name	Country	Type	Power (MW)	Flux (n/cm2-s)	Utilization
RA-3	Argentina	Pool	10	2.5E14(T)	NAA, material & fuel
OPAL	Australia	Pool	20	3E14 (T), 2.1E14 (F)	Beam science
BR-2	Belgium	Tank	100	1E15 (T), 7E14 (F)	Fuel & material, isotopes
LVR-15 REZ	Czech Republic	Tank	10	1E14 (T), 3E14(F)	Isotope, silicon, material
ETRR-2	Egypt	Pool	22	2.8E14 (T), 2.2E14 (F)	Neutron sci., fuel
OSIRIS	France	Pool	70	2.7E13 (T), 2.6E14 (F)	Material and fuel
FRM-II	Germany	Pool	20	8E14 (T), 5E14 (F)	Cold neutron, beam therapy
JOYO	Japan	Fast, Na	140	4E15 (F)	Material, fuel

Test Reactors, International (cont.)

Reactor Name	Country	Type	Power (MW)	Flux (n/cm2-s)	Utilization
JMTR	Japan	Tank	50	4E14 (T), 4E14 (F)	Isotope, material
HANARO	S. Korea	Pool	30	4.5E14 (T), 3E14 (F)	Isotope, beam, NAA, fuel, material
HFR	Netherlands	Tank in Pool	45	2.7E14 (T), 5.1E14 (F)	Material
HBWR	Norway	Heavy Water	20	1.5E14 (T), 8E13 (F)	Material, fuel
BOR-60	Russia	Fast Breeder	60	2E14 (T), 3.5E15 (F)	Material, isotopes
MIR-M1	Russia	Pool	100	5E14(T), 3E14 (F)	Fuel, material, isotopes
SAFARI-1	S. Africa	Tank in Pool	20	2.4E14 (T), 2.8E14 (F)	Isotopes, beam, radiography

Test Reactors, United States

Reactor Name	Location	Type	Power (MW)	Flux (n/cm2-s)	Utilization
ATR	Idaho	Tank	250	1E15 (T), 5E14 (F)	Material, fuel, isotopes
HFIR	Tennessee	Tank	85	2.5E15 (T), 1E15 (F)	Isotopes, beam, fuel, material
MURR	Missouri	Tank in Pool	10	6E14 (T), 1E14 (F)	Material, silicon, isotopes, NAA, gemstones
MITR	Massachusetts	Tank	5	5E13 (T), 1.7E14 (F)	Material, beam, NAA, silicon
NBSR	Maryland	Heavy Water	20	2E14 (T)	Neutron Scattering, beam

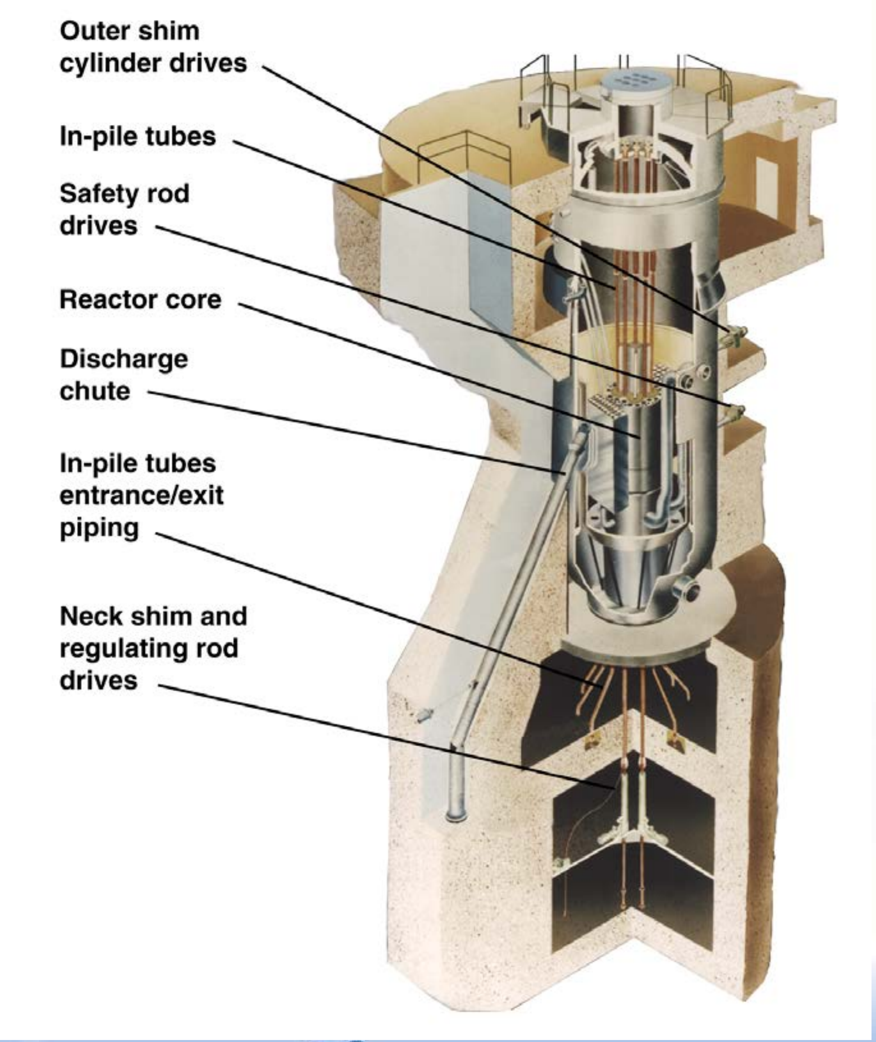
ATR Description

Reactor Type
Pressurized, light-water moderated and cooled; beryllium reflector
250 MWt design

Reactor Vessel
12 ft (3.65 m) diameter cylinder,
36 ft (10.67 m) high stainless steel

Maximum Flux, at 250 MW
1×10^{15} n/cm^2-sec thermal
5×10^{14} n/cm^2-sec fast

Reactor Core
40 fuel assemblies
U-Al plates – 19/assembly

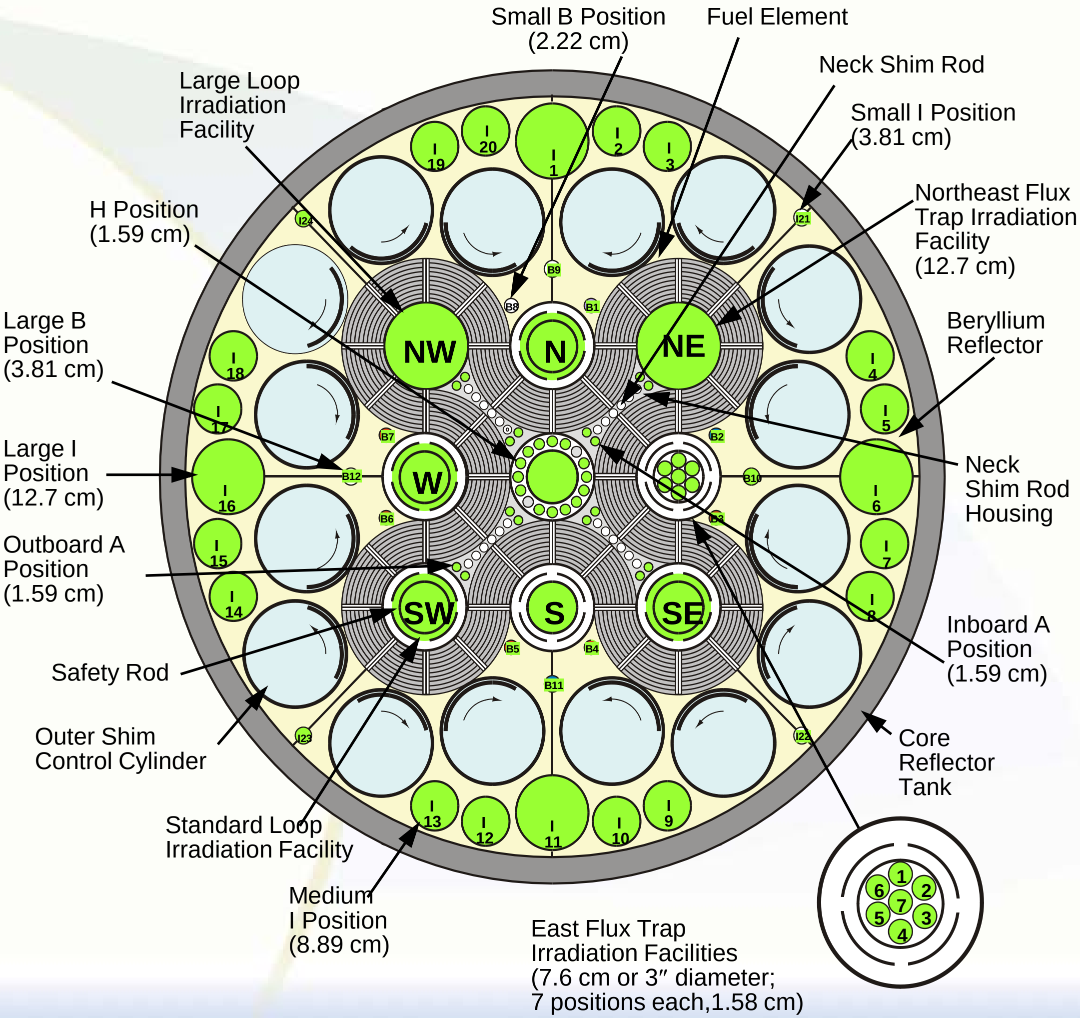


Comparison between ATR and Power PWR

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

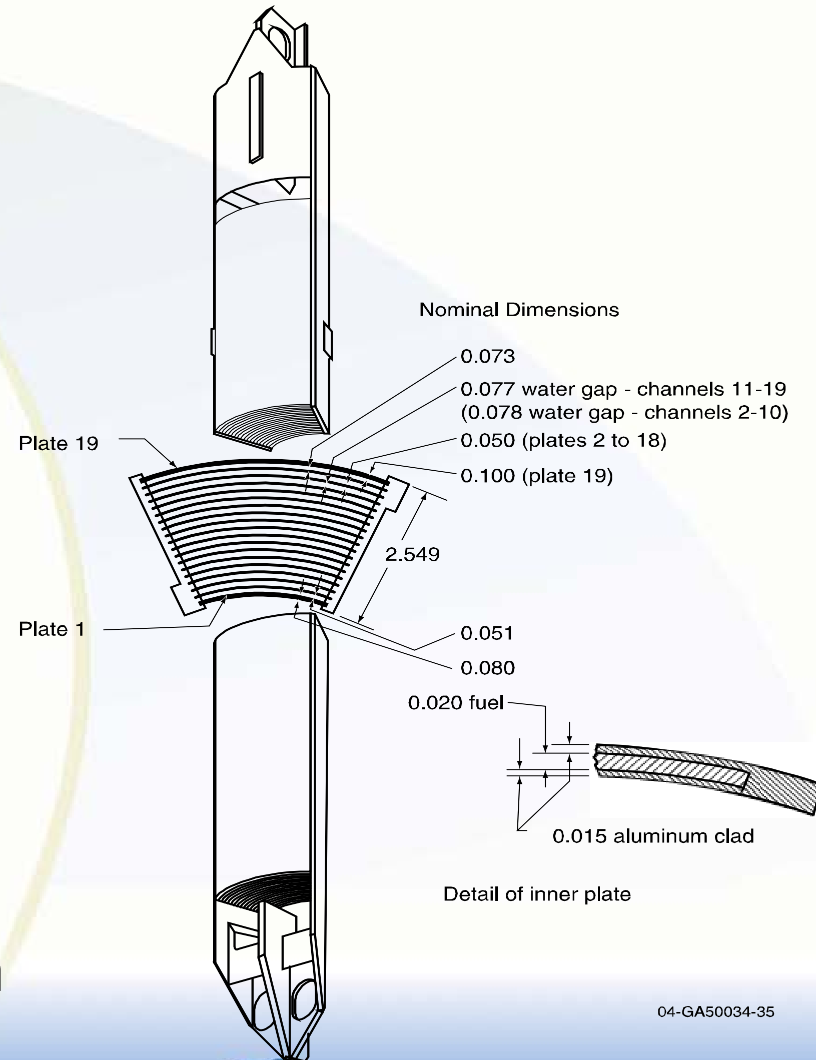
ATR Test Positions

- Test size – 122 cm length, 1.3 to 13 cm diameter
- 77 Irradiation Positions
- Rotating Hafnium Control Cylinders – symmetrical axial flux
- Power/Flux Adjustments (“Tilt”) across the Core - ≤ 3:1 ratio



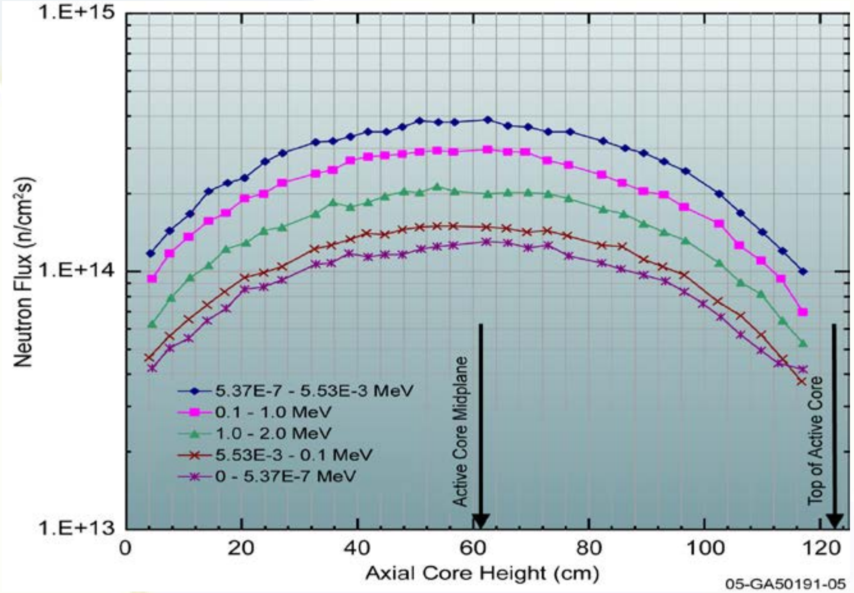
ATR Fuel Element Design

- Fuel plates contain HEU
- Fuel plates consist of UAl_x intermetallic powder dispersed in Al
  - high power density, high burnup
  - accommodate fission products in deliberate voidage
  - tolerance for fission gas
- Small amounts of natural B_4C powder included as burnable poison
  - minimize flux peaking, control reactivity
- Thin, plate-type fuel elements with narrow cooling channels maximize heat transfer



Unique Design & Experiment Features

- Combination of high flux and large test volumes
- Symmetrical axial power profile
- 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
- Flux Tailoring



Center Flux Trap Flux Profile (125 MW)

Experiment Configurations

Simple Static Capsules

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

Instrumented Lead Experiments

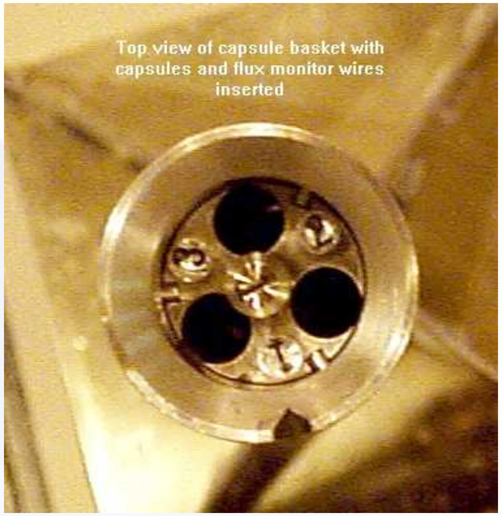
- On-line experiment measurements
- With or without temperature control
- Structural materials, cladding, fuel pins

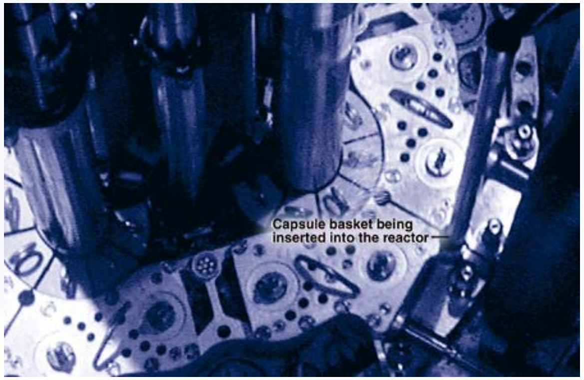
Pressurized Water Loops

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

Hydraulic Shuttle Irradiation System

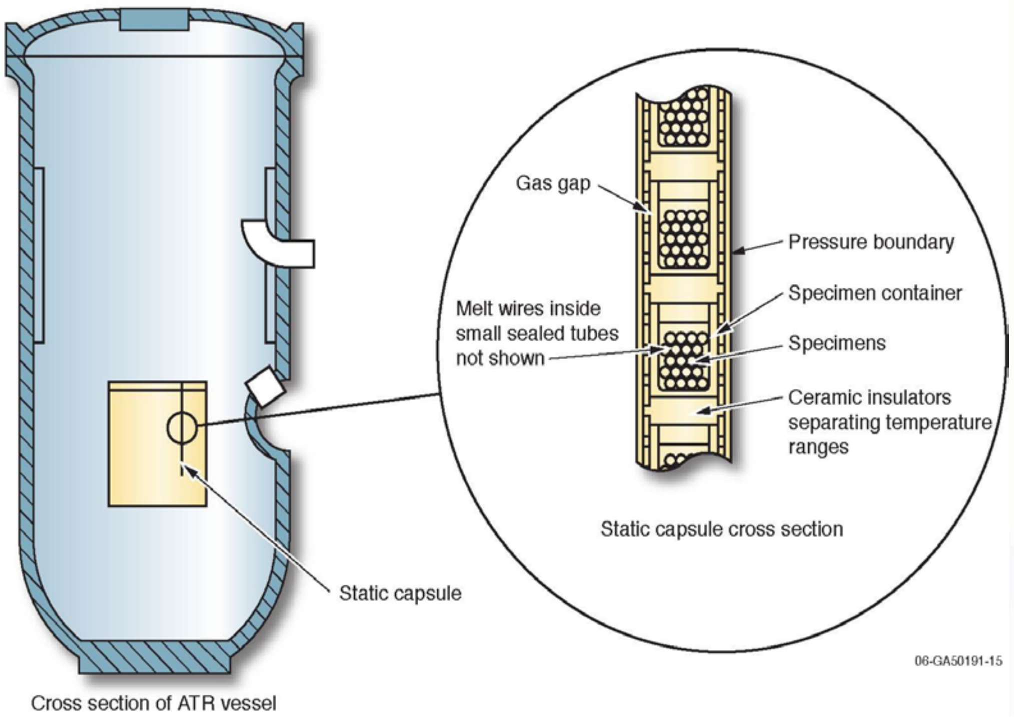
- 14 capsules in a set
- Inserted and removed during reactor operations



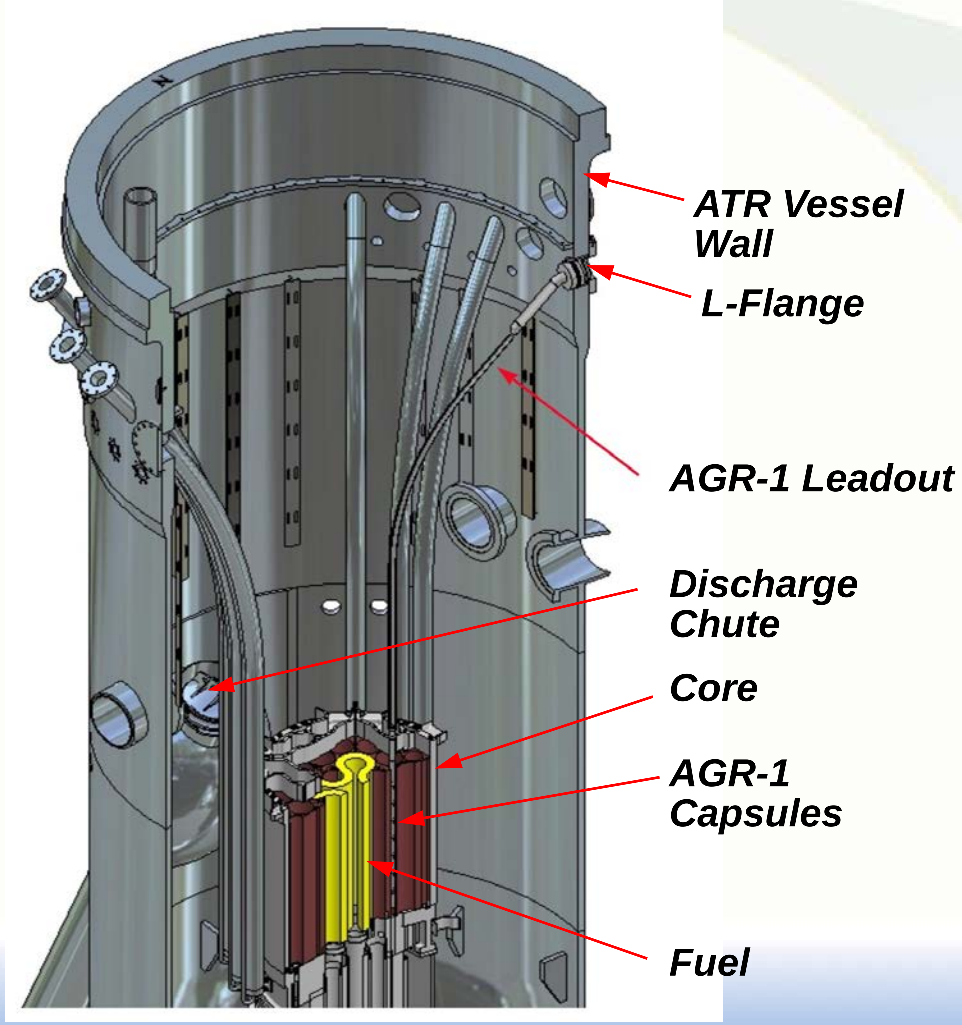


Simple Static Capsule Experiments

- 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
- Lengths up to 122 cm; diameter 1.3 – 13 cm
- Used for isotope production, fuel and material testing



Instrumented Lead Experiments

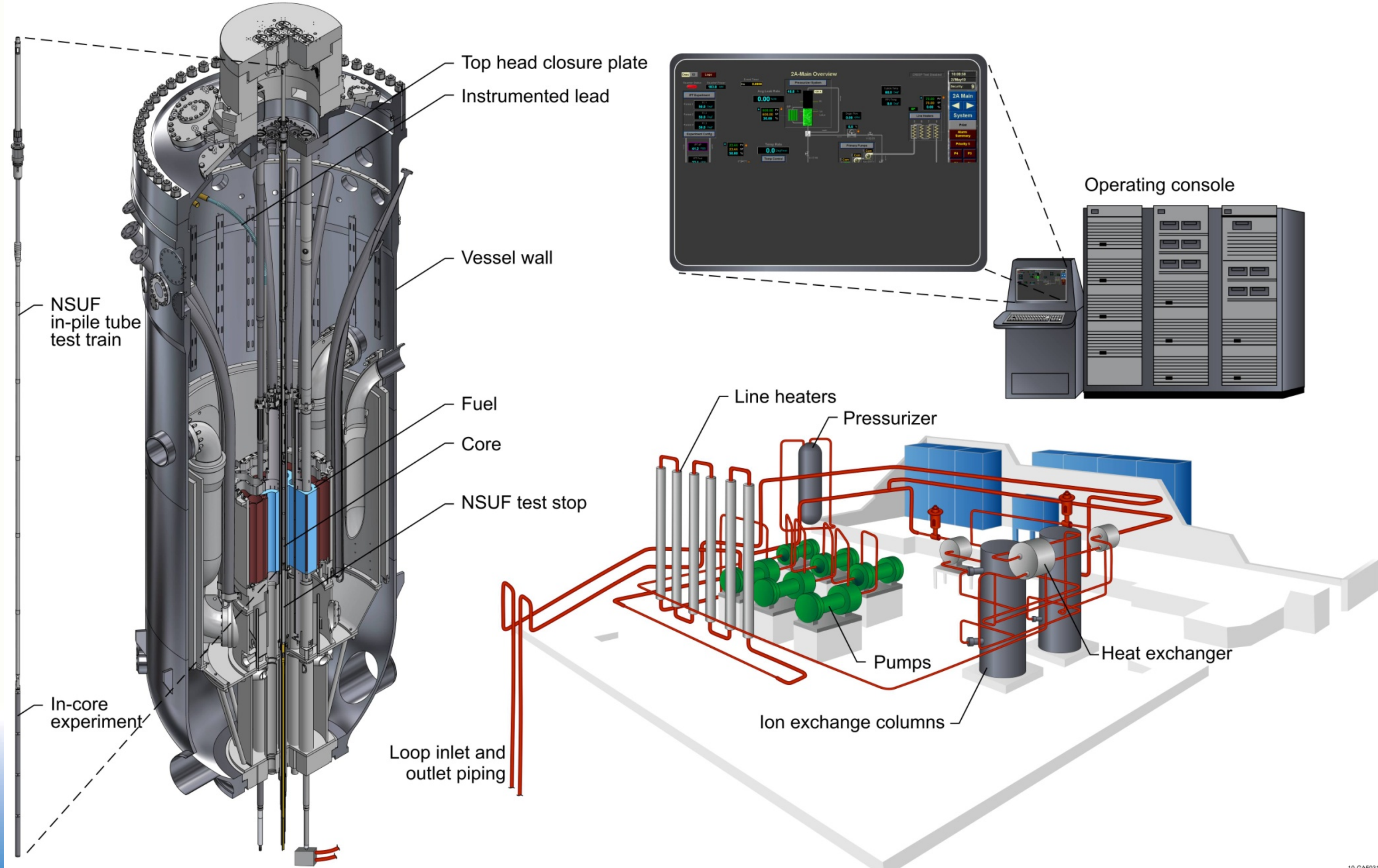


- On-line experiment measurements
- Temperature control range 250-1200°C, within +/- 15°C
- Monitoring of temperature control exhaust gases for experiment performance (e.g., fission products, leaking materials, etc.)
- Specialized gas environments (oxidized, inert, etc.)

Pressurized Water Loop Tests

- Five flux trap positions currently have pressurized water in-pile loop tests
  - 1 large diameter, 4 small diameter
  - New loop to be operational in spring 2012
- 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

PWR Loop Layout



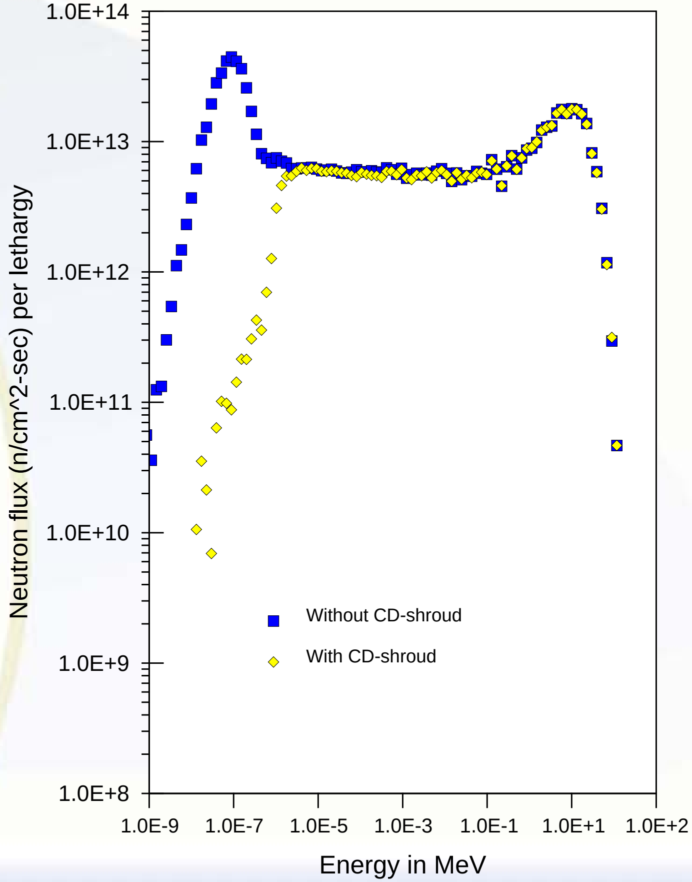
Previous Research Programs in the ATR

- 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

AFCI Flux Spectra with Cadmium Sleeved Basket

- Hard Spectrum Achieved in ATR by Use Of .114 cm 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.		



Magnox Graphite Irradiation

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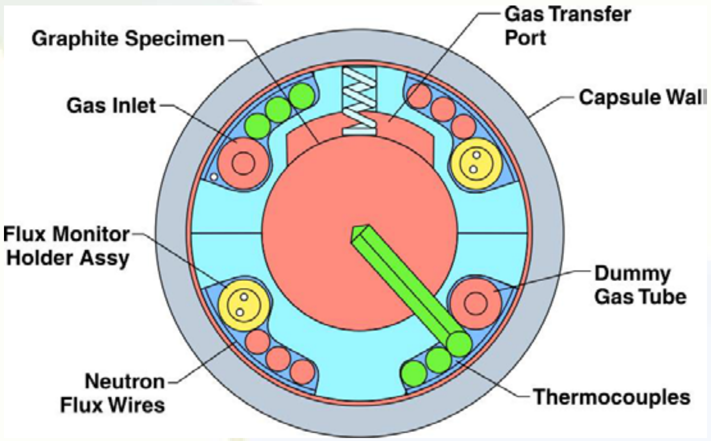
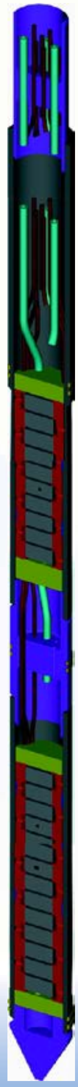
- Experiment Purpose - Extend data base on Magnox graphites for life extension support for UK Magnox power stations
- On-line temperature indication and control
- Two equal size capsules - one oxidizing & one inert, mirror images about ATR core centerline
- Inert Capsule
  - 99.996% pure helium (< 1 ppm O_2)



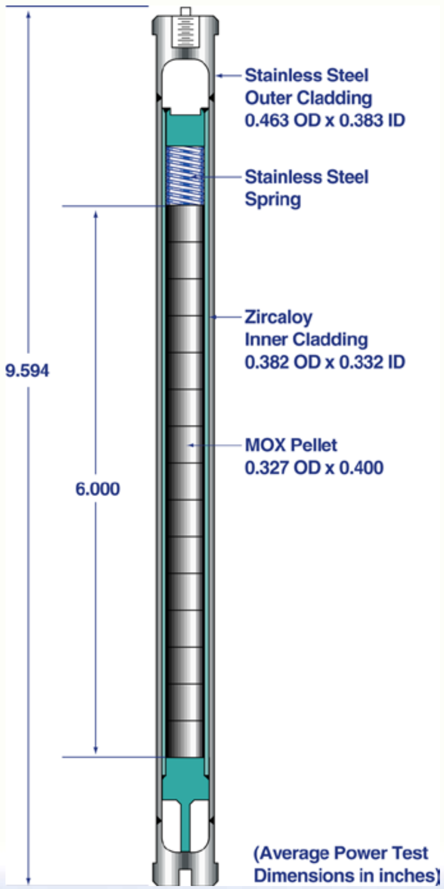
Capsule Cross Section

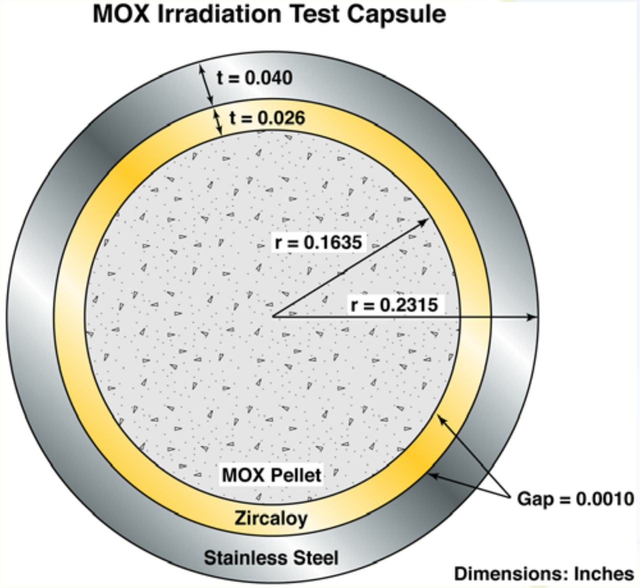


Vertical Section

Mixed Oxide (MOX) Fuel Irradiation

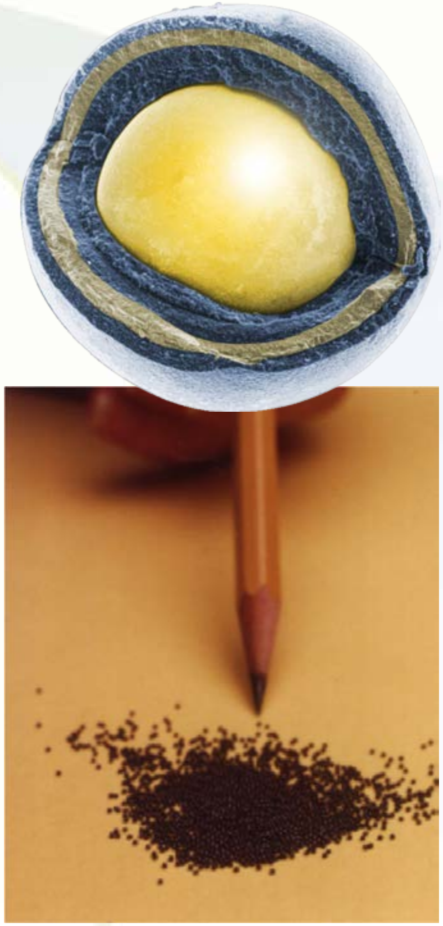
- Purpose: Obtain Mixed Oxide Fuel (MOX) fuel and cladding irradiation performance data on fuel pins made with weapons grade plutonium
- PWR temperature at surface of fuel pin cladding

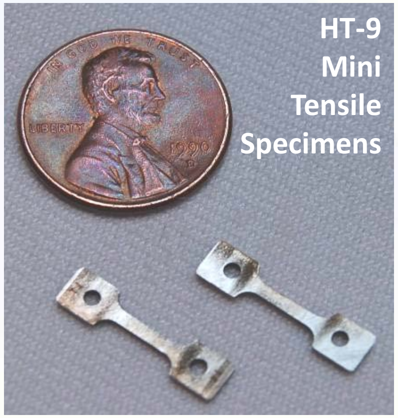




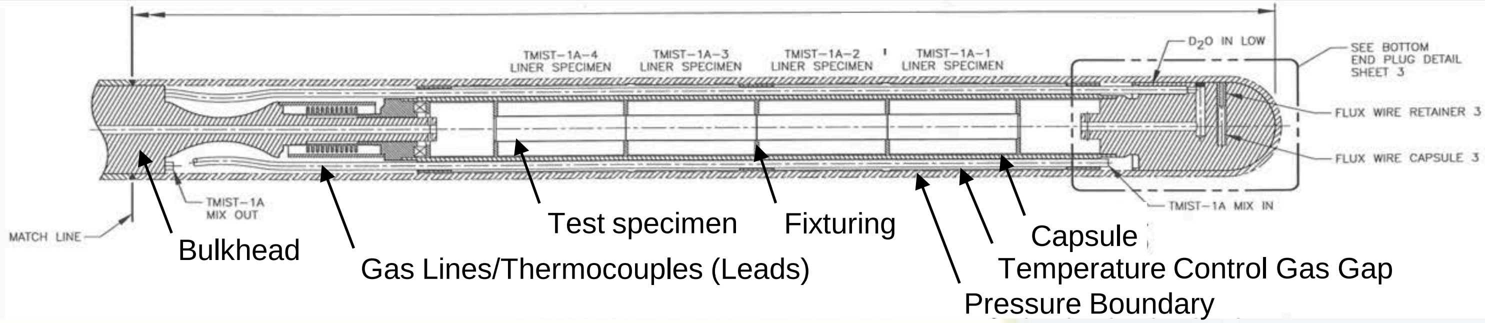
Current ATR Irradiation Projects

- Naval Reactors Materials
- Tritium Readiness Subprogram
- Advanced Fuel Cycle Initiative (AFCI)
- Next Generation Nuclear Plant, Fuel and Graphite
- National Scientific User Facility (NSUF), Fuels and Materials
- Reduced Enrichment for Research and Test Reactors (RERTR), Fuel
- Light Water Reactor (LWR) Materials
- Gadolinium-153
- Cobalt-60
- Zirconium





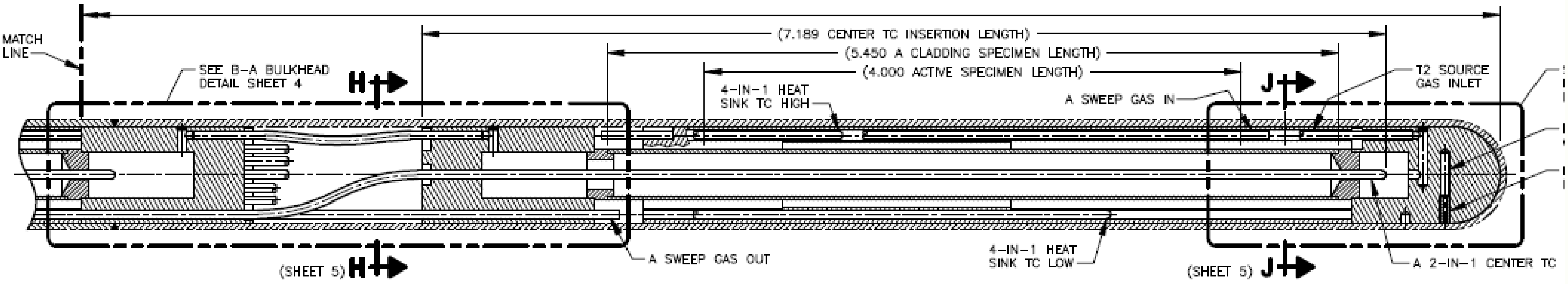
TMIST Liner Oxidation Capsules



Oxidation of Zircaloy in water vapor

- Four separate capsules
- Four Specimens per capsule
- Four independent temperature control systems
- Two independent control environments

TMIST Tritium Permeation Capsule

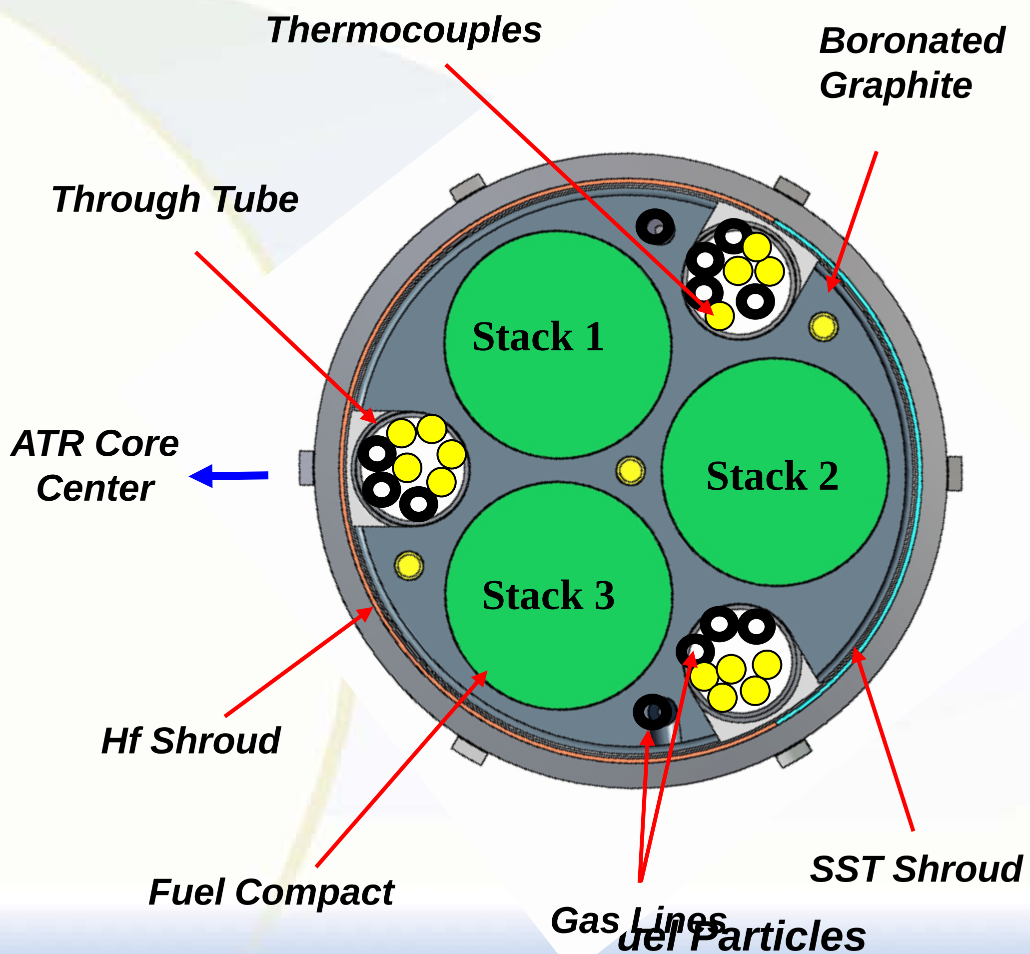


Permeation of tritium through stainless tubing

- Four separate capsules
- Four Specimens
- Four independent temperature control systems
- Two independent control environments
- Real time tritium permeation measurements

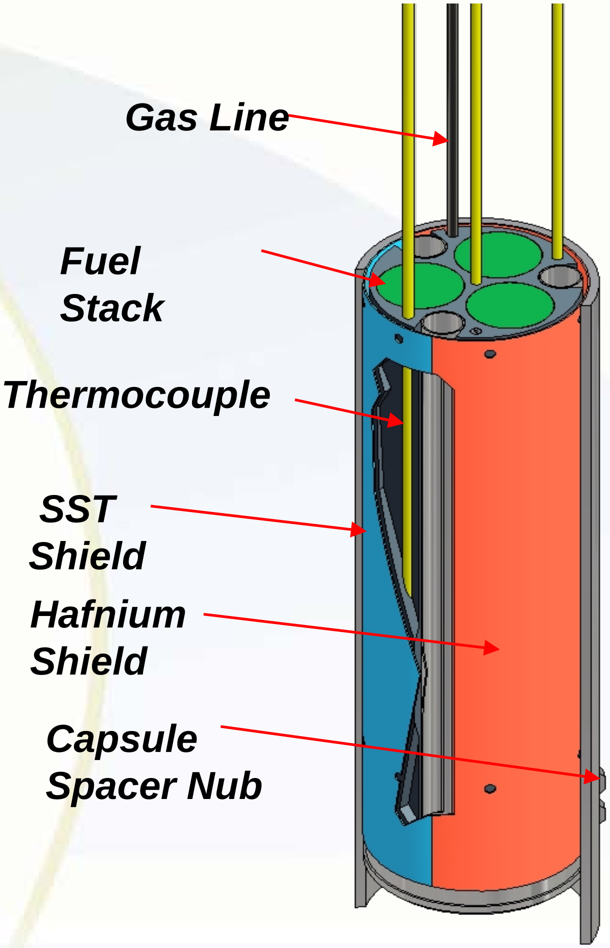
AGR-1 Fuel & Irradiation Requirements

- Program objective is to support development of next generation Very High Temperature Reactors - near term for the Next Generation Nuclear Plant
- AGR-1 Test Objectives
  - Shakedown of test design prior to fuel qualification tests
  - Irradiate early fuel from laboratory scale processes
- TRISO-coated, Uranium Oxycarbide (UCO)
- Fast neutron fluence
  - 1.5 -5 x 10^{25} n/m2 (E>0.18 MeV)



AGR-1 Capsule Design Features

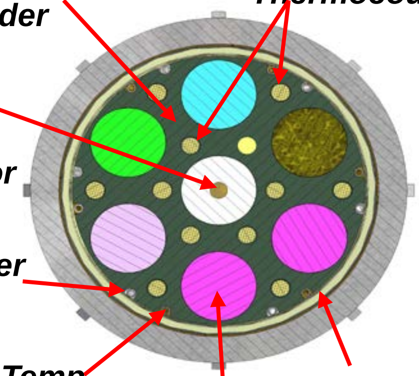
- Fuel stacks, Low Enriched Uranium (LEU), <20% enrichment,
  - 4,150 fuel particles/compact
  - 3 fuel compacts/level
  - 4 levels/capsule
  - Total of 12 fuel compacts/capsule
  - Surrounded by nuclear grade graphite
- Neutron shrouds
  - Boronated Graphite
  - Hafnium shroud toward core
  - SST shroud away from core to increase flux rate to stack 2



AGR-1 Capsule Vertical Section

AGC-1 Compressive Load System

- Nuclear grade graphites used in previous gas reactors unavailable due to loss of feedstock
- Experiments will be conducted at:
  - 600, 900 and 1200ºC
  - 4 to 7 dpa fast neutron damage levels (5.5 and 9.6×10^{21} n/cm^2 for E > 0.1 MeV)
  - Compressive loads of 2 to 3 ksi (14 to 21 MPa)
- 6 Pneumatic rams above core to provide compressive load on specimens in peripheral stacks during reactor operation



AGC-1 Capsule Cross Section

AGC-1 Test Train

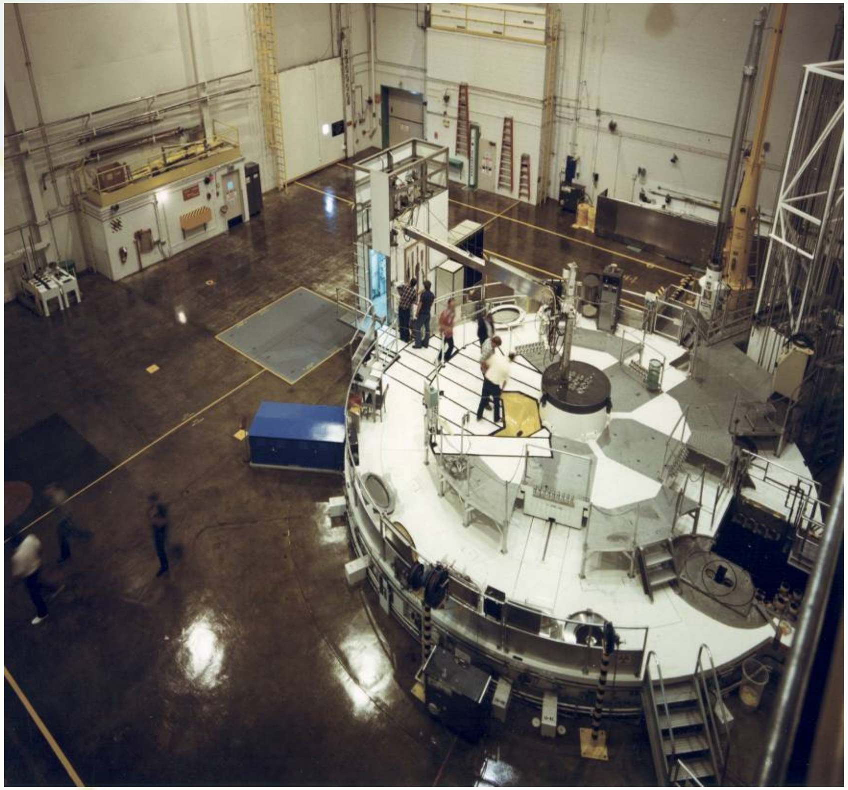
Hydraulic Shuttle Irradiation System

(aka Rabbit)

- 14 shuttle capsules in train, simultaneously irradiated
- Payload ~ 25 gm/capsule
- ^{235}U mass limit: 0.020 gm/capsule, 0.100 gm/train
- Flux (at 100 MW_t)
  - Thermal = 2.5×10^{14} n/cm^2-s
  - Fast (> 1MeV) = 8.1×10^{13} n/cm^2-s
- Gamma Heating: 4.2 W/gm for SST
- Dimensions (approximate)
  - 0.55 in. (1.4 cm) inner diameter
  - 2.0 in. (5.1 cm) inner height
  - 0.48 in.^{3} (7.8 cm^3) inner volume
- Irradiation Duration
  - 10 sec – few weeks
- Shuttle Wall Temperature
  - ~ 180°F to 240°F
    (~ 82°C to 115°C)
- Maximum Internal Pressure
  - < 215 psig (1.48 MPa)

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 230 days per year
- Currently, Standard Operation is ~110 MW
- Core Internals Changeout (CIC), every 7 to 10 years
- ATR Critical Facility – used for reactivity measurements



ATR Summary

- Wide Variety of Test Reactors
- Many Missions
- 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
  - Core internal changeout
  - Experiment capability upgrades